UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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October 9, 2013

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, COMPONENT DESIGN BASES INSPECTION REPORT 05000456/2013007; 05000457/2013007

Dear Mr. Pacilio:

On August 30, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection, (CDBI) at your Braidwood Station, Units 1 and 2. The enclosed report documents the results of this inspection, which were discussed on September 16, 2013, with Mr. Bashor, 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.

This inspection also evaluated activities taken to resolve Violation (VIO) 05000456/2011010-01; 05000457/2011010-01, Restoring Compliance with Respect to Single Failures. This cited violation was left open pending completion of the corrective actions. Please refer to Section 4OA5.1 of this report for more information.

Based on the results of this inspection, six NRC-identified findings of very low safety significance were identified. Five of the six 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 these 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 these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission – Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Braidwood 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/

Benny Jose, Acting Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-456 and 50-457
License Nos. NPF-72 and NPF-77

Enclosure: Inspection Report 05000456/2013007; 05000457/2013007(DRS)
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000456; 05000457
License Nos:	NPF-72 and NPF-77
Report No:	05000456/2013007; 05000457/2013007(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	July 29, 2013, through September 16, 2013
Inspectors:	C. Tilton, Senior Reactor Engineer, Lead N. Valos, Operations Engineer M. Munir, Reactor Engineer, Electrical D. Szwarc, Reactor Engineer, Mechanical W. Sherbin, Mechanical Contractor J. Chiloyan, Electrical Contractor
Approved by:	Benny Jose, Acting Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2013007; 05000457/2013007(DRS); 7/29/2013 – 9/16/2013; Braidwood 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. Six Green findings were identified by the inspectors. Five of these findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components within the Cross-Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance for the licensee's failure to ensure the system auxiliary transformer (SAT) 242-1 overcurrent relay provided protection coordination with upstream and downstream protective devices as required by Institute of Electrical and Electronics Engineers (IEEE)-242 and Design Document RPS-TG-3. Specifically, the licensee failed to demonstrate the relays would have provided upstream directional discrimination to allow the offsite power to clear a system fault before disconnecting the plant from the grid. The licensee entered this issue into their corrective action program and after further evaluation concluded the SAT overcurrent relay settings were still acceptable.

  The inspectors determined the performance deficiency was more than minor because if left uncorrected, the performance deficiency had the potential to lead to a more significant safety concern. Specifically, it would have increased the likelihood of events that upset plant stability and affected the availability and reliability of the preferred alternating current (AC) power supply. The inspectors determined the finding was of very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3b(1))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance (Green) and an associated NCV of Technical Specification Surveillance Requirement 3.7.7.1, for the licensee's failure to ensure six component cooling (CC) system manual valves in the flow path servicing safety-related equipment, that were not locked, sealed, or otherwise secured in position, were verified in the correct position every 31 days. The licensee entered this finding into their Correction Action Program, verified the correct position of the six CC system manual valves, and revised surveillance procedures to include the requirement to periodically verify the correct position of these valves.

The performance deficiency was determined to be more than minor because it was similar to IMC 0612, Appendix E, Example 3.c, since more than one valve was in the required position, but not locked, sealed, or otherwise secured in the correct position, and it impacted the Mitigating Systems cornerstone's objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences, (i.e., core damage). Since the finding did not represent an actual loss of safety function, the inspectors screened the finding as having very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3b(2))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to incorporate accident flows in component cooling water (CCW) pump net positive suction head (NPSH) available calculations. Specifically, the licensee failed to calculate the NPSH for the CCW pumps using the run-out flows, which would have resulted in much lower available NPSH. The licensee entered this issue into their Corrective Action Program and recalculated the CCW pump available NPSH and determined that margin remained.

  The inspectors determined that the performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the capability of the CCW system to respond to an initiating event to prevent undesirable consequences. Specifically, by failing to consider the accident loads in the CCW pumps NPSH calculations there was reasonable doubt as to whether the CCW pumps would have been operable during accident conditions. The inspectors determined that the finding was of very low safety significance (Green) because it did not result in the loss of operability or an actual loss of the CCW system. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3b(3))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to consider design control measures commensurate with those applied to the original essential service water (SX) design related to tornado missile protection. Specifically, the licensee processed a physical modification to the SX discharge pipe and failed to protect or evaluate the exposed portion from potential tornado missiles. The licensee entered this issue into their Corrective Action Program and showed by calculation that the modified SX pipe would shear off upon impact from the design basis tornado missile and the safety-related portion would be unharmed and capable of performing its intended function.

  The inspectors determined that the performance deficiency was more than minor because it was associated with the Mitigating Systems cornerstone attribute of design control and affected the cornerstone objective of ensuring the capability of the SX system to respond to an initiating event to prevent undesirable consequences. Specifically, by failing to consider tornado missile protection in the SX design, there was reasonable doubt as to whether the SX pumps would have been operable during accident conditions. Since the finding would degrade two or more trains of a multi-train system or function, the inspectors determined a Detailed Risk-Evaluation was required. Based on the Detailed Risk-Evaluation, the Senior Reactor Analysts determined the delta core damage frequency for the finding was 6.66E-7/yr and was of very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.3b(4))

- Green. The inspectors identified a finding of very low safety significance (Green) and an associated NCV of Technical Specification, Section 5.4.1b for the licensee's failure to establish the necessary actions as required in Emergency Operating Procedures (EOPs) 1(2)BwEP ES-1.3, "Transfer to Cold Leg Recirculation," Revision 201. Specifically, the licensee failed to ensure EOPs 1(2)BwEP ES-1.3 contained the necessary actions for transition to 1(2)BwCA-1.1, "Loss of Emergency Coolant Recirculation" for a small loss of coolant accident (SLOCA) or medium loss of coolant accident (MLOCA) with a concurrent failure of residual heat removal (RHR) heat exchanger (HX) to safety injection (SI) and centrifugal charging pump (CCP) isolation valves. The licensee entered this finding into their Correction Action Program to revise the subject procedures.

  The finding was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of procedure quality and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee failed to ensure the procedure for establishing containment sump recirculation for a SLOCA or MLOCA contained the necessary actions for potential equipment failures. Since the finding resulted in the potential for a loss of the containment sump recirculation function during a SLOCA or MLOCA for certain equipment failures when transferring to containment sump recirculation, the inspectors determined a Detailed Risk-Evaluation was required. Based on the Detailed Risk-Evaluation, the Senior Reactor Analysts determined the delta core damage frequency for the finding was 1.0E-8/yr. and was of very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.6b(2))

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to ensure abnormal operating Procedures (AOPs) 1(2)BwOA S/D-2, "Shutdown LOCA," Revision 104 (105 for Unit 2) contained the necessary actions to immediately terminate Safety Injection (SI) flow if reactor coolant system (RCS) leakage was isolated. Specifically, the licensee failed to update 1(2)BwOA S/D-2, "Shutdown LOCA" to Revision 2 of the Westinghouse Owners Group (WOG) Abnormal Response Guideline (ARG)-2, "Shutdown LOCA," that resulted in a CAUTION not added to terminate SI flow in a timely manner to prevent RCS over-pressurization, if RCS leakage was isolated. The licensee entered this finding into their Correction Action Program to add the CAUTION statement in the procedure.

  The finding was determined to be more than minor because it was associated with the Barrier Integrity cornerstone attribute of procedure quality and affected the cornerstone's objective of providing reasonable assurance that physical design barriers protect the public from radioactive releases caused by accidents or events. Operations in accordance with the procedure may have challenged the RCS barrier during a shutdown LOCA event. Specifically, the licensee failed to update Procedure 1(2)BwOA S/D-2, "Shutdown LOCA" to Revision 2 of the WOG ARG-2, "Shutdown LOCA" guideline that resulted in a CAUTION that was not added to terminate SI flow in a timely manner to prevent RCS over-pressurization, if RCS leakage was isolated. The inspectors conducted an assessment of the risk significance of the issue in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors determined the finding did not require a Phase II assessment and was of very low safety significance (Green). The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. (Section 1R21.6b(1))

B. Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's Corrective Action Program. This violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAIL

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 the design bases have been correctly implemented for the selected risk significant components and the 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 Braidwood Station Standardized Plant Analysis Risk-Model to identify a scenario to use as the basis for component selection. The two accident scenarios selected were loss of condenser heat sink and small loss of coolant accident (SLOCA). Based on these scenarios, 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 21 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 inspectors 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 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 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 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 16 components were reviewed:

- Component Cooling Water Surge Tank (2CC0IT): The inspectors reviewed design analyses associated with the surge tank capability to perform its required functions. The inspectors reviewed the sizing of the tank, refill capability, and overpressure calculations to ensure the tank was capable of performing its intended safety function under different operating scenarios. The inspectors reviewed the current system health report and condition reports associated with the surge tank to ensure that potential issues are being adequately addressed.

- Residual Heat Removal Pump Cubicle Cooler (2VA02SA): The inspectors reviewed the updated final safety analysis report, system description, the current system health report, selected drawings, maintenance and test procedures, and condition reports associated with the cubicle coolers. The inspectors reviewed tube fouling calculations to ensure that the cubicle cooler was meeting its design requirements. The inspectors reviewed eddy current examination results to verify that tube plugging was being appropriately evaluated.

- Component Cooling Water Pump (2CC01PA): The inspectors reviewed design analyses associated with the component cooling water (CCW) pump capacity, net positive suction head (NPSH), and minimum flow to verify the pump's capacity to perform its required functions. The inspectors reviewed surveillance results including quarterly pump inservice testing (IST), flow verification, and performance testing. The inspectors reviewed a sample of operating procedures associated with the pump under normal and accident conditions. The inspectors performed walkdowns of the pump and associated equipment, conducted interviews with the responsible system engineer, and reviewed a sample of corrective action and maintenance documents to verify the material condition of the equipment.

- <u>Component Cooling Water Heat Exchanger (0CC01A)</u>: The inspectors reviewed tube fouling calculations to ensure the heat exchanger was meeting its design requirements. The inspectors reviewed eddy current examination results to verify tube plugging was being appropriately evaluated. The inspectors reviewed heat transfer calculations and test results to ensure the heat exchanger was meeting its design requirements. The inspectors performed walkdowns of the pump and associated equipment, conducted interviews with the responsible system engineer, and reviewed a sample of corrective action and maintenance documents to verify the material condition of the equipment.

- <u>Residual Heat Removal Heat Exchanger 2A Air Operated Flow Control Valves (2RH606 and 607)</u>: These valves are normally open and the safety-related position is open. The inspectors reviewed valve maintenance and limit switch alarm calibration records to ensure that valve alarms in the control room when not fully open, as stated in the Updated Final Safety Analysis Report (UFSAR). Additionally, the valve position verification procedure records required by Technical Specification were reviewed to ensure the valves are verified open periodically.

- <u>Essential Service Water System Pump (2SX01PA)</u>: The inspectors reviewed piping and instrumentation diagrams, pump line up, pump capacities, and in-service testing. Also, the inspectors reviewed calculations related to pump head, flow, and NPSH to ensure the pumps were capable of providing their accident mitigation function. Reviews of the water supply (suction) path, including the susceptibility of plugging or inadvertent bypassing of the main strainer were also conducted. In addition, the inspectors reviewed the licensee responses and actions taken for compliance with Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors reviewed system operating procedures to ensure they were consistent with design requirements. Additionally, a walkdown was performed to assess material condition of the pump and supporting components

- <u>Motor Driven Auxiliary Feedwater Pump (2AF01PA)</u>: The inspectors reviewed piping and instrumentation diagrams, pump line up, pump capacities, and in-service testing. Also, the inspectors reviewed calculations related to pump head, flow, and NPSH to ensure the pumps were capable of providing their accident mitigation function. Reviews of the water supply (suction) path, including the condensate storage tank preferred supply and the emergency service water safety-related supply path were reviewed for seismic design. In addition, the inspectors reviewed the licensee responses and actions taken for compliance with GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment." The inspectors reviewed system operating procedures to ensure they were consistent with design requirements. A walkdown was performed to assess material condition of the pump and supporting components. The inspectors also reviewed the licensee's response to Bulletin 88-04, "Potential Safety-related Pump

Loss”, to ensure pump minimum flow requirements were met, and pump to pump interaction was addressed

- Pressurizer Pneumatic Accumulator (Air Side) (2RY32MA): The inspectors reviewed the sizing basis and leak rate testing of the power operated relief valve (PORV) pneumatic accumulator to ensure it can provide the required amount of air pressure and volume to stroke open the PORV on a loss of normal supply air pressure. The inspectors reviewed recent corrective action documents and operability evaluations to determine whether problems are identified and corrected. The inspectors also reviewed vendor documents which provided the design requirements for the PORV related to the number of strokes required during certain events, such as low temperature overpressure protection LTOP and natural circulation cooldown.

- 125 Vdc Station Battery (2DC01E): The inspectors reviewed calculations and analyses related to battery sizing and capacity, hydrogen generation, and battery room transient temperature. The review was performed to ascertain the adequacy and appropriateness of design assumptions, and to verify the battery was adequately sized to support the design basis required voltage requirements of the 125Vdc safety-related loads under both normal and design basis accident conditions. The inspectors also reviewed a sampling of completed surveillance tests, service tests, performance discharge tests, and modified performance tests. The review of various discharge tests was to verify the battery capacity was adequate to support the design basis duty cycle requirements and to verify that the battery capacity meets Technical Specifications (TS) requirements.

- 125 Vdc Battery Charger (2DC03E): The inspectors reviewed calculations related to sizing and current limit setting to ascertain the adequacy and appropriateness of design assumptions, and to verify the charger was adequately sized to support the design basis duty cycle requirements of the 125Vdc safety-related loads and the associated battery under both normal and design basis accident conditions. The inspectors also reviewed a sampling of completed surveillance tests, service tests, discharge tests. In addition, the test procedures were reviewed to determine whether maintenance and testing activities for the battery charger were in accordance with vendor’s recommendations.

- 125Vdc Bus 211: The inspectors reviewed 125Vdc short circuit calculations and verified the interrupting ratings of the fuses and the molded-case circuit breakers were well above the calculated short circuit currents. The 125Vdc voltage calculations were reviewed to determine if adequate voltage would be available for the breaker open and close coils and spring charging motors. The inspectors reviewed the motor control logic diagrams and the 125Vdc voltage drop calculation to ensure adequate voltage would be available for the control circuit components under all design basis conditions. The inspectors also reviewed the 125Vdc short circuit and coordination calculations to assure coordination between the motor feed breaker open and close control circuit fuses, and 125Vdc supply breakers and to verify the interrupting ratings of the control circuit fuses and the 125Vdc control power feed breaker. The inspectors also reviewed the ground detection alarm setpoint calculation and ground detection procedure.

- 120 Vac Instrument Bus 211: The inspectors reviewed the voltage drop calculations to ensure the safety-related loads fed off the instrument buses have adequate voltage when the inverters are on battery supply during design basis accident condition. The inspectors review included looking at the most limiting load fed off the inverter buses.

- System Auxiliary Transformer (System Auxiliary Transformer 242-1): The inspectors reviewed the design basis descriptions, equipment specifications, system one-line diagrams, voltage tap settings, nameplate data, short circuit and voltage drop calculations, protective relay settings, and loading requirements to evaluate the capability of the transformer to supply the voltage and current requirements to one train of the electrical distribution loads. Transformer protective relay trip setting calculations were reviewed to verify whether adequate primary and backup protections were provided and appropriate coordination margins considered between upstream and downstream protective devices. Relay setpoint calibration test records were reviewed to verify whether appropriate settings were implemented and verify whether relay setpoint drifts were within design assumptions. The team reviewed minor plant modifications on recently installed transformers to verify transformer 242-1 functional design performance requirements were appropriately considered. Completed transformer maintenance records were also reviewed to evaluate whether the results were indicative of any adverse trends. The inspectors interviewed the system engineer and performed visual inspections of the 345/6.9/4,16KV system auxiliary transformer (SAT) 242-1 and its neutral grounding resistor to assess the installation configuration, instrument gauges, nameplates, material condition, and potential vulnerability to hazards.

- 4KVac Engineered Safety Feature Switchgear (Bus 241): The inspectors reviewed vendor specifications, name plate data, one-line diagrams, calculations, design basis descriptions, drawings, calculations of short circuit, voltage drop, protective relay trip setpoints and the ESF Bus 241 loading requirements to evaluate the capability of the 4KV ESF Bus 241 to supply the voltage and current requirements to one train of ESF loads. The inspectors performed independent calculations of short circuit, voltage drop, bus and feeder protective relay trip settings to verify the bus ratings were not exceeded and the bus and feeder relays were appropriately coordinated for normal and accident loading conditions. The inspectors reviewed the results of completed 4160 Vac Bus 241 preventive maintenance records to verify the test results were within their acceptable limits. The loss of voltage and degraded voltage relay settings were also reviewed to verify they satisfied the requirements of Technical Specifications (TS) 3.8.1. Records of system voltage profiles were reviewed to verify they were consistent with the design basis assumptions. The inspectors performed walkdowns of the 4KV ESF Bus 241 to verify equipment alignment, that the installed local and remote circuit breaker control switches and breaker position indicating lights were consistent with design drawings and to assess the observable material conditions and potential vulnerability to hazards.

- Emergency Diesel Generator (2A): The inspectors reviewed the Emergency Diesel Generator (EDG) loading calculations including voltage, frequency, current, and loading sequences during postulated loss of offsite power and loss of coolant accidents to verify the capability of the EDGs to perform their intended safety function. Short circuit calculations were reviewed to ensure the ratings of the generator output breaker were adequate. The inspectors also performed independent calculations of available phase and ground short circuit currents to ensure the maximum system short circuit duty was within equipment rating. Protective relay setpoint calculations and setpoint calibration test results were reviewed to assess the adequacy of protection during testing and emergency operations and to assure excessive setpoint drift had not taken place. The generator grounding scheme was also reviewed to determine the adequacy of ground overcurrent relay coordination, grounding transformer and grounding transformer secondary resistor ratings. The electrical drawings and calculations that describe the generator output breaker control logic, the permissive and inter-locks were reviewed to determine whether the breaker opening and closing control circuits were consistent with design basis documents. The inspectors also reviewed several TS surveillance test results to verify that applicable test acceptance criteria and test frequency requirements for the EDGs were satisfied. The inspectors interviewed system engineers and discussed system performance, and recent issue reports. The inspectors conducted a field walkdown of the electrical relay cabinets; output breaker control switches; breaker position indicating lights; and to assess material conditions.

- <u>480Vac Load Control Center (LCC 231X)</u>: The inspectors reviewed calculations, design basis descriptions, and drawings to verify the duty requirements of the LCC 231X were within the capability of the switchgear and of its power supply unit substation transformer (UST). The inspectors reviewed design assumptions and calculations related to short circuit currents, voltage drop and protective relay settings associated with UST 231 and breaker trip settings associated with Bus 231X to verify they were appropriate. The inspectors also reviewed maintenance procedures and design drawings to assess the adequacy of the ground detection design. The inspectors reviewed a sample of completed maintenance and breaker functional verification testing results to verify that the power supply breaker associated with UST 231 and the cabling to Bus 231X were capable of supplying the power requirement of the 480Vac LCC Bus 231X during normal and postulated accident conditions. The inspectors performed independent short circuit and voltage drop calculations to verify the values stated in the design bases documents were appropriate. The inspectors interviewed system engineers, and conducted a field walkdown of the 4160/480Vac UST 231 and 480Vac LCC Bus 231X to verify equipment alignment and nameplate data were consistent with design drawings and to assess the material condition of the 4160/480Vac UST 231 and that of the 480Vac LCC Bus.

b. <u>Findings</u>

(1) <u>Failure to Adequately Evaluate System Auxiliary Transformer Overcurrent Relay Settings in Design Calculations</u>

<u>Introduction</u>: The inspectors identified a finding of very low safety significance for the licensee’s failure to ensure the system auxiliary transformer (SAT) 242-1 overcurrent relay provided protection coordination with upstream and downstream protective devices as required by IEEE-242 and design document RPS-TG-3. Specifically, the licensee failed to demonstrate the relays would have provided upstream directional discrimination to allow the offsite power to clear a system fault before disconnecting the plant from the grid.

<u>Description</u>: While reviewing the SAT 242-1 relay settings calculations, the team identified design calculation 19-AN-9 had inadequate design inputs. Specifically, the

SAT 242-1 overcurrent relay trip setpoints did not ensure protection coordination with the upstream protective devices for postulated 345KV system faults as neither the values of fault current contributions from the SAT nor the fault clearing times by 345KV protective relays were provided in the SAT overcurrent relay setting calculation 19-AN-9. The inspectors had reasonable doubt the SAT overcurrent relays would have provided upstream directional discrimination to allow the offsite power to clear 345KV system faults before disconnecting the plant from the grid. This would have increased the likelihood of events that upset plant stability and affected the availability and reliability of the preferred alternating current AC power.

The issue was entered into the licensee's Corrective Action Program. After further review and verification performed between the licensee and ComEd, the licensee concluded the SAT overcurrent relay settings were still acceptable on the basis of current 345KV fault studies and 345KV transmission line protection relaying.

Analysis: The inspectors determined the failure ensure the SAT 242-1 overcurrent relay provided protection coordination with upstream and downstream protective devices was contrary to IEEE 242 and design document RPS-TG-3 and a performance deficiency.

The performance deficiency was determined to be more than minor because if left uncorrected, it would have the potential to lead to more significant safety concern. Specifically, by not allowing the offsite power to clear system faults before disconnecting the plant from the grid, it increased the likelihood of events that upset plant stability and affected the availability and reliability of the preferred alternating current AC power.

In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," Table 2 the inspectors determined the finding affected the Initiating Events cornerstone. As a result, the inspectors determined the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions." The inspectors determined that the finding was of very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. The licensee's calculation for determining the SAT 242-1 overcurrent relay trip setpoint was performed during original construction.

Enforcement: This finding does not involve enforcement action because no regulatory requirement violation was identified. Because this finding does not involve a violation and has very low safety significance, it is identified as FIN [05000456/2013007-01; 05000457/2013007-01], Failure to Adequately Evaluate SAT Overcurrent Relay Settings in Design Calculations.

(2) Failure to Ensure Six Component Cooling System Manual Valves Were in the Correct Position as Required by Technical Specification Surveillance Requirement 3.7.7.1

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated Non-Cited Violation (NCV) of TS SR 3.7.7.1 because, the licensee failed to ensure six CC system manual valves in the flow path servicing safety-related equipment, that were not locked, sealed, or otherwise secured in position, were verified in the correct position every 31 days.

Description: On February 19, 1999, Improved Technical Specifications (ITS) were implemented at Braidwood Station via License Amendment No. 98. Following implementation of ITS, SR 3.7.7.1 required the licensee every 31 days to "Verify each CC [system] manual and power operated valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is in the correct position."

During this inspection, the inspectors identified six CC system manual CC system pump and heat exchanger (HX) crosstie valves [1(2)CC9459A, 1(2)CC9467A, and 1(2)CC9467B] that were required to be periodically verified in their correct position every 31 days, since the valves were in the flow path servicing safety-related equipment, and the valves were not locked, sealed, or otherwise secured in the correct position. However, these CC system manual valves were not included in the licensee's surveillance Procedures 1(2)BwOSR 3.7.7.1, "Component Cooling Water System Valve Lineup to Safety-Related Equipment Surveillance," Revision 9 (11 for Unit 2) to meet this requirement. As a result, the licensee initiated Corrective Action Program document IR 01546578, "2013 CDBI - CC Pump and HX Crosstie Valves Question," dated August 14, 2013, to evaluate either adding the valves to surveillance Procedures 1(2)BwOSR 3.7.7.1 or securing the valves by locking them in their required position.

On August 13, 2013, the licensee verified the correct position of the six CC system manual valves 1(2)CC9459A, 1(2)CC9467A, and 1(2)CC9467B. The licensee revised surveillance Procedures 1(2)BwOSR 3.7.7.1 on August 28, 2013 (with Revision 10 for Unit 1, Revision 12 for Unit 2) to include the requirement to periodically verify the correct position of these six CC system valves.

Analysis: The inspectors determined the licensee's failure to ensure six CC system manual valves in the flow path servicing the safety-related equipment, that were not locked, sealed, or otherwise secured in position, were verified in the correct position every 31 days was contrary to TS, SR 3.7.7.1 and was a performance deficiency.

The finding was determined to be more than minor because it was similar to IMC 0612, Appendix E, Example 3.c, because more than one valve was in the required position, but not locked, sealed, or otherwise secured in the correct position, and because it impacted the Mitigating Systems cornerstone's objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, a potentially miss-positioned valve in a safety-related CC system flow path would render portions of the safety-related CC system incapable of performing its required safety function.

In accordance with IMC 0609, "Significance Determination Process," IMC 0609.04 Attachment, "Initial Characterization of Findings," Table 2, the inspectors determined the finding affected the Mitigating Systems cornerstone. As a result, the inspectors determined the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2 for the Mitigating Systems cornerstone. The inspectors answered "no" to all the Mitigating Systems Screening questions in Exhibit 2 and screened the finding as having very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance.

Enforcement: Technical Specification, Section SR 3.7.7.1 states, in part, that each CC manual valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is verified in the correct position in accordance with the Surveillance Frequency Control Program (SFCP). The SFCP requires that such valves be verified in the correct position every 31 days.

Contrary to the above, from February 19, 1999, to August 13, 2013, the licensee failed to ensure each CC manual valve in the flow path servicing safety-related equipment, that is not locked, sealed, or otherwise secured in position, is verified in the correct position every 31 days. Specifically, six manual valves [1(2)CC9459A, 1(2)CC9467A, and 1(2)CC9467B] are in the flow path servicing safety-related equipment and were not locked, sealed, or otherwise secured in position. These valves were not included in Procedures 1(2)BwOSR 3.7.7.1, "Component Cooling Water System Valve Lineup to Safety-related Equipment Surveillance," Revision 17 (15 for Unit 2); and therefore, were not verified to be in the correct position every 31 days.

The violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC's Enforcement Policy because it was of very low safety significance and was entered into the licensee's Corrective Action Program as IR 01546578. NCV 05000456/2013007-02; NCV 05000457/2013007-02, Failure to Ensure Six Component Cooling (CC) System Manual Valves Were in the Correct Position as Required by Technical Specification (TS) Surveillance Requirement (SR) 3.7.7.1]

(3) Failure to Incorporate Accident Flows in Component Cooling Water Pump Net Positive Suction Head Calculations

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to incorporate accident flows in component cooling water (CCW) pump net positive suction head (NPSH) available calculations. Specifically, the licensee failed to calculate the NPSH for the CCW pumps using the runout flows, which would have resulted in much lower available NPSH.

Description: The licensee calculated the available NPSH for the CCW pumps in calculation SD/SA-CVA-57, "CCW – Proof of Design," dated May 8, 1978. That calculation was based on the normal flow of CCW of 3,918 gallons per minute (gpm) rather than accident condition flows that would exceed 5,000 gpm per pump. The licensee did not determine the required NPSH at 3,918 gpm but rather at 4,800 gpm (the design of the pump) in that calculation. The required NPSH at 4,800 gpm was 16 feet while the available NPSH was 73.8 feet, a margin of 57.8 feet.

That analysis did not have a basis for limiting the pump flows to 3,918 or 4,800 gpm rather than post-accident or runout flows. Under accident conditions CCW flow would typically be to the residual heat removal heat exchangers and additional loads. This flow would be slightly over 10,000 gpm as shown in Updated Final Safety Analysis Report (UFSAR) Table 9.2-4 once non-essential loads were isolated. Since these flows typically assume two CCW pumps are in operation and a flow of slightly over 5,000 gpm would be expected for an individual pump. However, it may take some time to isolate the non-essential loads, and each CCW pump may be running at close to runout flow of 7,300 gpm before those loads are isolated. At runout flow the available NPSH margin is significantly smaller.

The licensee entered this issue into their Corrective Action Program as AR 01541318, "CDBI – Review Byron's Calculation BYR07-058 – 0CC01P," dated July 30, 2013. The licensee showed that a calculation performed at the Byron plant in 2007 was bounding for the CCW system configuration at Braidwood by doing a comparison of the piping between the systems. The result showed that a margin of 6.64 feet remained in the CCW pump NPSH.

Analysis: The inspectors determined the failure to incorporate accident flows in CCW pump NPSH calculations was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. Specifically, the licensee failed to calculate the NPSH for the CCW pumps using the runout flows, which would have resulted in much lower available NPSH.

The performance deficiency 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 CCW system to respond to an initiating event to prevent undesirable consequences. Specifically, by failing to consider the accident loads in the CCW pumps NPSH calculations there was reasonable doubt as to whether the CCW pumps would have been operable during accident conditions.

In accordance with IMC 0609, "Significance Determination Process," IMC 0609.04 Attachment, "Initial Characterization of Findings," Table 2 the inspectors determined whether the finding affected the Mitigating Systems cornerstone. As a result, the inspectors determined whether the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions." The inspectors determined the finding was of very low safety significance (Green) because it did not result in the loss of operability or an actual loss of the CCW system.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. The licensee's calculation of record for determining the NPSH for the CCW pumps was performed in 1978.

Enforcement: Title10 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 calculational methods, or by the performance of a suitable testing program.

Contrary to the above, from initial plant operation until August 30, 2013, the licensee failed to verify the adequacy of design of the CCW pumps under accident conditions. Specifically, the licensee failed to calculate the NPSH for the CCW pumps using the runout flows, which would have resulted in much lower available NPSH.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's Corrective Action Program as AR 01541318. The licensee recalculated the CCW pump available NPSH and determined margin remained.

[NCV 05000456/2013007-03; 05000457/2013007-03, Failure to Incorporate Accident Flows in Component Cooling Water Pump Net Positive Suction Head Calculations.]

(4) Failure to Consider Adequate Tornado Missile Protection in Service Water Discharge Pipe

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to consider design control measures commensurate with those applied to the original essential service water (SX) design related to tornado missile protection. Specifically, the licensee processed a physical modification to the SX discharge pipe and failed to protect or evaluate the exposed portion from potential tornado missiles.

Description: On December 27, 1999, the licensee issued 10 CFR 50.59 evaluation BRW-SE-2000-592, to document and evaluate a design change to the essential service water (SX) discharge pipe. This change would extend the SX return pipes from below the Braidwood Cooling Lake surface to above the surface in order to resolve a postulated non-design basis Auxiliary Building flood scenario.

Before the design change, the SX return lines were below the surface of the lake and as documented in UFSAR Section 9.2.5, Ultimate Heat Sink, were Seismic Category I and protected from tornado missiles.

In addition, General Design Criterion 2 requires in part that structures, systems, and components (SSCs) important to safety, such as SX, be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions.

As a result of the change, the licensee reclassified the portion of the pipes above the discharge structure and including the pipe extensions to above the surface from ASME Class 3 to nonsafety-related. In addition, the pipe additions extended above the surface of the lake and were exposed to the environment.

The inspectors were concerned since the licensee did not consider the susceptibility of the modified pipe to tornado missiles. As a response to the inspectors' questioning, the licensee stated they evaluated a tornado missile striking the pipe and the modified portion shearing off exposing the original pipe. With the original pipe exposed, SX would continue to perform its intended functions. When the inspectors requested the calculation that modeled the scenario presented by the licensee, the licensee explained it was a qualitative analysis and no calculation was recorded.

The inspectors were concerned the design change was never proven by test or calculation and therefore had reasonable doubt all trains of SX would be operable during a tornado missile strike.

The licensee entered this issue into their Corrective Action Program as AR 01552073, "CDBI Lack of Support Calculation for SX Discharge Pipe Extension" dated August 29, 2013. The licensee showed by calculation that the modified SX pipe would shear off upon impact from the design basis tornado missile and the safety-related portion would be unharmed and capable of performing its intended function.

Analysis: The inspectors determined whether the failure to protect or evaluate the exposed portion of the SX discharge pipe from potential tornado missiles was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. Specifically, the licensee failed to consider design control measures commensurate with those applied to the original SX design, which included tornado missile protection. This would have had the potential to affect the operability of all trains of SX.

The performance deficiency 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 SX system to respond to an initiating event to prevent undesirable consequences. Specifically, by failing to consider tornado missile protection in the SX design, there was reasonable doubt as to whether the SX pumps would have been operable during accident conditions.

In accordance with IMC 0609, "Significance Determination Process," IMC 0609.04 Attachment, "Initial Characterization of Findings," Table 2 the inspectors determined whether the finding affected the Mitigating Systems cornerstone. As a result, the inspectors determined whether the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions." Question B, External Event Mitigation Systems, referred us to Exhibit 4 since it involved the loss or degradation of equipment or function specifically designed to mitigate a severe weather initiating event (tornado). As a result, Exhibit 4, External Events Screening Questions, a Detailed Risk-Evaluation was required since the performance deficiency would potentially degrade two or more trains of a multi-train system or function.

The Senior Reactor Analysts (SRAs) performed a bounding risk evaluation for the delta core damage frequency (ΔCDF) of a tornado missile strike causing a core damage event at Braidwood due to damage to the Essential Service Water (SX) discharge piping:

- The SRAs assumed a tornado with wind speed exceeding 100 mph would be required to generate a damaging missile.

- The frequency of this tornado for Braidwood is approximately 1.29E-4/yr from the Risk-Assessment Standardization Project (RASP) website.

- The tornado was assumed to cause damage to both SX discharge lines back to the ultimate heat sink (UHS) (a conservative assumption).

- The SRAs further assumed the tornado also caused a severe weather loss of offsite power event with no offsite power recovery.

The Braidwood Standardized Plant Analysis Risk (SPAR) Model Version 8.21 and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 8.0.9.0 software were used by the Senior Reactor Analysts to evaluate the risk significance of this finding. Using the Braidwood SPAR model, the Conditional Core Damage Probability (CCDP) (i.e., if the tornado event occurred and resulted in a loss of offsite power with no offsite power recovery, and damaged both SX discharge lines back to the UHS is approximately 5.16E-3.

Thus, a bounding ΔCDF calculated due to the SX discharge lines vulnerability to missiles is approximately 6.66E-7/yr (i.e., 1.29E-4/yr x 5.16E-4 = 6.66E-7/yr). The dominant sequences were associated with a tornado causing a dual unit loss of essential service water event with a loss of offsite power, followed by a loss of reactor coolant pump (RCP) seal cooling, a failure of RCP seal No. 2, and then either a failure of high pressure injection or a failure of residual heat removal (RHR) with a failure of high pressure recirculation.

Since the total estimated change in core damage frequency was greater than 1.0E-7/yr, IMC 0609 Appendix H, "Containment Integrity Significance Determination Process" was used to determine the potential risk-contribution due to large early release frequency (LERF). Braidwood Station is a 4-loop Westinghouse PWR with a large dry containment. Sequences important to LERF include steam generator tube rupture events and inter-system loss-of-coolant-accident (LOCA) events. These were not the dominant core damage sequences for this finding.

Based on the Detailed Risk Evaluation, the Senior Reactor Analysts determined the finding was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance. The licensee modified the SX discharge pipe in 1999.

Enforcement: Title10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design.

Contrary to the above, as of December 27, 1999, the licensee failed to consider design control measures commensurate with those applied to the original essential service water (SX) design related to tornado missile protection. Specifically, the licensee processed a physical modification to the SX discharge pipe and failed to protect or evaluate the exposed portion from potential tornado missiles.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's Corrective Action Program as AR 01552073. The licensee showed by calculation that the modified SX pipe would shear off upon impact from the design basis tornado missile and the safety-related portion would be unharmed and capable of performing its intended function. [NCV 05000456/2013007-04; 05000457/2013007-04, Failure to Consider Adequate Tornado Missile Protection in SX Discharge Pipe.]

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed five operating experience issues to ensure NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed and are considered inspection samples:

- Information Notice 1989-54: Potential Over-pressurization of the Component Cooling Water System;
- Information Notice 2004-01: Auxiliary Feedwater Pump Recirculation Line Orifice Fouling-Potential Common Cause Failure;

- Regulatory Issue Summary 2005-29: Anticipated Transients That Could Develop Into More Serious Events;

- Information Notice 2011-14: Component Cooling Water System Gas Accumulation and Other Performance Issues; and

- Information Notice 2012-01: Seismic Considerations – Principally Issues Involving Tanks.

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

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

- EC 380048, Revision 2 Steam Generator Tube Rupture Margin to Overfill (SGTR MTO) – PORV Battery Backup Modification.

b. Findings

No findings of significance were identified.

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The inspectors performed a detailed review of the operator actions and the procedures listed below associated with the selected scenarios of (1) a Small Break Loss of Coolant Accident (SLOCA) event, and (2) a Loss of Condenser Heat Sink (LOCHS) event. For the procedures listed, time-critical operator actions were reviewed for reasonableness, simulator scenarios were observed, and in-plant actions were walked down with a non-licensed operator or a licensed operator as appropriate. It was evaluated whether there was sufficient information to perform the procedure, whether the steps could reasonably be performed in the available time, and whether the necessary tools and equipment were available. The procedures were compared to Updated Final Safety Analysis Report (UFSAR) and design assumptions. In addition, the procedures were reviewed to ensure the procedure steps would accomplish the desired result.

The following operator actions were reviewed:

- Operator actions to refill the Auxiliary Feedwater Diesel-Driven Pump day tank;

- Operator actions to stop the Residual Heat Removal pumps on low flow during a SLOCA; and

- Operator actions to depressurize the Reactor Coolant System/Secondary side during a SLOCA.

The following procedures were reviewed:

- 1BwEP-0, “Reactor Trip or Safety Injection,” Revision 206;
- 1BwEP-1, “Loss of Reactor or Secondary Coolant,” Revision 204;
- 1BwEP ES-1.2, “Post LOCA Cooldown and Depressurization,” Revision 203;
- 1BwEP ES-1.3, “Transfer to Cold Leg Recirculation,” Revision 201;
- 1BwCA-1.1, “Loss of Emergency Coolant Recirculation,” Revision 202
- 2BwOA S/D-2, “Shutdown LOCA,” Revision 105;
- 1BwFR-H.1, “Response to Loss of Secondary Heat Sink,” Revision 205; and
- BwOP DO-16, “Filling the Unit 2 Diesel Auxiliary Feedwater Pump Day Tank from the 125,000 or 50,000 Gallon Fuel Oil Storage Tanks,” Revision 17.

b. Findings

(1) Failure to Consider Multiple Failures in the Emergency Operating Procedures 1(2)BwEP ES-1.3, “Transfer to Cold leg Recirculation” as Required by Technical Specification Section 5.4.1b

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of TS, Section 5.4.1b because emergency operating Procedures (EOPs) 1(2)BwEP ES-1.3, “Transfer to Cold Leg Recirculation,” Revision 201 did not establish the necessary actions as required. Specifically, the licensee failed to ensure EOPs 1(2)BwEP ES-1.3 contained the necessary actions for transition to 1(2)BwCA-1.1, “Loss of Emergency Coolant Recirculation” for a small loss of coolant accident (SLOCA) or medium loss of coolant accident (MLOCA) with a concurrent failure of residual heat removal (RHR) heat exchanger (HX) to safety injection (SI) and centrifugal charging pump (CCP) isolation valves.

Description: The inspectors completed a review of EOPs 1(2)BwEP ES-1.3, “Transfer to Cold Leg Recirculation,” Revision 201 to verify the prescribed actions were in agreement with the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs). For the inspectors’ review of 1(2)BwEP ES-1.3, the inspectors chose a case where valve 1(2)CV8804A, “RH HX to CCPs Isolation Valve,” and valve 1(2)SI8804B, “RH HX to SI Pumps Isolation Valve,” both fail to open during a SLOCA or MLOCA where reactor coolant system (RCS) pressure is above the residual heat removal (RHR) pump shutoff head during transfer to cold leg recirculation. During this review, the inspectors noted in Step 8.a, which checks whether either valve 1(2)CV8804A or valve 1(2)SI8804B is open, the Response Not Obtained (RNO) column for this procedure step stated “Continue with Step 9 (Next Page). WHEN one valve is open, THEN do Steps 8b, 8c, and 8d. The result is that if valve 1(2)CV8804A or valve 1(2)SI8804B could not be opened, one would reach the end of 1(2)BwEP-1.3 and then transition back to 1(2)BwEP-1.2, “Post LOCA Cooldown and Depressurization” for a SLOCA or MLOCA.

The inspectors' review of WOG ERG ES-1.3, "Transfer to Cold Leg Recirculation," guideline and its associated background document directed the user in Step 3 (RNO) "IF at least one flow path from the sump to the RCS can NOT be established or maintained, THEN go to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1." The inspectors' evaluation of 1(2)BwEP ES-1.3, Step 8.a (RNO) concluded for the case reviewed Step 8.a (RNO) was not in conformance with WOG ERG ES-1.3 and would have resulted in an inappropriately remaining in 1(2)BwEP ES-1.3 instead of transitioning to 1(2)BwCA-1.1, "Loss of Emergency Coolant Recirculation," since there would be no flow path from the sump to the RCS if valves 1(2)CV8804A and 1(2)SI8804B fail to open with RCS pressure above the shutoff pressure of the RHR pumps during transfer to cold leg recirculation.

The inspectors' review of 1(2)BwCA-1.1, "Loss of Emergency Coolant Recirculation," Revision 202 concluded if a transition to this EOP had occurred from Step 8.a (RNO) following a concurrent failure of valves 1(2)CV8804A and 1(2)SI8804B to open during a SLOCA or MLOCA (with RCS pressure above the shutoff pressure of the RHR pumps during transfer to cold leg recirculation), the following actions in 1(2)BwCA-1.1 would occur that would extend the time until possible ECCS flow interruption (when the RWST is empty) and allow more time to restore a flow path from the sump to the RCS:

- Makeup to the Refueling Water Storage Tank (RWST) would be initiated.

- Actions to minimize Emergency Core Cooling System (ECCS) injection would be taken. The criteria for the securing of ECCS pumps and for SI termination are less restrictive in 1(2)BwCA-1.1 than those in 1(2)BwEP-1.2, "Post LOCA Cooldown and Depressurization." Thus, securing of ECCS pumps would occur at an earlier time and would conserve RWST inventory and extend the time that the ECCS pumps could take suction from the RWST.

Both 1(2)BW EP-1.3 and the WOG ERG ES-1.3 contain a CAUTION that states "ECCS recirculation flow to RCS must be maintained at all times." The basis for this CAUTION as stated in both WOG ERG ES-1.3 background document and the licensee's EOP background document (i.e., BD-EP ES-1.3) is that maintaining core cooling will minimize or prevent fuel damage.

A review of the licensee's EOP background document (BD-EP ES-1.3) for 1(2)BwEP-1.3, Step 8, revealed the licensee had not documented a related step deviation from the WOG ERGs for remaining in 1(2)BwEP ES-1.3 if at least one flow path from the sump to the RCS can NOT be established or maintained (i.e., if RCS pressure is above the RHR pump shutoff head with a failure of valves 1(2)CV8804A and 1(2)SI8804B to open).

Though this issue required two equipment failures (i.e., the failure of valves 1(2)CV8804A and 1(2)SI8804B to open), NUREG-0737, "Clarification of TMI Action Plan Requirements," Section I.C.1, and NUREG-0737, Supplement 1, Section 7, required the EOPs consider the occurrence of multiple failures. Technical Specification Section 5.4.1b required the EOPs to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1. As stated above, the licensee's 1(2)BwEP ES-1.3 was not in conformance with the WOG ERGs for EOP ES-1.3. As a result, the licensee initiated Corrective Action Program documents (IR 01541228, "2013 CDBI – Potential Revision to 1/2BwEP-1.3, dated July 30, 2013, and IR 01544787, "Extent of Condition Review Performed in Support of IR 01541228, dated August 8, 2013") to address the issue.

<u>Analysis</u>: The inspectors determined whether the licensee's failure to ensure EOPs 1(2)BwEP ES-1.3 contained the necessary actions for transfer to 1(2)BwCA-1.1, Loss of Emergency Coolant Recirculation," with a concurrent failure of valves 1(2)CV8804A and 1(2)SI8804B to open was contrary to the requirements of WOG ERG ES-1.3 and was a performance deficiency. Specifically, the licensee failed to ensure EOPs 1(2)BwEP ES-1.3 contained the necessary actions for transfer to 1(2)BwCA-1.1, Loss of Emergency Coolant Recirculation," for a SLOCA or MLOCA with a concurrent failure of valves 1(2)CV8804A and 1(2)SI8804B to open (with RCS pressure above the shutoff pressure of the RHR pumps during transfer to cold leg recirculation).

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone's attribute of procedure quality and affected the cornerstone's objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the licensee failed to ensure the Procedure for establishing containment sump recirculation for a SLOCA or MLOCA contained the necessary actions for potential equipment failures.

In accordance with IMC 0609, "Significance Determination Process," Attachment IMC 0609.04, "Initial Characterization of Findings," Table 2, the inspectors determined the finding affected the Mitigating Systems cornerstone. As a result, the inspectors determined the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 2 for the Mitigating Systems cornerstone. Since the finding resulted in the potential for a loss of the containment sump recirculation function during a SLOCA or MLOCA for certain equipment failures when transferring to containment sump recirculation, the inspectors answered "Yes" to the Mitigating Systems Question A.2 in Exhibit 2 and determined a Detailed Risk-Evaluation was required.

The Braidwood Standardized Plant Analysis Risk (SPAR) Model, Version 8.21 and Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 8.0.9.0 software was used by the Senior Reactor Analysts to evaluate the risk significance of this finding. From the SPAR Model, the following information was obtained:

SPAR Model Designation	Description	Value
IE-SLOCA	Initiating Event Frequency for a SLOCA Event	3.67E-4/yr
IE-MLOCA	Initiating Event Frequency for a MLOCA Event	1.50E-4/yr
HPI-MOV-CF-8804AB	Probability of Common Cause Failure of Valves CV8804A and SI8804B	1.86E-5
HPI-MOV-CC-8804A	Probability of Independent Failure of Valve CV8804A To Open	9.63E-4
HPI-MOV-CC-8804B	Probability of Independent Failure of Valve SI8804B To Open	9.63E-4

The exposure time for the finding was assessed to be one year, since the finding duration is greater than one year [and one year is the maximum exposure time per the NRC's Risk-RASP) Handbook]. Making the conservative assumption that the failure of valves 1(2)CV-8804A and 1(2) SI8804B during a SLOCA or MLOCA initiating event would result in core damage, the delta core damage frequency (ΔCDF) for the finding is obtained as the product of the following factors from the table above:

ΔCDF = [IE-SLOCA + IE-MLOCA] x [HPI-MOV-CF-8804AB + [HPI-MOV-CC-8804A] x [HPI-MOV-CC-8804B]]
= [3.67E-4/yr + 1.50E-4/yr] x [1.86E-5 +(9.63E-4) x (9.63E-4)]
= 1.0E-8/yr

Based on the Detailed Risk-Evaluation, the Senior Reactor Analysts determined the finding was of very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance.

Enforcement: Technical Specification, Section 5.4.1b states, in part, that "Written procedures shall be established, implemented, and maintained covering the following activities: The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in GL 82-33."

NUREG-0737, Supplement 1, Section 7.1.c, requires licensees upgrade their EOPs to be consistent with Technical Guidelines. The Technical Guidelines are specified, in part, by WOG ERG ES-1.3, Transfer to Cold Leg Recirculation," dated April 30, 2005.

The licensee established 1(2)BwEP ES-1.3, "Transfer to Cold Leg Recirculation," as the implementing procedures for WOG ERG ES-1.3 to specify the actions required for transfer to containment sump recirculation.

Contrary to the above, through August 30, 2013, EOPs 1(2)BwEP ES-1.3, "Transfer to Cold Leg Recirculation," did not establish the necessary actions as required. Specifically, the licensee failed to ensure EOPs 1(2)BwEP ES-1.3 contained the necessary actions for transfer to 1(2)BwCA-1.1, Loss of Emergency Coolant Recirculation," for a SLOCA or MLOCA with a concurrent failure of valves 1(2)CV8804A and 1(2)SI8804B to open (with RCS pressure above the shutoff pressure of the RHR pumps during transfer to cold leg recirculation).

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's Corrective Action Program as IR 01541228 and IR 01544787. [NCV 05000456/2013007-05; NCV 05000457/2013007-05, Failure to Consider Multiple Failures in the Emergency Operating Procedures (EOPs) 1(2)BwEP ES-1.3, "Transfer to Cold leg Recirculation" as Required by Technical Specification (TS) Section 5.4.1b]

(2) Procedures for Shutdown Loss of Coolant Accident Not Appropriate If Reactor Coolant System Leakage Is Isolated

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to ensure Abnormal Operating Procedures (AOPs) 1(2)BwOA S/D-2, "Shutdown LOCA," Revision 104 (105 for Unit 2) contained the necessary actions to immediately terminate safety injection (SI) flow if reactor coolant system (RCS) leakage was isolated. Specifically, the licensee failed to update 1(2)BwOA S/D-2, "Shutdown LOCA" to Revision 2 of the Westinghouse Owners Group (WOG) Abnormal Response Guideline (ARG) -2, "Shutdown LOCA," that resulted in a CAUTION that was not added to terminate SI flow in a timely manner to prevent RCS over-pressurization, if RCS leakage was isolated.

Description: The inspectors completed a review of AOPs 1(2)BwOA S/D-2, "Shutdown LOCA," Revision 104 (105 for Unit 2) to verify the prescribed actions were in agreement with WOG ARG-2, "Shutdown LOCA," Revision 2, dated April 30, 2005. During this review, the inspectors noted a CAUTION that was inserted before Step 4 in WOG ARG-2 during Revision 2 (the equivalent step in 1(2)BwOA S/D-2 is also Step 4) stated "If RCS leakage is isolated, Steps 31 through 35 should be immediately performed to terminate SI flow." Steps 31 through 35 (the equivalent steps in 1(2)BwOA S/D-2 are Steps 34 – 38) provide actions to terminate SI flow and establish normal charging flow. The background information for this CAUTION stated the purpose of the CAUTION was to alert the operator of the need to terminate SI in a timely manner to avoid RCS over-pressurization. It further stated the large amount of sub-cooling required to meet the SI reduction criteria in earlier steps may delay reduction of injection flow and lead to RCS over-pressurization. The RCS pressure and pressurizer level criteria in ARG-2 Steps 31 through 33 are less restrictive than the criteria normally used for SI termination to minimize the potential for RCS over-pressurization.

The licensee stated 1(2)BwOA S/D-2 were based on Revision 1 of ARG-2 dated September 30, 1997. This revision did not contain the CAUTION before Step 4 to immediately terminate SI flow if RCS leakage was isolated. The licensee initiated a Corrective Action Program document (IR 01541239, "2013 CDBI – 1/2BwOA S/D-2 Procedure vs ARG Revision, dated July 30, 2013) to revise AOPs 1(2)BwOA S/D-2 to add the CAUTION statement before Step 4 associated with Revision 2 of ARG-2.

Analysis: The inspectors determined the licensee's failure to ensure AOPs 1(2)BwOA S/D-2 contained the necessary actions to immediately terminate SI flow if RCS leakage was isolated was contrary to the requirements of WOG ARG-2 and was a performance deficiency. Specifically, the licensee failed to ensure AOPs 1(2)BwOA S/D-2 contained the necessary actions to immediately terminate SI flow if RCS leakage was isolated to avoid RCS over-pressurization.

The finding was determined to be more than minor because the finding was associated with the Barrier Integrity cornerstone's attribute of procedure quality and affected the cornerstone's objective of providing reasonable assurance that physical design barriers protect the public from radioactive releases caused by accidents or events. Operations in accordance with the procedure may have challenged the RCS barrier during a shutdown LOCA event. Specifically, the licensee failed to update 2BwOA S/D-2, "Shutdown LOCA" to Revision 2 of the WOG ARG-2, "Shutdown LOCA" guideline that resulted in a CAUTION that was not added to terminate SI flow in a timely manner to prevent RCS over-pressurization, if RCS leakage was isolated.

In accordance with IMC 0609, "Significance Determination Process," IMC 0609.04 Attachment, "Initial Characterization of Findings," Table 2, the inspectors determined the finding affected the Barrier Integrity cornerstone. In accordance with IMC 0609.04 Attachment, Table 3, since the finding was associated with a Mode 4 issue, the inspectors conducted an assessment of the risk significance of the issue in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." The inspectors reviewed Appendix G, Attachment 1, "Phase I Operational Checklists for Both PWRs and BWRs. The applicable checklist was Checklist 1, "PWR Hot Shutdown Operation: Time to Core Boiling < 2 hours." The inspectors reviewed the performance deficiency against the safety functions of core heat removal, RCS inventory control, power availability, containment control, and reactivity control as described in Checklist 1. The inspectors determined the licensee reasonably met these safety functions and did not require a Phase II assessment. Therefore, the finding screened as having very low safety significance (Green).

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of current performance.

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

Contrary to the above, since April 30, 2005, the licensee failed to ensure AOPs 1(2)BwOA S/D-2, "Shutdown LOCA," Revision 104 (105 for Unit 2) contained the necessary actions to immediately terminate SI flow if RCS leakage was isolated. Specifically, the licensee failed to update 1(2)2BwOA S/D-2, "Shutdown LOCA" to Revision 2 of the WOG ARG-2, "Shutdown LOCA" Guideline that resulted in a CAUTION that was not added to the procedure that required actions to immediately terminate SI flow in a timely manner to prevent RCS over-pressurization, if RCS leakage was isolated.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's Corrective Action Program as AR 01541239. [NCV 05000456/2013007-06; 05000457/2013007-06, Procedures for Shutdown Loss of Coolant Accident (LOCA) Not Appropriate If RCS Leakage Is Isolated]

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 one issue that was identified during a previous CDBI to verify the concern was adequately evaluated and corrective actions were identified and implemented to resolve the concern, as necessary. The following issue was reviewed:

- NCV 05000456/457/2010007-06, EDGs Fuel Oil Consumption Calculation Failed to Account for Frequency Variations.

b. Findings

No findings of significance were identified.

4OA5 OTHER ACTIVITIES

.1 (Closed) Verification of Margin-to-Overfill Backfit Corrective Actions and Extent of Condition Review: (VIO 05000456/2011010-01; 05000457/2011010-01, Restoring Compliance with Respect to Single Failures

On May 8, 2012, NRC issued IR 05000456/2012002; 05000457/2012002, documenting follow-up actions taken to address a Braidwood Station Steam Generator Tube Rupture (SGTR) Margin-to-Overfill (MTO) item that resulted from the 2009 component design bases inspection at Byron Station. Based on their review, the inspectors concluded the licensee's extent of condition review appeared to be adequate and the issue will remain open pending verification that the proposed PORV power supply modifications are complete.

During this CDBI, the inspectors reviewed actions pertaining to the licensee's addition of an independent PORV power supply. The inspectors reviewed the modification package including the MOD 50.59 Evaluation and also performed a walkdown of the installed modification. The inspectors did not identify any anomaly between the design and the as built condition.

Based on the above review, the inspectors conclude the licensee's PORVs power supply modifications are adequate and this issue is closed.

The documents that were reviewed are included in the Attachment to this report

4OA6 Meeting(s)

.1 Exit Meeting Summary

On September 16, 2013, the inspectors presented the inspection results to Mr. J. Bashor, 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.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) or Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

.1 Failure to Verify the Cooling Water System Capability to Withstand a Thermal Barrier Break

The licensee identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," through an operating experience review for the failure to verify the Component Cooling Water System was capable of withstanding a reactor coolant pump (RCP) thermal barrier break.

Specifically, the licensee failed to evaluate the impact of a failure of valve CC685 to automatically isolate during a postulated RCP thermal barrier rupture event.

The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems 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 (i.e., core damage). The inspectors determined whether finding was of very low safety significance (Green) using because it did not result in a loss of operability or function. The licensee entered this issue into their Corrective Action Program as AR 01452558 and determined design pressures or temperatures would not be exceeded in the event of a thermal barrier break.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

J. Bashor, Site Engineering Director
C. VanDenburgh, Regulatory Assurance Manager
R. Belair, Mechanical Design Engineering Manager
M. Abbas, NRC coordinator
J. Gastouniotis, Design Engineer
A. Totleben, Design Engineer

Nuclear Regulatory Commission

B. Jose, Acting Chief, Engineering Branch 2, DRS
J. Benjamin, Senior Resident Inspector
A. Garmoe, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

05000456/2013007-01; 05000457/2013007-01	FIN	Failure to Adequately Evaluate SAT Overcurrent Relay Settings in Design Calculations. (Section 1R21.3b(1))
05000456/2013007-02; 05000457/2013007-02	NCV	Failure to Ensure Six Component Cooling (CC) System Manual Valves Were in the Correct Position as Required by Technical Specification (TS) Surveillance Requirement (SR) 3.7.7.1. (Section 1R21.3b(2))
05000456/2013007-03; 05000457/2013007-03	NCV	Failure to Incorporate Accident Flows in Component Cooling Water Pump Net Positive Suction Head Calculations. (Section 1R21.3b(3))
05000456/2013007-04; 05000457/2013007-04	NCV	Failure to Consider Adequate Tornado Missile Protection in SX Discharge Pipe. (Section 1R21.3b(4))
05000456/2013007-05; 05000457/2013007-05	NCV	Failure to Consider Multiple Failures in the Emergency Operating Procedures (EOPs) 1(2)BwEP ES 1.3, “Transfer to Cold leg Recirculation” as Required by Technical Specification. (TS) (Section 5.4.1b) (Section 1R21.6b(1))
05000456/2013007-06; 05000457/2013007-06	NCV	Procedures for Shutdown Loss of Coolant Accident (LOCA) Not Appropriate If RCS Leakage Is Isolated. (Section 1R21.6b(2))

Closed

05000456/2011010-01; 05000457/2011010-01	VIO	Restoring Compliance with Respect to Single Failures. (Section 4OA5.1)

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 stated in the body of the inspection report.

CALCULATIONS

Number	Description or Title	Revision/Date
19-AN-29	Second Level Undervoltage Relay Setpoint	2
19-AN-7	Protective Relay Settings for 4.16KV ESF Switchgear	11A
19-AN-9	Relay Settings for Generator, MPT, UAT and SAT	1A
19-AQ-70	Determination of the Minimum Allowable Starting Voltages	1
19-AU-4	480V Unit Substation Breaker and Relay Settings	18M
89-0189	Component Cooling System Over-pressurization Analysis	1
BRW-00-0237-E	Voltage Drop Calculation for 4160V Switchgear Breaker Control Circuits	0
BRW-96-089-M	Verification of Braidwood 125 VDC Battery Room 111, 112, 211 and 212 Ventilation Requirements and Hydrogen Concentration-Evaluation following a Loss of Battery Room Ventilation	002
BYR07-058	Component Cooling Water Pump NPSH Adequacy	0
BYR2000-014/BRW-00-0017-M	Byron/Braidwood Uprate Project – Post LOCA Component Cooling Water System Temperature Analysis	1
BYR97-204/BRW-97-0384-E	125 VDC Battery Sizing Calculation	3
BYR97-205/BRW-97-0383-E	125 VDC Battery Charger Sizing Calculation	2
BYR97-224/BRW-97-0472-E	125 Vdc Voltage Drop Calculation	003
BYR97-225/BRW-970473-E	Circuit Breaker Trip Settings – 125V DC and 250V DC Distribution Centers	1
BYR97-226/BRW-970474-E	125V DC System Short Circuit Calculation	2
BYR97-227/BRW-97-0475-E	125 V DC Fuse Sizing and Coordination	0
BYR97-467/BRW-97-1072-M	Component Cooling Heat Exchanger Tube Plugging Evaluation	3
CC-MP-01	Verification of CC System Overpressure Protection	3
FAI/02-75	Byron/Braidwood Units 1 and 2 – TREMOLO 3 Analysis for MOV 1/2CC9438	0
FSD/SS-M-434	412 Upgrade / Component Cooling System Sizing	11/12/82

CALCULATIONS

Number	**Description or Title**	**Revision/Date**
L-VA-803	Heat Capacity Verification RHR Pump Rooms 1 A/B and 2 A/B	1
SD/SA-CVA-57	CCW – Proof of Design	5/8/78

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	**Description or Title**	**Revision/Date**
01541228	2013 CDBI – Potential Revision to 1/2BwEP ES-1.3	7/30/13
01547000	Typo in Title of BwOP AF-E1	8/15/13
01541035	Typographical Error on Drawing 20E-2-4001A	7/30/13
01541239	2013 CDBI – 1/2BwOA SD-2 Procedure vs ARG Revision	7/30/13
01552073	CDBI Lack of Support Calculation for SX Discharge Pipe Extension	8/29/13
01542430	2A DG Neutral Grounding Resistor Measurement Error	8/1/13
01544787	Extent of Condition Review Performed in Support of IR 01541228	8/8/13
01546578	2013 CDBI – CC Pump and HX Crosstie Valves Question	8/14/13
01551545	Enhancements to Calculation 19-AN-1 and 19-AN-9	8/28/13
01551706	Editorial Error on Drawing 20E-2-4001D	8/28/13

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	**Description or Title**	**Revision/Date**
01101858	Unit 1 Trip Due to Loss of Circulatory Water	8/16/10
01106403	NRC ID'D Unit 2 Vent Overflow Not Corrected in a Timely Manner	8/26/10
01235783	Need Review of Robinson OE 33781 Non-Seismic FC System	7/1/11
01303199	Braidwood Review of RWST Purification	12/15/11
01018119	CDBI FASA Additional Actions Required to Address EDG Frequency Variation	1/19/10
01377834	NRC CDBI – Lack of Formal Analysis	6/14/12
01452558	Byron NRC CDBI – Document Lack of Formal Analysis	12/14/12
01521929	U-2 Surge Tank Level Rising	6/6/13
00985622	DC Bus 211 Ground Evaluation	10/28/09
01003426	Abnormal Ground on DC Bus 211	12/09/09
01100527	Ground on DC Bus 211 – 2DC05E	08/11/10
00990107	Battery 211 Does Not Meet Resistance Acceptance Criteria	11/06/09
01466061	Part 21 ABB Protective Relay Defect	1/23/13
01454406	IER L3-88 Scram Caused by 4 Kv Bus Lockout During Maint	2/12/13
01496503	Part 21 Review ENS 48872 ZPA 3 Phase Relay Type SSC-T	4/2/13
01127318	231X Cubicle 5D RTM not Centered	10/17/10

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Revision/Date
01336181	Plastic Push to Close Pushbutton Disconnected from 2AP10EJ	3/5/12

DRAWINGS

Number	Description or Title	Revision
20E-2-4002E	Single Line Diagram 120V AC ESF Instrument Inverter Bus 211 and 213 125V DC ESF Distribution Center 211	H
20E-2-4030	Schematic Diagram 4.16KV ESF SWGR Bus 241 Feed To 480V Auxiliary Transformer 231X – ACB 2415X	J
20E-2-4030AP01	Schematic Diagram System Auxiliary Transformer 242-1 Tripping Relays	L
20E-2-4030DG01	Schematic Diagram Diesel Generator 2A Feed To 4.16KV ESF SWGR Bus 241 ACB No. 2423	V
20E-2-4030DG31	Schematic Diagram Diesel Generator 2A Starting Sequence Control 2DG01K Part 1	AL
20E-2-4030DG35	Schematic Diagram Diesel Generator 2A Generator Control 2DG01KA	I
20E-2-4030DG40	Schematic Diagram Diesel Generator 2A Shutdown and Alarm System 2DG01KA	O
20E-2-4030AF01	Schematic Diagram Auxiliary Feedwater Pump 2A, 2AF01PA	Y
20E-2-4030SX01	Schematic Diagram Essential Service Water Pump 2A, 2SX01PA	N
20E-2-4030AP23	Schematic Diagram System Auxiliary Transformer 242-1 Feed To 4160V ESF Switchgear Bus 241 ACB No. 2412	Y
20E-2-4030DG01	Schematic Diagram Diesel Generator 2A Feed To 4.16 KV ESF SWGR Bus 241 ACB No. 2413	V
20E-2-4030AP25	Schematic Diagram Reserve Feed From 4.16 KV ESF SWGR Bus 141 to 4.16 KV ESF SWGR Bus 241 ACB No. 2414	Y
20E-2-4030DC05	Schematic Diagram – 125V DC ESF Dist. Center Bus 211	S
20E-2-4020A	Relaying and Metering Diagram Diesel Generator 2A – 2DG01KA Generator Control Part 1	P
20E-2-4019A	Relaying and Metering Diagram 480V ESF Switchgear Bus 231X	I
20E-2-4012A	Key Diagram 120 Vac Instrument Bus 211 (2IP01J) ESF Div. 21 – Channel I	G
20E-2-4010A	Key Diagram 125V DC ESF Distribution Center Bus 211 (2DC05E) Part - 1	L
20E-2-4010B	Key Diagram 125V DC ESF Distribution Center Bus 211 (2DC05E) Part - 2	I
20E-2-4002B	Single Line Diagram System Auxiliary Transformer and 6.9KV Switchgear	F
20E-2-4001A	System One Line Diagram	N
20E-0-4001	Station One Line Diagram	AB

DRAWINGS

Number	Description or Title	Revision
20E-0-4001A	Station One Line Diagram	N
M-139, Sh. 1	Diagram of Component Cooling Unit 2	AW
M-98	Diagram of Diesel Generator Rooms 2A and 2B Ventilation System	V
M-82, Sh. 11	Diagram of Auxiliary Building Equipment Drains Units 1 and 2	AB
M-66, Sh. 3A	Diagram of Component Cooling Units 1 and 2	AU
M-66, Sh. 3B	Diagram of Component Cooling Water	AX
M-66, Sh. 4B	Diagram of Component Cooling	BC
M-49, Sh. 1A	Make-Up Demineralizer Unit 1 and 2	L

MISCELLANEOUS

Number	Description or Title	Date or Revision
OPXR ATI No. 01328230-01	NRC Information Notice 2012-01: Seismic Considerations – Principally Issues Involving Tanks	5/10/12
EC 381986	NRC Regulatory Issue Summary 2005-29: Anticipated Transients That Could Develop Into More Serious Events	0
WOG ARG-2	Shutdown LOCA	1,2
WOG ARG-2 Background	Shutdown LOCA	2
WOG ERG ECA-1.1	Loss of Emergency Coolant Recirculation	2
WOG ERG ECA-1.1 Background	Loss of Emergency Coolant Recirculation	HP-Rev 2
WOG ERG ES-1.3	Transfer to Cold Leg Recirculation	2
WOG ERG ES-1.3 Background	Transfer to Cold Leg Recirculation	2
11-025	Unit 1 and 2 Standing Order - Auxiliary Feedwater Unit Crosstie Valves	2

MODIFICATIONS

Number	Description or Title	Date or Revision
EC 380048	SGTR Margin To Overfill – PORV Battery Backup Mod Main Steam System	002
EC 0000368346	2A Diesel Generator Overcurrent Relay (CO-6) Setpoint Change	000

PROCEDURES

Number	Description or Title	Revision
0BwOA PRI-8	Auxiliary Building Flooding Unit 0	6

PROCEDURES

Number	Description or Title	Revision
0BwOS XKK-M1	U0, U1, and U2 Locked Safety-Related Valve Key Audit	23
1(2)BwCA-1.1	Loss of Emergency Coolant Recirculation	202
1(2)BwEP ES-1.3	Transfer to Cold Leg Recirculation	201
1(2)BwFR-H.1	Response to Loss of Secondary Heat Sink	205
1(2)BwOA S/D-2	Shutdown LOCA	104(105)
1(2)BwOS CC-1	Unit 1(2) Crosstie IST Valve Strokes	4(5)
1BwEP ES-0.2	Natural Circulation Cooldown Unit 1	203
1BwEP ES-0.3	Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS)	202
1BwEP ES-1.1	SI Termination	202
1BwEP ES-1.2	Post LOCA Cooldown and Depressurization	203
1BwEP ES-1.3	Transfer to Cold Leg Recirculation Unit 1	201
1BwEP-0	Reactor Trip or Safety Injection	206
1BwEP-1	Loss of Reactor or Secondary Coolant	204
1BwOA PRI-8	Essential Service Water Malfunction Unit 1	104
1BwOSR 3.7.7.1	Component Cooling System Valve Lineup to Safety-Related Equipment Surveillance	9,10
1BwOSR 3.7.8.1	Unit One Essential Service Water System Surveillance	19
2BwHSR 384-1	125 Volt ESF Battery Charger 211 Setpoints and Alarms Test	0
2BwOSR 3.7.7.1	Component Cooling System Valve Lineup to Safety-Related Equipment Surveillance	11,12
2BwOSR 3.8.1.2-1	Unit Two 2A Diesel Generator Operability Surveillance	35
BD-CA-1.1	Loss of Emergency Coolant Recirculation	202
BD-EP ES-1.3	Transfer to Cold Leg Recirculation	201
BwAR 1-17-A12	Condenser Hotwell Level High Low	13
BwAR 1-2-E4	CC Surge Tank Auto-M/U On	1E3
BwAR 2-2-A5	CC Surge Tank Level High Low	8
BwAR 2-3-D6	AF Pump DO Day Tank Level Low	5E3
BwOP CC-10	Alignment of the “0” CC Pump to a Unit	27
BwOP CC-14	Post LOCA Alignment of the CC System	13
BwOP CC-14	Post LOCA Alignment of the CC System	13
BwOP CC-16	Chromated Drain Tank Transfer to Component Cooling Water Surge Tabk	0
BwOP CC-3	Component Cooling System Filling and Venting	10
BwOP CC-8	Isolation of CC Between Units 1 and 2	22
BwOP CC-8	Isolation of CC Between Units 1 and 2	22
BwOP CC-M1	Operating Mechanical Lineup Unit 1	17
BwOP CC-M2	Operating Mechanical Lineup Unit 2	15
BwOP CC-M3	Operating Mechanical Lineup Unit 0	3E1
BwOP CD-M1	Condensate Operating Mechanical Lineup Unit 1	18
BwOP DC-23-211	125V DC Bus 211/213 Ground Detection	3

PROCEDURES

Number	Description or Title	Revision
BwOP DC-5-211	125V DC ESF Battery 211 Equalization	4
BwOP DG-11	Diesel Generator Startup and Operation	42
BwOP DO-16	Filling the Unit 2 Diesel Auxiliary Feedwater Pump Day Tank from the 125,000 or 50,000 Gallon Fuel Oil Storage Tanks	17
BwOP SX-M1	Operating Mechanical Lineup Unit 1	27
BwOP WE-M1	Operating Mechanical Lineup Unit 0 Auxiliary Building Equip Drain System Operating	10
CC-AA-102	Design Input and Configuration Change Impact Screening	26
EDMG-1	Extensive Damage Mitigation Guideline	5
OP-AA-102-106	Operator Response Time Program	1
OP-BR-102-106	Operator Response Time Program at Braidwood Station	1
WC-AA-80003	Interface Procedure Between COMED/PECO and Exelon Generation (Nuclear/Power) For Design Engineering and Transmission Planning Activities	5

SURVEILLANCES

Number	Description or Title	Date Completed
2BwOSR 3.8.1.10-1	2A DG Full Load Rejection and Simulated SI in Conjunction with UV During Load Testing	10/23/2012
2BwOSR 3.8.1.11-1	2A DG Loss of ESF Bus Voltage with no SI Signal	10/24/2012
2BwOSR 3.8.1.13-1	2A DG Bypass of Auto Trips Not W/Slave Start	12/8/2011
2BwOSR 3.8.1.14-1	Unit 2 2A Diesel Generator 24 Hour Endurance Run	1/12/2013
2BwOSR 3.8.1.19-1	2A Diesel Generator ECCS Sequencer Surveillance	10/24/2013
2BwOSR 3.8.1.2-1	Unit Two 2A Diesel Generator Operability Surveillance	6/5/2013
2BwOSR 3.8.3.35-1	Unit 2 2A Diesel Generator Hot Restart Test	6/5/2013

WORK ORDERS

Number	Description or Title	Date
0113248	Unit Two 125V ESF Battery Charger 211 Capacity Test	11/14/09
00695102	Unit Two 125 Volt Battery Modified Performance Test	10/26/06
01018163	2A RHR Cubicle Cooler Eddy Current Examination Final Report	8/19/09
01064743	Unit 0 CC Heat Exchanger Eddy Current Examination Final Report	2/19/09
01250062 04	SAT 242-1 Neutral Resistor Replacement	4/15/10
01270636	Thermal Performance Test at Start of Outage	5/1/11
01280095	Unit Two 125V ESF Battery Bank 211 Service Test	4/20/11
01305090 01	SAT 242-1 Relay Calibration, Lamping, Trip Checks	5/12/10

WORK ORDERS

Number	Description or Title	Date
01365423	Unit 0 CC Heat Exchanger Eddy Current Examination Final Report	2/10/12
01455066	2DC03E Battery Charger 211 Performance Monitoring	9/30/11
01465298	Unit Two 125V Battery 211 Modified Performance Test	11/3/12
01629821	U2 125V DC Battery Bank and Charger 211 Operability Weekly Surveillance	4/7/13
970006965	2A RHR PP Cubicle Cooler Disassemble and Inspect	7/8/97

LIST OF ACRONYMS USED

AC	Alternating Current
AOP	Abnormal Operating Procedure
AR	Action Request
ASME	American Society of Mechanical Engineers
CC	Component Cooling
CCDF	Conditional Core Damage Frequency
CCW	Component Cooling Water
CDBI	Component Design Basis Inspection
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CW	
DRS	Division of Reactor Safety
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedures
ESF	Engineered Safety Feature
GL	Generic Letter
GPM	Gallons per Minute
HX	Heat Exchanger
IE	
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IST	Inservice Testing
LCC	Load Control Center
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LOCHS	Loss of Condenser Heat Sink
LTOP	Low Temperature Overpressure Protection
MLOCA	Medium Loss of Coolant Accident
MTO	Margin to Overfill
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
PORV	Power Operated Relief Valves
PRA	Probabilistic Risk-Assessment
PWR	Pressurized Water Reactor
RASP	Risk-Assessment Standardized Project
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RIS	Regulatory Issue Summary
RNO	Response Not Obtained
SAPHIRE	System Analysis Program for Hands-On Integrated Release
SAT	System Auxiliary Transformer
SDP	Significance Determination Process
SFCP	Surveillance Frequency Control Program
SI	Safety Injection

SLOCA	Small Loss of Coolant Accident
SRA	Senior Risk Analyst
SX	Essential Service Water
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate Heat Sink
UST	Unit Substation Transformer
VIO	Cited Violation
WOG	Westinghouse Owners' Group

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Sincerely,

/RA/

Benny Jose, Acting Chief
Engineering Branch 2
Division of Reactor Safety

Docket Nos. 50-456 and 50-457
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