UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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July 27, 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: BYRON STATION, UNITS 1 AND 2 COMPONENT DESIGN BASES INSPECTION 05000454/2012007; 05000455/2012007(DRS)

Dear Mr. Pacilio:

On June 15, 2012, the U.S. Nuclear Regulatory Commission, (NRC) completed a Component Design Bases Inspection, (CDBI) at your Byron Station, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on June 15, 2012, with Mr. T. Tulon, and other members of your staff.

The 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, five NRC-identified findings of very low safety significance were identified. The findings involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy

If you contest the subject or severity of this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Byron Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at Byron Station.

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

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-454; 50-455
License Nos. NPF-37; NPF-66

Enclosure: Inspection Report 05000454/2012007; 05000455/2012007(DRS)
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000454; 05000455
License Nos:	NPF-37; NPF-66
Report No:	05000454/2012007; 05000455/2012007(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	May 14, 2012, through June 15, 2012
Inspectors:	A. Dunlop, Senior Engineering Inspector, Lead C. Brown, Engineering Inspector, Electrical E. Sanchez Santiago, Engineering Inspector, Mechanical R. Langstaff, Senior Engineering Inspector, Operations J. Chiloyan, Electrical Contractor C. Baron, Mechanical Contractor
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000454/2012007; 05000455/2012007(DRS); 05/14/2012 – 06/15/2012; Byron Station, Units 1 and 2; Component Design Bases Inspection (CDBI).

The inspection was a 3-week onsite baseline inspection that focused on the design of components. The inspection was conducted by regional engineering inspectors and two consultants. Five Green findings were identified by the inspectors. The findings were considered Non-Cited Violations (NCV) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure qualified components were installed in the plant. Specifically, purchase orders did not specify the minimum pickup voltage for NEMA Size 1 through Size 4 safety-related motor-control contactors such that the installed contactors were not rated to function at the design basis minimum voltage. The licensee entered the issue into their corrective action program and based on a sample testing of contactors demonstrated there was adequate margin between the highest found minimum-pickup voltage and the design basis pickup voltage.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating System Cornerstone attribute of Design Control, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, having installed contactors that may not function under degraded voltage conditions could affect the operability of multiple safety-related structures, systems and components during an event. The finding screened as of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(1))

- Green: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to ensure the component cooling water (CCW) system was capable of withstanding a reactor coolant pump thermal barrier break. Specifically, when assuming a single failure of the automatic isolation function, the licensee failed to evaluate the break effect on the CCW system during the 3 minutes postulated to isolate the leak. The licensee entered the issue into their corrective action program; verified the CCW system would be able to withstand the postulated event, and planned to perform a detailed evaluation of the effect of a thermal barrier break on the CCW system.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Design Control and objective of ensuring the capability of the system to respond to an initiating event to prevent undesirable consequences. Specifically, the failure to evaluate the effect of the thermal barrier rupture on the CCW system created reasonable doubt whether the system would be capable of withstanding the applied forces of this event. The finding screened as very low safety significance (Green) because the design deficiency did not result in a loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(2))

- Green: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to specify in a design calculation the allowable relay setpoint calibration tolerances. Specifically, the acceptance criteria used in relay setting calibration procedures was not bounded by the relay setting design calculations. The licensee entered this finding into their corrective action program and verified the calibrated relay settings would still provide adequate electrical protection coordination capability.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to adequately evaluate the design requirements of the relay settings could have resulted in a loss-of-relay coordination and could allow a fault on one piece of equipment to propagate to other safety-related equipment outside the designed isolation boundary. The finding screened as very low safety significance (Green) because the finding was design deficiency confirmed not to result in a loss of operability or functionality. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3.b.(3))

- Green: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to adequately analyze potential design basis internal flooding events in the auxiliary building. Specifically, the licensee's analysis did not account for the possible single failure of an essential service water motor-operated isolation valve or its associated power supply, which would have prevented break isolation within 30 minutes. The licensee entered the issue into their corrective action program; verified essential service water piping in the auxiliary building would meet the "crack exclusion" pipe stress criteria, and planned to the revise the flooding analysis.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Design Control and objective of ensuring the capability of the system to respond to an initiating event to prevent undesirable consequences. Specifically, the failure to adequately analyze potential design basis internal flooding events in the auxiliary building would affect the capability of safety-related equipment to withstand the postulated event. The finding screened as very low safety significance (Green) because the design deficiency did not result in a loss of operability or functionality. The inspectors did not identify a

cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.4.b.(1))

Cornerstone: Barrier Integrity

- Green: The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to provide a means to detect and isolate a leak in the emergency core cooling flow path within 30 minutes, which was contrary to the Updated Final Safety Analysis Report. Specifically, the licensee failed to provide a means to detect and isolate a leak within 30 minutes in that neither sump alarms nor radiation monitors were provided for the safety injection pump rooms. The licensee entered the issue into their corrective action program and planned to evaluate options for modifications to address detection of emergency core cooling system leakage.

  The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity cornerstone attribute of Design Control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to provide a means to detect and isolate a leak in the emergency core cooling flow path within 30 minutes could result in a delayed isolation of such a leak after an accident and result in a greater radionuclide release to the auxiliary building and the environment. The finding screened as very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.6.b.(1))

B. Licensee-Identified Violations

No violations were identified.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify that design bases have been correctly implemented for the selected risk significant components and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The Probabilistic Risk-Assessment (PRA) model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for which there are no indicators to measure performance.

Specific documents reviewed during the inspection are listed in the Attachment to the report.

.2 Inspection Sample Selection Process

The inspectors used information contained in the licensee's PRA and the Byron Station Standardized Plant Analysis Risk-Model to identify a scenario to use as the basis for component selection. The scenario selected was a reactor coolant pump loss of seal cooling event. Based on this scenario, a number of risk significant components were selected for the inspection.

The inspectors also used additional component information such as a margin assessment in the selection process. This design margin assessment considered original design reductions caused by design modification, power uprates, or reductions due to degraded material condition. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as performance test results, significant corrective actions, repeated maintenance activities, Maintenance Rule (a)(1) status, components requiring an operability evaluation, NRC resident inspector input of problem areas/equipment, and system health reports. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

The inspectors also identified procedures and modifications for review that were associated with the selected components. In addition, the inspectors selected operating experience issues associated with the selected components.

This inspection constituted 24 samples as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TS), design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code, Institute of Electrical and Electronics Engineers (IEEE) Standards and the National Electric Code, to evaluate acceptability of the systems' design. The NRC also evaluated licensee actions, if any, taken in response to NRC issued operating experience, such as Bulletins, Generic Letters (GLs), Regulatory Issue Summaries (RISs), and Information Notices (INs). The review was to verify that the selected components would function as designed when required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability was consistent with the design bases and was appropriate may include installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, preventive maintenance activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, and licensee corrective action program documents. Field walkdowns were conducted for all accessible components to assess material condition and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 18 components were reviewed:

- Residual Heat Removal (RHR) Pump (1RH01PA): The inspectors reviewed design analyses associated with RHR pump capacity, net positive suction head (NPSH), and minimum flow to verify the equipment's capacity to perform its required functions. The pump test procedures and recent results were reviewed to verify the actual capability of the installed equipment. The inspectors reviewed industry experience issues associated with the pump thrust bearing loading to verify appropriate maintenance intervals were being maintained. The inspectors reviewed industry experience associated with operation of the RHR pump under recirculation conditions for extended periods of time to verify this operation would not result in unacceptable RHR system pressures and temperatures. The potential susceptibility of the pump to internal flooding events was reviewed to verify the capability of the pump to perform its required function. The inspectors reviewed a sample of operating procedures associated with the pump under normal and accident conditions.

- RHR Pump Recirculation Valve (1RH611): The inspectors reviewed motor-operated valve (MOV) calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, degraded voltage, and valve weak link analysis. Diagnostic testing and inservice testing (IST) surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified. The setpoint calibration was reviewed to ensure the valve would open to provide a minimum flow path for the RHR pump and close to ensure design flow requirements were met. Short-circuit calculations, breaker interrupting ratings, thermal overload (TOL) sizing calculations, TOL rating, voltage drop calculations, MOV motor terminal voltage requirements, MOV control circuit requirements and logic were reviewed to ensure the valve would function as required.

- Essential Service Water (SX) Pump (1SX01PA): The inspectors reviewed design analyses associated with SX pump capacity and NPSH to verify the equipment's capacity to perform its required functions. The inspectors also reviewed test procedures and recent results to verify the actual capability of the installed pump. The inspectors reviewed the performance changes associated with replacement of pump rotating elements, including the potential effects on the pump motors and electrical distribution system. The potential susceptibility of the SX pump to internal flooding events was reviewed to verify the capability of the pump to perform its required function after any design basis event. The inspectors reviewed a sample of operating procedures associated with the pump under normal and accident conditions. The inspectors also reviewed the motor feeder circuit sizing, to ensure adequacy of ampacity, short circuit current capability, and voltage requirements at the motor terminals under the most limiting conditions. Electrical calculations were also reviewed to ensure the adequacy of the motor feeder circuit phase and ground protective device trip settings.

- SX Strainer (1SX01FA): The inspectors reviewed the capability of the SX strainer to perform its required functions. The inspectors reviewed inspection procedures and recent results to verify the actual capability of the installed strainer. Operating procedures associated with the strainer under normal and accident conditions were reviewed, including the capability of the operators to clean the strainer without normal electrical power available.

- SX Unit Cross-tie Valve (1SX005): The inspectors reviewed calculations, operations history, and design requirements to verify the equipment's capacity to perform its required functions. The inspectors reviewed the design differential pressure for this valve to verify its capability to operate under the most limiting conditions. The inspectors also reviewed MOV test procedures and recent results to verify the actual capability of the installed equipment. Short-circuit calculations, breaker interrupting ratings, TOL sizing calculations, TOL rating, voltage drop calculations, MOV motor terminal voltage requirements, MOV control circuit requirements and logic were reviewed to ensure the valve would function as required.

- Centrifugal Charging (CV) Pump (1CV01PA): The inspectors reviewed design analyses associated with CV pump capacity, NPSH, and minimum flow to verify the equipment's capacity to perform its required functions. The inspectors also reviewed test procedures and recent results to verify the actual capability of the installed pump. The "weak pump/strong pump" analysis was reviewed to verify that all operating pumps would have adequate minimum flow under all conditions. The inspectors reviewed the performance changes associated with replacement of pump rotating elements. The inspectors reviewed operating procedures associated with the pump under normal and accident conditions. The inspectors also reviewed the motor feeder circuit sizing to ensure adequacy of ampacity, short circuit current capability, and voltage requirements at the motor terminals under the most limiting conditions. Electrical calculations were reviewed to ensure the adequacy of the motor feeder circuit phase and ground protective device trip settings.

- CV Reactor Coolant Pump Seal Injection Valve (1CV8355A): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, degraded voltage, and valve weak link analysis. Diagnostic testing and IST surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified. Breaker interrupting ratings, short-circuit calculations, TOL sizing calculations, voltage drop calculations, TOL rating, MOV motor terminal voltage requirements, MOV control circuit requirements and logic were reviewed to ensure the valve would function as required.

- CV Pump Suction Check Valve (1CV8546): The inspectors reviewed maintenance history, operations history, and design requirements to verify the check valves capacity to perform its required functions. The inspectors also reviewed test procedures and recent results to verify the actual capability of the installed check valve. The inspectors reviewed an analyses associated with a material change of a valve component.

- Component Cooling Water (CCW) Pump (1CC01PA): The inspectors reviewed design analyses associated with the CCW pump capacity, acceptance criteria, and seismic evaluations to verify the equipment's capacity to perform its required functions. The inspectors reviewed operating procedures associated with the pump under normal and accident conditions. The inspectors reviewed surveillance results including IST quarterly pump testing, flow verification, and performance testing to verify acceptance criteria were met and performance degradation could be identified. The inspectors also reviewed the motor feeder circuit sizing, to ensure adequacy of ampacity, short-circuit current capability, and voltage requirements at the motor terminals under the most limiting conditions. Electrical calculations were also reviewed to ensure the adequacy of the motor feeder circuit phase and ground protective device trip settings.

- Reactor Coolant Pump Cooling from CCW Valve (1CC9413A): The inspectors reviewed MOV calculations and analysis to ensure the valve was capable of functioning under design conditions. These included calculations for required thrust, maximum differential pressure, degraded voltage, and valve weak link analysis. Diagnostic testing and IST surveillance results, including stroke time and available thrust, were reviewed to verify acceptance criteria were met and performance degradation could be identified. Breaker interrupting ratings, short-circuit calculations, TOL sizing calculations, voltage drop calculations, TOL rating, MOV motor terminal voltage requirements, MOV control circuit requirements and logic were reviewed to ensure the valve would function as required.

- CCW Heat Exchanger (1CC01A): The inspectors reviewed the design and licensing basis of the heat exchanger to ascertain compliance with system operation requirements. The inspectors reviewed calculations to determine fouling criteria to ensure the heat exchanger was meeting its design requirements and the results of the calculations were properly translated into the procedure acceptance criteria. The inspectors also reviewed test procedure for appropriate acceptance criteria; including the testing and inspection results to verify compliance with heat exchanger program requirements.

- CCW Surge Tank (1CC01T): The inspectors reviewed design analyses associated with the surge tank capability to perform its required functions. The inspectors also reviewed the capacity and level set points for the tank to ensure compliance with design requirements. Internal/external inspection results were reviewed to ensure the integrity of the tank was being monitored and maintained. The inspectors reviewed minimum and maximum temperature limits as well as refill capability and overpressure calculations to ensure the tank was capable of performing its intended safety function under different operating scenarios.

- CCW Heat Exchanger Outlet Header Crosstie Isolation Valve (1CC9467B): The inspectors reviewed the design and licensing basis of the valve to ascertain compliance with system design function. The inspectors reviewed testing results to ensure compliance with the IST program. Operating procedures were reviewed to verify capability to operate the valve under various scenarios.

- Unit Auxiliary Transformer (UAT) 141-1: The inspectors reviewed transformer windings configuration, circuit breaker control schematics, protective relay settings, and surveillance test records to assess the status and maintenance condition of the transformer. The inspectors reviewed the electrical distribution system calculations and performed independent calculations using the transformer nameplate data to determine the adequacy of the transformer to supply required power to the associated 4160 Vac switchgear and to verify short circuit current interrupting duty requirements were within the switchgear breaker ratings. To determine whether the 6.9kV and 4.16kV neutral grounding resistors were adequately maintained, the inspectors reviewed the completed preventive maintenance test procedures to verify that the measured resistance values of the grounding resistors remained within manufacturer's tolerances.

Emergency Diesel Generator (EDG) (1DG01KA): The inspectors reviewed the load voltage drop calculation, maximum and minimum voltage profile, and DC field flashing circuit design. The inspectors also verified that the EDGs start properly under degraded voltage conditions. Surveillance test results were reviewed to ensure TS requirements were met. The inspectors reviewed the adequacy and appropriateness of design assumptions and calculations related to EDG protection and relay coordination during test mode and during emergency operation. The EDG output breaker control logic diagrams were reviewed to verify the breaker tripping and closing logic was consistent with design basis description and interlocking requirements. The inspectors reviewed the adequacy of the EDG's high resistance neutral grounding equipment and whether appropriate periodic maintenance and measurements were performed to ensure the design basis of the ground fault detection was maintained. The inspectors performed independent calculations of available fault current contributions from the EDG, UAT, and SAT (system auxiliary transformer) for postulated phase and ground faults on safety Bus 141 and compared them with the relay setting calculations to verify the appropriateness of the applied over current and voltage relay settings.

- Switchgear Bus 141: The inspectors reviewed circuit breaker control schematics, protective relay settings, loss of voltage and degraded voltage relay settings, and electrical distribution system calculations to assess the status and maintenance condition of the equipment and to verify the adequacy of bus and circuit breaker load capacity and short circuit interrupting ratings for full loading and emergency loading. Operating and maintenance test procedures were reviewed to assess whether component operation and alignments were consistent with design and licensing bases assumptions. The switchgear protective relay testing procedures and recently completed calibration test results were reviewed to verify the acceptance criteria for tested parameters were supported by calculations, or other engineering documents. The inspectors performed independent calculations of available fault current contributions from the EDG, UAT, and SAT for postulated phase and ground faults and compared them with the phase and ground over-current relay setting calculations to verify the appropriateness of the applied over-current relay settings. The inspectors also reviewed the 4160 Vac Bus 141 loss of voltage and bus over-current relay settings to ensure adequate coordination was maintained between the bus over-current and the bus undervoltage relay settings to ensure the over-current relays function as designed during postulated electrical bus faults.

- 480 Vac ESF Bus 131X: The inspectors reviewed bus design capability, bus and load breaker protection settings, control circuits for adequate power, electrical separation/isolation, seismic qualifications, and degraded voltage effects on 480/120 Vac motor control relays to ensure conformance with applicable design standards.

- 120 Vac Instrument Bus 111: The inspectors reviewed bus design capability related to loading and short circuit protection, breaker ratings and settings to prevent spurious tripping and protection against low magnitude faults, breaker coordination, and electrical separation/isolation to ensure conformance with applicable design standards. Test procedures and associated results were reviewed to verify bus components were adequately tested and degradation would be identified. The inspectors also reviewed the design compliance with the single failure criterion.

b. Findings

(1) Non-Conforming 480/120 Vac Motor Control Contactors

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," involving the licensee's failure to ensure qualified components were installed in the plant. Specifically, purchase orders did not specify the minimum pickup voltage for NEMA Size 1 through size 4 safety-related motor-control contactors such that the installed contactors were not rated to function at the design basis minimum voltage.

Description: During the review of 480 Vac engineered safety features (ESF) Bus 131x, the inspectors requested information on periodic testing of the minimum pickup voltage for NEMA Size 1 and 2 Contactors used in the control circuits for safety-related motors. The licensee responded that no periodic testing was performed as all of the size 1 through 4 contactors were purchased under a standard specification to have minimum pickup voltages of 70 percent of the rated voltage (84 Vac for 120 Vac rated size 1 and 2 contactors and 322 Vac for 460 Vac rated size 3 and 4 contactors). The 70 percent minimum pickup voltage had been specified in the original specification (F-2755, Proposed Technical Data for 480 Volt Motor Control Centers). During the inspection, the licensee discovered that the contactor supplier (Westinghouse) was not currently testing at the 70 percent pick-up voltage value and initiated Action Request (AR) 1368220. The AR documented that contactors purchased since 1999 had been tested at 85 percent of the rated voltage and that the population of affected contactors could include all of the 120 Vac contactors in the plant. Operability of the installed contactors was supported by the results of 2011 testing in which eight contactors tested had a highest pickup voltage of 78 Vac; 6 volts below the minimum of 84 Vac. This AR also stated that the 70 percent minimum pickup voltage was applicable to the size 3 and 4 contactors; however, no operability concern existed as they were powered from 480 Vac power with minimum voltage drops due to the physical location and short cable runs.

The licensee stated that all of the contactors had been purchased certified to the Westinghouse Environmental and Seismic Qualification reports 11210-CCR-1 and 11210SCR-1 associated with Westinghouse General Order INU-11210; however, the inspectors' review revealed that the purchase orders did not include any minimum pickup voltage requirements. The licensee had been relying only on the certification to the qualification reports to maintain the original technical design and qualification basis for the equipment. As a result, the licensee had not verified the minimum pickup voltage for the contactors actually installed in the safety-related applications in the plant. The inspectors verified that Calculation 19-AQ-24, "Voltage Drop on 480V-120V AC Control Transformer Circuits," assumed the 70 percent minimum pickup voltage and that adequate margin existed between the calculated voltages and the 84 volt minimum

pickup value. For the size 3 and 4 contactors, Appendix F to Calculation 19-AQ-63, "480V Switchgear and MCC Voltages," demonstrated that the minimum expected voltage at the most limiting 480V ESF bus was 88.4 percent or 424 Vac. Since the size 3 and 4 contactors have a published minimum pickup rating of 85 percent of 460 Vac (391 Vac), there was no operability concern; however, Specification F-2755 requirements did not match the actual purchased components.

On June 4, 2012, the licensee received information from Westinghouse that confirmed that none of the contactors had been tested to the 70 percent standard. In addition, the licensee tested an additional 35 NEMA Size 1 Contactors previously removed from the plant. The worst case minimum pickup voltage was 76 volts, which was less than the 70 percent design standard of 84 volts. However, on June 11, 2012, the licensee initiated AR1376793 to document that the 120 Vac contactors purchased before 1999 had only been tested to a 75 percent of rated voltage standard. As a result, all of the installed size 1 or 2 contactors were non-conforming to the design basis. The licensee generated this AR due to the inspectors' concerns with the operability determination process. Specifically the engineering staff had received information that the installed contactors were non-conforming to the design, but had decided that it did not need to go back to shift management for another operability call since the information did not contradict the information provided in the original AR. The inspectors questioned the applicability of Procedure OP-AA-108-115, "Operability Determinations (CM)," in this situation. Specifically, Step 4.1.14, stated "RE-EVALUATE, as necessary, SSC operability following a change in conditions or as additional information about the cause of the degradation/non-conformance becomes known." Shift management had not been informed that all of the installed contactors were non-conforming to design or of the results of the additional contactor testing. The on-duty shift manager agreed and the licensee initiated AR1368220. The shift manager concluded that there was reasonable assurance of operability, but also requested a full operability evaluation. The inspectors reviewed EC 0000389469, "OP EVAL 12-006 – Westinghouse NEMA Size 1 and Size 2 Contactor Pick-Up Voltage Concerns," and had no concerns with the determination that the contactors were operable but non-conforming.

Analysis: The inspectors determined that the failure to specify the minimum pickup voltage in the purchase orders for NEMA Size 1 through 4 safety-related contactors was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Design Control and affected the cornerstone objective of ensuring the capability of the systems to respond to an initiating event to prevent undesirable consequences. Specifically, having installed contactors that may not function under minimum voltage conditions could affect the operability of multiple safety-related structures, systems and components during an event.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I-Initial Screening and Characterization of Findings," Table 4a for the Mitigating System cornerstone. The finding screened as very low safety significance (Green) because the finding was a design deficiency that did not result in a loss of operability or functionality. Specifically, the contactors tested were shown to operate below the 70 percent minimum pickup rating in the design calculation.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the finding was related to original plant design, which had not been reviewed as part of recent licensee activities.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that design control measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, from initial construction until May 18, 2012, the design basis minimum pickup voltage was not specified in purchase order specifications and no testing had been performed to verify the minimum pickup voltage for the installed safety-related motor-control contactors. Because this violation was of very low safety significance and was entered into the licensee's corrective action program as AR01368220 and AR01376793, this violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 5000254/2011009-01; 5000265/2011009-01, Non-Conforming 480/120 Vac Motor Control Contactors).

(2) Failure to Verify the CCW System Capability to Withstand a Thermal Barrier Break:

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to verify the CCW system was capable of withstanding a reactor coolant pump (RCP) thermal barrier break. Specifically, when assuming a single failure of the automatic isolation function, the licensee failed to evaluate the break effect on the CCW system during the 3 minutes postulated to isolate the leak.

Description: Section 9.2.2.4.4, "Shared Function," of the UFSAR stated that the RCP thermal barrier outlet header had a flow indicating switch to close MOV CC685 in the event of high flow, which would be an indication of a tube rupture in a RCP thermal barrier heat exchanger. If the MOV or switch did not operate properly, MOV CC9438 could be used to manually isolate the CCW return line from the RCP thermal barrier. The licensee analyzed the thermal barrier break event in FAI/02-75 "Byron/Braidwood Units 1 and 2 TREMOLO Analysis for MOV 1/2CC9438," to ensure the MOV could be closed. The evaluation determined the maximum differential pressure across CC9438 following the rupture of a single tube in an RCP thermal barrier heat exchanger during full power conditions. The results were then included in the MOV thrust calculations, which verified the valve would function as required under these conditions. For the postulated scenario, it was estimated that it would take 3 minutes to manually close CC9438 if CC685 fails to automatically close. During these 3 minutes, water at reactor coolant system (RCS) pressure (2250 psi) and temperature (550°F) would be entering the CCW system, which would be normally at 40 psi pressure and 105°F. The calculation determined 195 gallons per minute (gpm) of RCS water would be entering the CCW system and flashing to steam causing the pressure and temperature in the line to rapidly increase.

Based on this scenario, the inspectors were concerned the pressure in the CCW system would exceed the design pressure limits for the piping. Although the break would occur in piping rated to withstand the event, the high pressures would also impact downstream piping with a lower rated design pressure. The inspectors were also concerned with the potential effects of water hammer on the piping and piping supports, after the valve was closed and the voided portions of the piping quickly collapsed. The licensee did not have an evaluation addressing these potential effects on the CCW system, creating reasonable doubt the system would be able to withstand the conditions resulting from an RCP thermal barrier break. This issue was documented in AR01377834.

The licensee performed a preliminary evaluation and determined the CCW system would be able to withstand the postulated event. As part of the corrective actions, the licensee was planning to perform a more thorough evaluation of this unanalyzed condition to also demonstrate this event will not cause them to be outside their design basis and the bounds of the ASME Code.

Analysis: The inspectors determined that the failure to verify the CCW system was capable of withstanding a RCP thermal barrier break was a performance deficiency. Specifically, when assuming a single failure of the automatic isolation function, the licensee failed to evaluate the break effect on the CCW system during the 3 minutes postulated to isolate the leak. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Design Control and affected the cornerstone objective of ensuring the capability of the systems to respond to an initiating event to prevent undesirable consequences. Specifically, by failing to evaluate the effect of the thermal barrier rupture on the CC system there was reasonable doubt whether the system was capable of withstanding the forces applied during the course of this event.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of findings,” Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability or functionality.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the finding was related to a calculation performed in 2004, which had not been reviewed as part of recent licensee activities.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculations methods or by the performance of a suitable testing program.

Contrary to the above, as of June 15, 2012, the licensee failed to verify the adequacy of design; specifically they failed to verify the CCW system was able to withstand the loads resulting from a thermal barrier rupture scenario in conjunction with a single failure. Because this violation was of very low safety significance, it was entered into the licensee’s corrective action program as AR01377834, and the licensee planned to

perform an evaluation of the CCW system in response to a thermal barrier break, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000454/2012007-02; 05000455/2012007-02, Failure to Verify the CCW System Capability to Withstand a Thermal Barrier Break).

(3) Non-Conservative Calibration Tolerance Limits for Electrical Relay Settings

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control", for the failure to specify in a design calculation the allowable relay setpoint calibration tolerances. Specifically, the acceptance criteria used in relay setting calibration procedures was not bounded by the relay setting design calculations.

Description: During the review of licensee's protective relay trip setpoint calibration procedures and relay setting calculations to verify whether the applied relay settings were consistent with the designed basis calculations, the inspectors noted that the stated allowable relay setpoint setting tolerances in the relay setting calibration procedure MA-MW-772-701, "Calibration of Over-current Protective Relays," were neither specified nor analyzed in the design basis relay setting calculation 19-AN-3, "Protective Relay Settings For 4.16kV ESF Switchgear." The acceptance criteria in relay setting calibration procedure were not bounded by the relay setting calculations to ensure the relay settings achieved selective tripping under postulated electrical fault or overload conditions. Following discovery, the licensee performed a preliminary evaluation for affected components using the worst case scenario of relay setpoint tolerances stated in the relay setting calibration procedure and concluded that at the limits of the setting tolerances, the relay setpoints would not always meet the acceptance criteria, and selective tripping was no longer ensured. The licensee entered this finding into their corrective action program as AR01377764. Based upon the actual as-left setpoints of the affected components, the licensee determined they would still perform their required safety basis functions.

Analysis: The inspectors determined that the failure to establish adequate relay setpoint tolerances in relay setpoint setting calibration procedures and verify the effects on relay coordination margin in relay setting calculations for relays used on 4.16kV emergency safety feature switchgears was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems Cornerstone attribute of Equipment Performance, and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the allowed relay setpoint calibration limits would not ensure the calibration activities implement the design basis established by the relay setting calculations. At the limits of the allowable relay setting tolerances, selective tripping, during electrical faults was no longer ensured, and, the loss of relay coordination could allow a fault on one piece of equipment to propagate to other safety-related equipment outside the designed isolation boundary.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of Findings," Table 4a for the Mitigating System cornerstone. The finding screened as of very low safety significance (Green) because the finding involved a design deficiency that did not result in a loss of operability or functionality.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the finding was related to original plant design which had not been reviewed as part of recent licensee activities.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that design control measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of June 15, 2012, the acceptance criteria established in relay setting calculations were not translated into relay setpoint calibration procedures. Because this violation was of very low safety significance, it was entered into the licensee's corrective action program as AR01377764, and the licensee planned to revise the calibration procedures, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 5000454/2012007-03; 05000455/2012007-03), Non-Conservative Calibration Tolerance Limits for Electrical Relay Settings).

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed six operating experience issues to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- Industry Issue (AT 00371721), "Water Solid RH During SBLOCA";
- NER QC-12-013, "Diesel Generator Technical Specification Frequency and Voltage Variation not Considered in Loading Calculations";
- IN 1991-18, "Switchyard Problems that Contribute to Loss of Offsite Power";
- IN 1993-92, "Plant Improvements to Mitigate Common Dependencies in Component Cooling Water Systems";
- IN 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design"; and
- IN 2011-14, "Component Cooling Water System Gas Accumulation and Other Performance Issues."

b. Findings

(1) Design Analyses Did Not Adequately Address Potential Flooding of the Auxiliary Building

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to adequately analyze potential design basis internal flooding events in the auxiliary building. Specifically, the licensee's analysis did not account for the possible single failure of an SX isolation valve or its associated power supply, which would have prevented break isolation within 30 minutes.

Description: The inspectors reviewed calculations associated with the consequences of postulated moderate energy piping cracks within the auxiliary building. Specifically, design calculation 3C8-1281-001, “Auxiliary Building Flood Level Calculations,” assumed that the most limiting pipe cracks could be isolated by operator action within 30 minutes. Calculation 3C8-0787-001, “Confirmation of Safe Shutdown Capability after Auxiliary Building Flooding,” evaluated the consequences of the postulated floods, including consideration of an active single failure. In addition, the UFSAR discussed the capability of the plant to withstand flooding resulting from SX pipe cracks in the auxiliary building.

The inspectors reviewed these calculations and identified portions of SX piping that could not be isolated within 30 minutes in the event of a single failure of an SX motor-operated isolation valve (1/2SX001A/B) or its associated power supply. In some cases, other isolation valves would not be accessible to the operators and failure to isolate this piping could result in a leakage path from the safety-related SX cooling tower basin to the lower levels of the auxiliary building.

In response to this issue, the licensee reviewed existing pipe stress analyses and verified that this SX piping in the auxiliary building would meet the “crack exclusion” pipe stress criteria and that these flood sources may be removed from the design basis. The licensee also indicated that there were existing corrective actions to revise these flooding calculations, but these specific vulnerabilities had not been identified. This single failure issue was documented in AR01378533. The licensee also documented the need to revise the UFSAR in AR01377546.

Analysis: The inspectors determined that the failure to adequately analyze potential design basis internal flooding events in the auxiliary building was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of Design Control and affected the cornerstone objective of ensuring the capability of the systems to respond to an initiating event to prevent undesirable consequences. Specifically, the failure to adequately analyze potential design basis internal flooding events in the auxiliary building would affect the capability of safety-related equipment to withstand the postulated event.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase I - Initial Screening and Characterization of findings,” Table 4a for the Mitigating System Cornerstone. The finding screened as of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability or functionality.

Specifically, based on the licensee’s review of existing pipe stress analyses, these specific SX pipe cracks would not need to be postulated in accordance with the design basis.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the finding was related to original plant design which had not been reviewed as part of recent licensee activities.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, design control measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of June 13, 2012, design control measures failed to translate the design basis into design calculations and into plant procedures. Specifically, the licensee did not adequately analyze potential design basis internal flooding events in the auxiliary building based on a postulated worse case single-failure. Because this violation was of very low safety significance, it was entered into the licensee's corrective action program as AR01378533 and AR01377546, and the licensee was revising the flooding analysis, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000454/2012007-04; 05000455/2012007-04, Design Analyses Did Not Adequately Address Potential Flooding of the Auxiliary Building).

.5 Modifications

a. Inspection Scope

The inspectors reviewed four permanent plant modifications related to selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modifications listed below were reviewed as part of this inspection effort:

- DCP 9400205, Emergency Diesel Generator Governor Upgrade;
- EC 382815, Replacement 1A SX Pp Requires IEE;
- EC 379433, Calculate BHP Values for SX Pump Based on Field Test Data Gathered During B2R15; and
- WO 00343479, Replacement of Charging Pump Rotating Elements.

b. Findings

No findings of significance were identified.

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The inspectors performed a detailed review of the procedures listed below associated with the selected scenario, the reactor coolant pump loss of seal cooling event and internal flooding scenarios in the auxiliary building. For the procedures listed, time critical operator actions were reviewed for reasonableness, in plant action were walked down with a licensed operator, and any interfaces with other departments were evaluated. The procedures were compared to UFSAR, design assumptions, and training materials to assure for constancy.

The following operating procedures were reviewed in detail:

- 1BEP ES-1.3; Transfer to Cold Leg Recirculation, Unit 1;
- BOP CC-14; Post LOCA Alignment of the CC System;
- BOP CC-10; Alignment of the U-0 CC Pump and U-0 HX to a Unit;
- 1BOA PRI-7; Essential Service Water Malfunction, Unit 1;
- 0BOA PRI-8; Auxiliary Building Flooding, Unit 0;
- 1BOA PRI-6; Component Cooling Malfunction, Unit 1;
- 1BOA RCP-1; Reactor Coolant Pump Seal Failure, Unit 1; and
- 1BOA RCP-2; Loss of Seal Cooling, Unit 1.

b. Findings

(1) Failure to Provide Means to Detect Leak in Emergency Core Cooling Flow Path:

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the failure to provide a means to detect and isolate a leak in the emergency core cooling system (ECCS) flow path within 30 minutes, which was contrary to the UFSAR. Specifically, the licensee failed to provide a means to detect and isolate a leak within 30 minutes in that neither sump alarms nor radiation monitors were provided for the safety injection (SI) pump rooms.

Description: Section 6.3.2.5, "System Reliability," of the UFSAR stated that the ECCS had been designed and proven by analysis to withstand any single credible active or passive failure during the recirculation phase of an accident. Additionally, Section 6.3.2.5 of the UFSAR stated that the design of the auxiliary building and related equipment was based upon handling of leaks up to a maximum of 50 gpm. The UFSAR also stated that means were provided to detect and isolate such leaks in the ECCS flow path within 30 minutes. As part of the review of the licensee's application for a license, the NRC stated that "the applicant has provided a system of water-level monitors and radiation detectors located in each compartment that contains ECCS components," (Section 6.3.2, "Evaluation of Single Failures," of NUREG-0876, "Safety Evaluation Report related to the operation of Byron Station, Units 1 and 2, Docket Nos. STN 50-454 and STN 50-455, Commonwealth Edison Company," February 1982). In addition, by letter dated April 20, 1982, the licensee informed the NRC that "the auxiliary building is equipped with leak detection sumps which will detect any leakage above normal rates."

The inspectors noted that the RHR and containment spray (CS) pump rooms were equipped with sump alarms. However, such alarms were powered from non-safety-related power sources and would not initially be available after a loss of offsite power (LOOP). The alarms would be repowered from safety-related buses after the transfer from the injection phase to the recirculation phase of an accident was completed. Leaks collected by the floor drain system would eventually be detected, although it would take over an hour for a 50 gpm leak to activate a collection tank high

level alarm. The CV pump rooms, RHR pump rooms, and CS pump rooms were also equipped with air sampling radiation monitors for detection of leaks. However, similar to sump alarms, it was not clear that these monitors would be available initially after a LOOP. Like the sump alarms, they would be repowered from a safety-related power supply after the transfer to the recirculation phase was completed. Additionally, the auxiliary building was equipped with radiation monitors for the ventilation stacks. However, it was not clear that increased radioactivity due to a leak would be readily detected due to dilution and, even if detected, the location would not be readily identifiable. Operating procedures directed operators to check auxiliary building radiation trends after the transfer to the recirculation phase and, if required, after power was restored to equipment. Operators performed rounds of auxiliary building areas on an 8-hour frequency.

Based on the above information, the inspectors concluded that leaks from ECCS components and piping may not be detected and isolated for certain areas such as the SI pumps rooms and portions of the auxiliary building not having sump alarms or local radiation detectors. In response to the inspectors' concerns, the licensee initiated AR01378257. The licensee planned to evaluate options for modifications to either the plant or the plant design bases to address detection of ECCS leakage.

The licensee determined that the dose consequences associated with an ECCS component leak that was not readily detected would be bounded by a single active failure of an electrical train, which had been previously analyzed and found to be acceptable. Additionally, the licensee calculated that a 50 gpm leak would have to be present for over 23 hours before the NPSH for RHR pumps would be affected due to the loss of inventory in the containment sump.

Analysis: The inspectors determined that the failure to provide a means to detect and isolate a leak in the ECCS flow path within 30 minutes was contrary to the UFSAR and was a performance deficiency. Specifically, the licensee failed to provide a means to detect and isolate a leak within 30 minutes in that neither sump alarms nor radiation monitors were provided for the SI pumps rooms. The finding was determined to be more than minor because the finding was associated with the Barrier Integrity cornerstone attribute of Design Control and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the failure to provide a means to detect and isolate a leak in the ECCS flow path within 30 minutes could result in a delayed isolation of such a leak after an accident and result in a greater radionuclide release to the auxiliary building and the environment.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I - Initial Screening and Characterization of findings," Table 4a for the Barrier Integrity Cornerstone. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. Specifically, the finding was related to original plant design, which had not been reviewed as part of recent licensee activities.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The UFSAR stated that means were provided to detect and isolate a leak in the ECCS flow path within 30 minutes.

Contrary to the above, from original construction through June 15, 2012, the licensee failed to provide a means to detect and isolate a leak in the ECCS flow path within 30 minutes. Specifically, the licensee failed to provide a means to detect and isolate a leak within 30 minutes in that neither sump alarms no radiation monitors were provided for the SI pumps rooms. Because this violation was of very low safety significance, it was entered into the licensee's corrective action program as AR 01378257, and the licensee planned to evaluate options for modifications to address detection of ECCS leakage, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000454/2012007-05; 05000455/2012007-05, Failure to Provide Means to Detect Leak in ECCS Flow Path).

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the corrective action program. The inspectors reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action program. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

The inspectors also selected five issues that were identified during previous CDBIs to verify that the concern was adequately evaluated and corrective actions were identified and implemented to resolve the concern, as necessary. The following issues were reviewed:

- AR00897537, AC Power Feed to River Screen House;
- AR00898000, Auxiliary Feedwater Suction Pressure Calculation Enhancement;
- AR00892033, Perform Monitoring Ultrasonic Testing of 0SX10AB-8;
- NCV 05000454/455/2009007-01, Failure to Maintain/Extend the Qualification Basis for Molded-Case Circuit Breakers (MCCBs) Used in Safety-Related Applications Greater than 20 Years; and
- NCV 05000454/455/2009007-02, Inadequate Analysis of Molded-Case Circuit Breaker Test Data.

b. Findings

No findings of significance were identified.

4OA6 Meeting(s)

.1 Exit Meeting Summary

On June 15, 2012, the inspectors presented the inspection results to Mr. T. Tulon, and other members of the licensee staff. The licensee acknowledged the issues presented. Several documents reviewed by the inspectors were considered proprietary information and were either returned to the licensee or handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Tulon, Site Vice President
B. Youman, Plant Manager
D. Basina, Electrical Engineer
E. Blondin, Mechanical/Structural Design Manager
S. Briggs, Operations Manager
A. Corrigan, Mechanical, Structural Design Manager
J. Feimster, Design Engineering Manager
D. Gudger, Regulatory Assurance Manager
N. Halsey, Mechanical Engineer
E. Hernandez, Engineering Director
T. Hulbert, Reg Assurance NRC Coordinator
B. James, Instrument Maintenance Superintendent
C. Keller, Electrical I&C
S. Kerr, WM Director
M. Krawczyk, System Engineer
B. Runde, System Engineer
E. Stender, Mechanical Engineer
D. Sargent, Structural Engineer
H. Welt, Operations SRO
L. Zurawiski, Nuclear Oversight

Nuclear Regulatory Commission

A. M. Stone, Chief, Engineering Branch 2, DRS
B. Bartlett, Senior Resident Inspector
J. Robbins, Resident Inspector
J. Hafeez, Reactor Inspector, DRS

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000454/2012007-01; 05000455/2012007-01	NCV	Non-Conforming 480/120 Vac Motor Control Contactors (Section 1R21.3.b.(1))
05000454/2012007-02; 05000455/2012007-02	NCV	Failure to Verify the CCW System Capability to Withstand a Thermal Barrier Break (Section 1R21.3.b.(2))
05000454/2012007-03; 05000455/2012007-03	NCV	Non-Conservative Calibration Tolerance Limits for Electrical Relay Settings (Section 1R21.3.b.(3))
05000454/2012007-04; 05000455/2012007-04	NCV	Design Analyses Did Not Adequately Address Potential Flooding of the Auxiliary Building (Section 1R21.4.b.(1))
05000454/2012007-05; 05000455/2012007-05	NCV	Failure to Provide Means to Detect Leak in Emergency Core Cooling Flow Path (Section 1R21.6.b.(1))

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.

CALCULATIONS

Number	Description or Title	Revision
002-M-014	RHR MOV Differential Pressure Calculation	4
002-M-063	Byron Unit 1 and Unit 2 CCW System MOV Differential Pressure Calculation	000A
002-M-064	CVCS Differential Pressure Calculation	1B
19-AN-1	Relay settings for Generator, MPT, UAT and SAT	5C
19-AN-3	Protective Relay Settings For 4.16KV ESF Switchgear	16C
19-AN-4	Protective Relay Settings for 4160V Non-Safety-Related Switchgear	15B
19-AN-5	Diesel Generator Protective Relay Settings	3
19-G-1	Cable Ampacity	1
3C8-0787-001	Confirmation of Safe Shutdown Capability After Auxiliary Building Flooding	2
3C8-1281-001	Auxiliary Building Flood Level Calculations	12
95-044	Thermal Endurance Evaluation of CV and SX Pumps	0
ATD-0250	Determination of Hydraulic Characteristics for SX System MOVs	1H
ATO-0021	Heat Load to the Ultimate heat Sink During Station Blackout	1
ATO-0024	SX System Alignment Variations For A Single-Unit LOCA	1
BYR 98-211	RHR/ECCS Pump Flow &Pressure Accuracy Evaluation	0F
BYR 98-212	RHR Pump ASME Surveillance Instrument Accuracy	2
BYR01-089	Motor Operated Valves (MOV) Actuator Motor Terminal Voltage and Thermal Overload Sizing Calculation - CV System	1
BYR03-096	Chemical and Volume Control Strong Pump / Weak Pump Interaction on Recirculation Flow	0
BYR04-016	RHR, SI, CV, and CS Pump NPSH During ECCS Injection Mode	2
BYR05-098	NPSH for RHR Pumps During RC System Mid-Loop Operation	0
BYR06-029	SI/RHR/CS/CV System Hydraulic Analysis in Support of GSI-191	3
BYR06-058	NPSHA for RHR & CS Pumps During Post-LOCA Recirculation	1
BYR07-058	Component Cooling Water NPSH Adequacy	0
BYR07-059	RH System Heat-up During RH Pump Recirculation Without CC Flow Through the RH Heat Exchanger	0
BYR07-089	Review of Aerofin's Evaluation Report for Handhole Cover of SX Strainers	0
BYR07-091	Assessment of Short-Circuit Withstand Capability of a 5KV Power Cable	0
BYR-1 CV8355A	AC Motor Operated Globe Valve Calculation	2
BYR-1 RH611	Thrust/Torque MidaCalc for 1RH611	3
BYR11-031	Evaluate 1SX005 Flange Joint for Reduced Number of Studs	0
BYR-1CC9413A	Midas Data Sheet for 1CC9413A	3
BYR-1CC9438	Midas Data Sheet for 1CC9438	3
BYR-1CV8355A	Midas Data Sheet for 1CV8355A	2
BYR-1SX005	AC Motor Operated Butterfly Valve Calculation	5
BYR95-022	HELB/MELB Evaluation for DCP 9500108	0
BYR96-259	SX System FLO-SERIES Analysis	2E
BYR97-158	SX Water Temperature Rise Due to Pump Heat	0
BYR-97-387	CVCS MOV Differential Pressure Calculation	0
BYR97-467	Component Cooling Heat Exchanger Tube Plugging Evaluation	3

CALCULATIONS

Number	Description or Title	Revision
CN-SEE-04-14	Addenda to Design Report EM-4868, Rev.1 for Non-Stellite Bearing Block	0
COD-013935	Seismic Qualification Review of Westinghouse (NSSS) Class	0
CQD-038779	Modification of Shaft Mechanical Seals for 1,2SX01PA, B	0
CWS-CAE-472C	Flow Switch Setpoints	0
DD-RH-030382	RHR Orifice Plate Sizing for Flow Limiting on RHR Pump Recirculation	1
EC376794	CC System Evaluation	1
EC378827	Technical Evaluation of Potential Gas Voids in CC System	0
FAI/02-75	Byron/Braidwood Units 1 &2 TREMOLO 3 Analysis for MOV 1/2CC9438	0
FIA/02-75	Byron/Braidwood Units 1&2 – TREMOLO 3 Analysis for MOV 1/2CC9438	0
MAD 90-0060	Essential Service Water System Hydraulic Analysis	1
MAD 90-0094	Essential Service Water System Station Blackout Analysis	0K
MAD 91-0080	Service Water Model Calibration	2
MSC-BB-001	MOV Seismic Qualification Re-evaluation at Byron and Braidwood Due to Revised Operating Loads Design Input for Westinghouse Supplied Valves	0
NED-M-MSD-38	Seismic Qualification Reevaluation of the MOVs Listed Below	2
PSA-B-98-08	Byron/Braidwood ECCS Flow Calculations for Safety Analysis	3D
SX2-76	SX Pump Head Check	4A
T-1	Calculations to verify suitability of the EDG neutral grounding resistor	1
19-AQ-24	Voltage Drop on 480-120Vac Control Transformer Circuits	7
19-AQ-63	Division Specific Degraded Voltage Analysis	6
19-AQ-65	Overvoltage Evaluation 4160-480 V Unit Substation Transformer Tap Changes	0
BYR01-087	Motor Operated Valves (MOV) Actuator Motor Terminal Voltage & Thermal Overload Sizing Calculation – Component Cooling (CC) System	001A
BYR01-095	Motor Operated Valves (MOV) Actuator Motor Terminal Voltage and Thermal Overload Sizing Calculation – Essential Service Water (SX) System	000

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
01367734	Seismic Concern on the SX/CC Make-Up Mod Install	05/17/12
01367989	WCAP-17308 Requires Detailed Review	05/13/12
01368220	ESF MCC Contactors Not Tested At Assumed Pickup Voltages	05/18/12
01369698	Assumption in Calc PSA-B-98-08	05/23/12
01372521	NRC CDBI – Regarding Baffle Plate Welds in CC Tanks	05/30/12
01373143	OPEX Evaluation Computation Incorrect	06/01/12
01373652	NRC CDBI –No 50.59 for Operator Time Reduction	06/02/12
01374953	NRC CDBI Identified a Discrepancy with MOV DB and DP Calc	06/06/12
01376426	NRC CDBI – Misleading Opening Description of NER	04/10/12
01376793	CDBI Followup on MCC Contactors (IR 1368220)	06/11/12
01377546	Change Required to Eliminate UFSAR Deficiency	06/13/12
01377764	Protective Relay Setting Tolerances	06/14/12
01377770	NRC CDBI – Questions on CC Surge Tank Baffle Plate Weld	06/14/12
01377834	NRC CDBI - Lack of Formal Analysis	06/13/12
01377869	NRC-CDBI – Preliminary Testing Results For C-H Contactors	06/14/12
01377901	CDBI – Followup on MCC Contactors (IR 1376793)	06/14/12
01377946	NRC CDBI – NRC Concern With Operability Determination Procedure	06/14/12
01378257	CDBI, Question about ECCS Leakage	06/15/12
01380744	Action Tracking Needed For Size 3 And 4 Contactors	06/22/12

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
00253533	Floor Plug Design Requirements	09/16/04
00544821	Water Solid RH During SBLOCA	10/16/06
00629356	CDBI FASA – Westinghouse DC Contractor Pick-Up Voltage Issue	05/11/07
00665855	1B D/G Output BKR Manually Opened Due to Abnormal Indication	08/29/07
00670260	Relay House South Battery Capacity Degradation	09/11/07
00741054	DG Frequency Not Addressed on Calcs	02/26/08
00763421	NER NC-08-019 Yellow NRC CDBI findings – IN 2008-02	04/15/08
00771016	Very Light Surface Corrosion, Spotty Corrosion on 1CC01A	05/02/08
00810972	Review MCCB Breaker	07/22/09
00853952	IST Trend Observation for 1CC01PA Differential Pressure	12/08/08
00892033	Perform Monitoring UT on SX10AB-8	03/12/09
00897537	2009 CDBI Issue – AC Power Feed to the River Screen House	03/25/09
00897630 02	Review Exelon Fleet Practices Concerning Testing & Replacement of MCCBs	07/15/09
00898000	2009 CDBI Issue – AF Suction Pressure Calculation Enhancement	03/26/09
00898543	Westinghouse TB 06-02 Review Issue -2009 CDBI	03/27/09
00907731	OOT Safety-Related HFB Breakers Installed Since 09/2003	04/15/09
00916063	1CV8355A – ASSY – MOV 1A RC PP Seal WTR INJ Inlet ISOL VLV	05/05/09
00920470	U1 CC Surge Tank Level Dropping	05/15/09
00930284	NRC Finding Documented in Inspection Report (MCCBs)	06/11/09
00930301	NRC Finding Documented In Inspection Report (MCCB Test Data)	06/11/09
00968169	Indication of Leak BY From SX Isolations to OCC01A	09/08/09
00971233	1CV8355A – ASSY – MOV 1A RC PP Seal WTR INJ Inlet ISOL VLV	09/27/09
01018676	Review Needed of Braidwood IR 1018119 – CDBI FASA Actions	01/20/10
01046109	Byron Review of BWD CDBI IR 1043396	03/22/10
01056849	Minor Dry Chem Buildup at Stem/Packing Area 1CC9413A	04/13/10
01056849	1CC9413A – MOV U-1 RC PPS SUP UPST ISOL VLV	04/13/10
01076087	2B D/G Voltage Pegged High	06/02/10
01147743	RH Miniflow Closure Accident Analysis Concern	12/02/10
01149502	1CV8355A – MOV 1A RC PP Seal WTR INJ Inlet ISOL VLV	12/06/10
01192924	Piping Configuration on 1SX005 Prevents Stud Removal	03/27/11
01212704	Actuator Indicator Needs To Be Installed	05/06/11
01220072	UAT 141-1 Oil Sample Port Plugged	05/21-11
01225159	Replace Lockout Relay For UAT 141-1	06/06/11
01247004 03	NRC IN 2011-14 Component Cooling Water Sys Gas Accumulation	09/20/11
01250432	NRC Questions Regarding Byron/Braidwood Exceptions to R.G. 1.9	08/11/11
01258339	Byron Station DG 24-Hour Run Needs To Be Revised	09/01/11
01258343	Byron DG Full Load Reject SURV May Need To Be Enhanced	09/01/11
01258653	Byron IR for DG 24 Hour Run Applicable to Braidwood	09/02/11
01258666	Byron IR for DG Full Load Reject Applicable to Braidwood	09/02/11
01342772	Perform DG Full Load Reject at Rated Power Factor in A1R16	03/19/12
01346061	Perform DG Full Load Reject At Rated Power Factor	03/27/12
01348279	NOS ID: Errors Impacting SBO Calculation	03/30/12
01351084	Incorrect Implementation of DG TS License Amendment	04/06/12
01354220	Need to Replace Primary Rosettes on S.O. #01Y017B4-7	04/16/12
01359198	DG Full Load Reject Testing	04/26/12
01359466	Commitment Discrepancy in 2000 TS Change for DG AOT	04/27/12
01362200	0BOSR WF-SA1 Acceptance Criteria Needs Engineering Calc	05/03/12

DRAWINGS

Number	Description or Title	Revision
118E02	Tank Component Cooling Surge Vol. 2,000 Gal	
6E-0-4030VA13	Schematic Diagram Auxiliary Building Charcoal Booster Fan 0A (0VA03CA)	O
6E-0-4030VC01	Schematic Diagram Control Room HVAC System Supply Fan 0A - OVC01CA	T
6E-1-4001A	Station One Line Diagram	O
6E-1-4002A	Single Line Diagram Generator, Main Power & Unit Auxiliary Transformer Unit 1	S
6E-1-4002B	Single Line Diagram System Auxiliary Transformer & 6.9KV Switchgear	K
6E-1-4002C	Single Line Diagram 4.16KV SWGR Bus 141 & 143 Diesel Generator 1A & 480V SWGR	T
6E-1-4007A	Byron – U1- Key Diagram 480V ESF Substation Bus 131X (1AP10E)	M
6E-1-4008C	Tabulation of Trip Settings 480V Auxiliary Building ESF MCC 131X1 (1AP21E) Part 1 and 131X1A (1AP21EA)	AN
6E-1-4016A	Relaying & Metering Diagram Unit Auxiliary Transformer 141-1	I
6E-1-4018A	Relaying & Metering Diagram 4160V ESF SWGR Bus 141	U
6E-1-4018C	Relaying & Metering Diagram 4160V SWGR Bus 143	L
6E-1-4020A	Relaying & Metering Diagram Diesel Generator 1A-1DG01KA Generator Control Part 1	T
6E-1-4020B	Relaying & Metering Diagram DG 1A-1DG01KA Generator Control & Engine Governor Control System Part – 2	V
6E-1-4030AP23	Schematic Diagram System Auxiliary Transformer 142-1 Feed to 4.16KV ESF SWGR. BUS 141-ACB #1412	X
6E-1-4030AP25	Schematic Diagram Reserve Feed From 4.16KV ESF SWGR BUS 241 TO 4.16KV ESF SWGR Bus 141 & ACB 1414	AE
6E-1-4030AP26	Schematic Diagram Bus Tie Breaker – ACB # 1411	K
6E-1-4030AP41	Schematic Diagram Unit Auxiliary Transformer 141-1 Feed to 4.16KV SWGR Bus 143-ACB#1431	J
6E-1-4030AP42	Schematic Diagram Unit Auxiliary Transformer 142-1 Feed to 4.16KV SWGR Bus 143-ACB#1432	G
6E-1-4030AP45	Schematic Diagram 4160V SWGR Bus 143 Undervoltage Relays	I
6E-1-4030AP60	Byron –U1 – Schematic Diagram 480V ESF SWGR. 131X (1AP10E) Manually Operated Breakers	F
6E-1-4030DC01	Schematic Diagram 125V DC Battery Charger III (1DC03E)	N
6E-1-4030DG01	Schematic Diagram DG 1A Feed to 4.16KV ESF SWGR BUS 141 ACB #1413	Z
6E-1-4030DG31	Schematic Diagram DG 1A Starting Sequence Control 1DG01KA, Part-1	AM
6E-1-4030DG32	Schematic Diagram DG 1A Starting Sequence Control 1DG01KA Part-2	AF
6E-1-4030DG35	Schematic Diagram DG 1A Generator Control 1DG01KA	
6E-1-4030DG36	Schematic Diagram DG 1A Generator & Eng. Governor Control 1DG01KA	N
6E-1-4030VD01	Schematic Diagram Diesel Generator Room 1A HVAC System Ventilation Fan 1A – 1VDo1CA	K
6E-1-4030VP01	Schematic Diagram Reactor Containment Fan Cooler 1A – Low Speed 1VP01CA	S
6E-1-4030VP02	Schematic Diagram Reactor Containment Fan Cooler 1A – High Speed 1VP01CA	T
6E-1-4030VP05	Schematic Diagram Reactor Containment Fan Cooler 1C – Low Speed 1VP01CC	U
6E-1-4030VP06	Schematic Diagram Reactor Containment Fan Cooler 1C – High Speed 1VP01CC	T
6E-1-4611M	Unit 1 Internal/External Wiring Diagram 4160V ESF SWGR Bus 141 Cub 12	N
A-206	Auxiliary Building Pump Floor Plan Area 6	AB

DRAWINGS

Number	Description or Title	Revision
A-207	Auxiliary Building Basement Floor Plan Area 2	BC
A-208	Auxiliary Building Basement Floor Plan Area 3	CC
A-209	Auxiliary Building Basement Floor Plan Area 5	AB
A-210	Auxiliary Building Basement Floor Plan Area 6	BO
A-211	Auxiliary Building Basement Floor Plan Area 7	Y
A-212	Auxiliary Building Basement Floor Plan Areas 5 & 7	AG
AC-3	AC One Line Diagram	0
CC-1	CC System	25
M-126, Sh. 1	Diagram of Essential Service Water	BA
M-126, Sh. 2	Diagram of Essential Service Water	AD
M-243, Sh. 3	Auxiliary Building Piping	C
M-243, Sh. 4	Auxiliary Building Piping	E
M-257, Sh. 1	Auxiliary Building Piping	P
M-42, Sh. 1A	Diagram of Essential Service Water	AP
M-42, Sh. 1B	Diagram of Essential Service Water	AP
M-42, Sh. 2A	Diagram of Essential Service Water	BA
M-42, Sh. 2B	Diagram of Essential Service Water	BB
M-42, Sh. 3	Diagram of Essential Service Water	AZ
M-42, Sh. 4	Diagram of Essential Service Water	AN
M-42, Sh. 5A	Diagram of Essential Service Water	AF
M-42, Sh. 5B	Diagram of Essential Service Water	AF
M-42, Sh. 6	Diagram of Essential Service Water	BB
M-42, Sh. 7	Diagram of Essential Service Water	AE
M-62	Diagram of Residual Heat Removal	BD
M-64	Diagram of Chemical & Volume Control & Boron Thermal Regen	AE
M-64, Sh. 1	Diagram of Chemical & Volume Control & Boron Thermal Regen	AE
M-64, Sh. 2	Diagram of Chemical & Volume Control & Boron Thermal Regen	AG
M-64, Sh. 3A	Diagram of Chemical & Volume Control & Boron Thermal Regen	AY
M-64, Sh. 3B	Diagram of Chemical & Volume Control & Boron Thermal Regen	AS
M-64, Sh. 4A	Diagram of Chemical & Volume Control & Boron Thermal Regen	K
M-64, Sh. 4B	Diagram of Chemical & Volume Control & Boron Thermal Regen	L
M-64, Sh. 5	Diagram of Chemical & Volume Control & Boron Thermal Regen	AV
M-64, Sh. 6	Diagram of Chemical & Volume Control & Boron Thermal Regen	AL
M-64, Sh. 7	Diagram of Chemical & Volume Control & Boron Thermal Regen	AM
M-64, Sh. 8	Diagram of Chemical & Volume Control	AD
M-64A	Diagram of Chemical & Volume Control & Boron Thermal Regen	C
M-66, Sh. 1A	Diagram of Component Cooling	AW
M-66, Sh. 1B	Diagram of Component Cooling	AJ
M-66, Sh. 3A	Diagram of Component Cooling	AT
M-66, Sh. 3B	Diagram of Component Cooling	AN
M-66, Sh. 4D	Diagram of Component Cooling	AQ
S-671	Auxiliary Building Essential Service Water Pump Room EL. 330’-0”	W

MISCELLANEOUS

Number	Description or Title	Date or Revision
	Letter: Commonwealth Edison to NRC – Auxiliary Building Flooding	04/12/82
	ComEd and PECO Transmission Planning Criteria	02/27/12
2012-119	Ultrasonic Thickness Calibration Sheet	05/01/12
20897-DB-BYR-CC	MOV Design Basis Document; Component Cooling Water	1
B1A4141.M97 Report SL 101	ELMS DATA: Running Voltage Summary	05/15/12
B1A4141.M97 Report SL 112	MCC 131X1 Bus Short Circuit Information	05/17/12
B1A4141.M97 Report SL 103	ELMS DATA: Short Circuit Summary for Low Voltage Buses	05/15/12
BB-PRA-012	Internal Flood Evaluation Summary Notebook	6
BRW-SE-1997-676	10CFR50.59 – PDP Out of Service for Extended Period of Time	09/24/98
EC 366877	Min. Wall Thickness Evaluation for Lines 0SX10AB-8" & 0SXB3AB-2" Eval	0
EC 379179	Review of Byron Station Bus 141 Molded Case Circuit Breaker (MCCB) Testing Results (B1R16)	000
ECR 393171	Proceduralize TCCP to Install Gasoline-Powered Generators to Provide AC Power to the SX Make-up Pump Battery Chargers in the Event that the RSH Losses all AC Power for an Extended Period	0
ER-AA-321, Attachment 4	IST Pump Evaluation Form	11/28/06, 11/17/08, 03/25/11
ER-AA-321-1005	Condition Monitoring Plan	4
ESC-284	Electrical Engineering Reference For Relays and Current Transformers For Medium Voltage Switchgear	01/26/79
F/L-2708	Unit Auxiliary Transformers Specifications	08/20/75
F/L-2737-01	4160 AND 6900 Volt Switchgear Specification	05/11/76
F-2755 L-2755	Proposal Technical Data for 480 Volt Motor Control Centers, Byron Station – Units 1 and 2, Braidwood Station – Units 1 and 2	09/08/77
FASA 780286	Readiness Review for 2009 NRC Component Design Basis Inspection	0
FASA 01288088	Readiness Review for 2012 NRC Component Design Basis Inspection	02/14/12
IST-BYR-BDOC-V-03	Inservice Testing Bases Document	N/A
Memo 131813	Premature Degradation of the RHR Pump Thrust Bearing	11/03/89
MOV-DB-BYR-CV	MOV Design Basis Document; Chemical & Volume Control	3
MPM-WOGRTSDB50-01	Maintenance Program for Westinghouse Type DB-50 Reactor Trip Circuit Breakers and Associated Switchgear Manual	0
OP Eval 12-006	Westinghouse NEMA Size 1 & 2 Contactor Pick-up Voltage Concerns	000
PE Eval 51481/51482	Item Equivalency Evaluation for the replacement Neutral Grounding Resistor for the UAT 6.9KV winding	07/20/06
PO 00000358	SR Procurement Specification	1

MODIFICATIONS

Number	Description or Title	Date
DCP 9400205	Emergency Diesel Generator Governor Upgrade	08/16/96
EC 382815	Replacement 1A SX Pp Requires IEE	04/27/10
EC 379433	Calculate BHP Values for SX Pump Based on Field Test Data Gathered During B2R15	04/27/10

PROCEDURES

Number	Description or Title	Revision
0B0A ELEC-1	Degraded SWYD Voltage Unit 0	9
0BOA PRI-8	Auxiliary Building Flooding, Unit 0	4
0BOSR WF-SA1	Auxiliary Building Floor Drain Semi-Annual Surveillance	6
0BVSR DC-3b	Unit Common 125V DC Relay House Battery Bank System 2 (South) Performance Discharge Test	2
1/2BHSR DG-1	Diesel Generator 18 month Electrical Inspection	12
1BEP ES-1.3	Transfer to Cold Leg Recirculation, Unit 1	201
1BEP-1	Loss of Reactor or Secondary Coolant, Unit 1	202
1BOA ELEC-2	Table A (Loss of INST Bus 111 Effects)	102
1BOA PRI-1	Excessive Primary Plant Leakage	106
1BOA PRI-6	Component Cooling Malfunction, Unit 1	107
1BOA PRI-7	Essential Service Water Malfunction, Unit 1	105
1BOA RCP-2	Loss of Seal Cooling	105
1BOSR 5.5.38.CC.5-1c	Comprehensive IST Surveillance Requirements for Component Cooling Pump 1CC01PA	0
1BOSR 5.5.8.CV.S-1c	Comprehensive IST Requirements for Centrifugal Charging Pump 1CV01PA	2
1BOSR 5.5.8.RH.5-1c	Group A IST Requirements for Residual Heat Removal Pump 1RH01PA	1
1BOSR 5.5.8.SX.5-1c	Comprehensive IST Requirements for the Essential Service Water (SX) Pump 1SX01PA and Unit 1 SX Pumps Discharge Check Valves	2
1BOSR 5.c.2-1	Charging/Safety Injection System Flow Balance	1
1BOSR 6.6.2-1	Reactor Containment Fan Cooler Monthly Surveillance	27
BAR 1-20-B1	UAT 141-1 Unit Trouble	52
BAR 1-20-D1	UAT 141-1 Oil Flow Low Temp High	1
BAR 1-2-A4	CC Pump Trip	6
BAR 1-2-A5	CC Surge Tank Level High Low	7
BAR 1-2-E4	CC Surge Tank Auto-M/U On	8
BAR 1-7-E3	RCP Therm Barr CC Wtr Temp High	51
BOP CC-10	Alignment of the U-0 CC Pump and U-0 CC HX to a Unit	26
BOP CC-14	Post LOCA Alignment of the CC System	9
BOP RH-5	RH System Startup for Recirculation	24
BOP RH-6	Operation of the RH System in Shutdown Cooling	40
BOP RH-7	Boration of the RH System	11
BOP RH-8	Filling the Refueling Cavity for Refueling	20
BOP RH-9	Pump Down of the Refueling Cavity to the RWST	25
BOP SX-22	Essential Service Water Leak Isolation	5
BOP SX-4	Essential Service Water Strainer Manual Operation	12
BVP 600-10	Auxiliary Power System Breaker Program	0
BVP 800-30	Essential Service Water Fouling Monitoring Program	14
CC-AA-109	Equipment Abandoned via Operational Configuration Change	6
LS-AA-1110	Reportable Event SAF 1.24: Notification of Failure to Comply or Existence of a Defect in a Procured Component	16
LS-AA-120	Issue Identification and Screening Process	14
LS-AA-125	Corrective Action (CAP) Program	16
MA-AA-716-210-1001	Performance Centered Maintenance (PCM) Templates	9
MA-AA-725-102	Preventive Maintenance on Westinghouse Type DHP 4KV ,6.9 and 13.8KV Circuit Breakers	6
MA-AA-725-103	Preventive Maintenance of Westinghouse 4KV, 6.9KV, and 13.8KV Switchgear Cubicles	2

PROCEDURES

Number	Description or Title	Revision
MA-AP-723-450	Molded Case Circuit Breaker ODEN Testing	2
MA-AP-725-101	Preventive Maintenance on Westinghouse 480V Switchgear Cubicles	5
MA-AP-725-562	Preventive Maintenance on Westinghouse Type DS 480V Circuit Breakers	6
MA-BY-723-053	Station Battery Charger 18 Month Surveillance	17
MA-BY-773-503	Unit 1 – 6.9KV UAT and SAT Breakers Relay Routine	2
MA-BY-773-511	Unit 1 – 480 V Unit Substation Feed Breaker Relay Routine	1
MA-MW-772-701	Calibration of Over-current Protective Relays	3
OO-AA-102-102	General Area Checks and Operator Field Rounds	11
OP-AA-108-107	Switchyard Control	3
OP-AA-108-115	Operability Determinations (CM-1)	11
WC-AA-8003	Interface Procedure Between COMED/PECO and EXELON Generation for Design Engineering and Transmission Planning Activities	3

WORK ORDERS

Number	Description or Title	Date
00149305	Internal/External CC Surge Tank PCM Inspection	03/17/08
00343479	Document Acceptability of Replacement CV Pump Rotating Element	11/04/04
00362498	Revise Drawings to Support Installation of Replacement Filnor Neutral Grounding Resistors on UAT's	10/10/06
00847710-01	Manually Backwash SX Strainer Due to Loss of Power	N/A
00974466	125V DC SRH BATT SYS 2 South ""	09/12/07
01058018	1A DG Cubicle Prevent Maint Bus 141 Cub 6	01/16/09
01116821	Replace FF and VR Relays in 1PL07J-9OFF	01/13/09
01119377	DC111 BAT CHGR Breaker Inspection SUB 131X COMPT 4B	01/22/10
01119876-01	Perform MOV Operator Inspection - 1CV8355A	09/18/09
01123163	Perform Preventive Maintenance Inspection and Testing UAT 141-1	10/12/09
01155284	DG 1A Crosstie To Bus241 Sa 242-1 and Crosstie to Bus141	01/20/10
01158660	1RH611Position Indication Test	01/26/10
01170764	1A D/G ESF Actuation and Non-Emer Trip and Gen Trip Surv.	03/18/10
01181874	Unit CC Crosstie IST Valve Strokes – Required During Cold Shutdown	04/02/10
01228567	UAT 141-1 Feed Bus 143 ACB 1431 Relay Routine Cal	10/18/10
01237087	Diagnostic Testing of MOV 1RH611	04/27/12
01252181	UAT 141-1 Feed To Bus 159	05/16/11
01252526	Diesel Generator Electrical Inspection	01/13/11
01252527	1A DG Relay Routine Cal Bus 141 Cub 6	01/14/11
01261677-01	"A" Train Essential Service Water Valve Indication Test	02/10/11
01266811	UAT 141-1 Feed to Bus 157 Relay Routine	04/14/11
01266838	Bus 143 Undervoltage Relay Routine	03/28/11
01266933-01	UT Inspection	04/02/11
01267580	Bus 157 Bus Undervoltage Relay Routine Cal	03/23/11
01273485	Bus 141 Tech Spec Undervoltage Relays Cub 5	03/20/11
01273789-01	Containment Isolation Valve Stroke Test	03/17/11
01275004	1A Diesel Generator Sequencer Test	03/25/11
01275830	1A Diesel Generator Safe S/D Sequence and Single Load Reject	03/26/11
01278245	Unit 1 Generator Relay Routine Calibration	04/20/11
01288621	Battery Charger Operability Test	08/15/11

WORK ORDERS

Number	Description or Title	Date
01290054	125V Battery Charger Operability Test	09/02/11
01305163	DG 1A Crosstie To Bus241 Sat242-1 and Crosstie to Bus141	07/14/11
01322212	1A D/G ESF Actuation and Non-Emer Trip and Gen Trip Surv	11/09/11
01325230	Unit CC Crosstie IST Valve Strokes – Required During Cold Shutdown	07/29/11
01331609	1A DG Breaker Prevent Maint Bus 141 Cub 6	01/14/11
01353253	1A DG Vent Fan Breaker Inspection SUB 131X COMPT 3D	01/12/11
01409331	Perform Calibration of 1FIS-0611	04/23/12
01462990-01	STT for ISX005	11/10/11
01466625	1CC01PA Comprehensive IST Requirements for Component Cooling Water	11/11/11
01484344	ASME Surveillance Requirements for RHR Mini Flow Valve 1RH611	01/24/12
01509398-01	Essential Service Water System Surveillance	02/21/12
06029654	SC-Inspect/Operate/Lubricate Disconnect	11/04/09
20030407	Diagnostic Testing of MOV 1RH611	04/08/03

LIST OF ACRONYMS USED

°F	Fahrenheit Degrees
AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CCW	Component Cooling Water
CS	Containment Spray
CV	Chemical and Volume Control
DC	Direct Current
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ESF	Engineered Safety Feature
GL	Generic Letter
gpm	Gallons per Minute
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
IR	Inspection Report
IST	Inservice Testing
kV	Kilovolt
LOOP	Loss of Off-site Power
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PRA	Probabilistic Risk assessment
psi	Pounds Per Square Inch
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SAT	System Auxiliary Transformer
SDP	Significance Determination Process
SI	Safety Injection
SSC	Systems, Structures, and Components
SX	Emergency Service Water
TOL	Thermal Overload
TS	Technical Specification
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
Vac	Volts Alternating Current
WO	Work Order

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Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-454; 50-455
License Nos. NPF-37; NPF-66

Enclosure: Inspection Report 05000454/2012007; 05000455/2012007(DRS)
w/Attachment: Supplemental Information

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Letter to Mr. Michael Pacilio from Ms. Ann Marie Stone dated July 27, 2012.

SUBJECT: BYRON STATION, UNITS 1 AND 2 COMPONENT DESIGN BASES INSPECTION 05000454/2012007; 05000455/2012007(DRS)

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