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

August 9, 2011

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, NUCLEAR REGULATORY COMMISSION INTEGRATED INSPECTION REPORT 05000456/2011003; 05000457/2011003

Dear Mr. Pacilio:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Braidwood Station, Units 1 and 2. The enclosed inspection report documents the results of this inspection, which were discussed on July 6, 2011, with Mr. M. Kanavos, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, five NRC-identified findings and two self-revealed findings of very low safety significance were identified. Six of the findings were determined to involve violations 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 violations as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. Additionally, two licensee-identified violations, which have been determined to be of very low safety significance, are listed in Section 4OA7 of this report.

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 Braidwood Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the Resident Inspector Office at 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/

Eric R. Duncan, Chief
Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000456/2011003; 05000457/2011003
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report No:	05000456/2011003; 05000457/2011003
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	April 1, 2011, through June 30, 2011
Inspectors:	J. Benjamin, Senior Resident Inspector A. Garmoe, Resident Inspector D. Betancourt-Roldan, Acting Resident Inspector N. Adorno, Reactor Engineer T. Bilik, Reactor Inspector N. Feliz-Adorno, Reactor Inspector T. Go, Health Physics Inspector J. Gilliam, Reactor Engineer V. Meghani, Reactor Engineer R. Ng, Project Engineer J. Robbins, Byron Resident Inspector M. Perry, Resident Inspector Illinois Emergency Management Agency
Approved by:	E. Duncan, Chief Branch 3 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000456/2011003, 05000457/2011003; 04/01/2011 – 06/30/2011; Braidwood Station, Units 1 & 2; Adverse Weather Protection; Fire Protection; Inservice Inspection Activities; Operability Evaluations; Post-Maintenance Testing; Identification and Resolution of Problems.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Five NRC-identified Green findings and two self-revealed Green findings were identified. Six of the findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Assigned cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

Green. The inspectors identified a finding of very low safety significance when licensee personnel failed to adhere to housekeeping and severe weather abnormal operating procedures to ensure specified materials were not stored in the vicinity of the station offsite power transformers. The licensee had implemented these standards to reduce the possibility of material impacting offsite power during severe weather conditions, such as high winds. Corrective actions included the immediate removal of the material from the prohibited areas, reinforcement of the procedural standards to the licensee's staff, and entering the issue into the corrective action program as Issue Reports (IRs) 1221226 and 1221435.

The inspectors determined that the failure to adhere to procedural standards was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Human Performance attribute of the Initiating Events Cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Attachment 4, "Phase 1 – Initial Screening and Characterization of Findings." Using IMC 0609, Attachment 4, and because this finding was associated with the Transient Initiator area of the Initiating Events Cornerstone and did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, the finding was determined to be of very low safety significance (Green). The inspectors determined that this finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area (H.4(c)) because the licensee did not ensure adequate supervisory and management oversight of work activities such that nuclear safety was supported. (Section 1R01.2)

Green. The inspectors identified a finding of very low safety significance and an associated NCV of Braidwood Operating License Condition 2.E when licensee personnel

failed to fireproof a structural steel beam to achieve a required 3-hour fire rating. Specifically, the lack of fireproofing on the structural steel beam degraded a 3-hour rated fire barrier between the auxiliary building laundry room and the Unit 1 lower cable spreading room. The licensee implemented compensatory measures that included hourly fire watches and entered this issue into the corrective action program as IR 1209808.

The inspectors determined that the failure to fireproof the structural steel beam in the auxiliary building laundry room as specified in the Fire Protection Report was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Protection Against External Events attribute of the Mitigating Systems Cornerstone 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 the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Appendix F, "Fire Protection Significance Determination Process," because this finding was associated with or involved the impairment or degradation of a fire protection barrier. This finding was determined to be of very low safety significance (Green) because there were no fire ignition source scenarios that would have caused the structural steel beam to weaken and collapse the ceiling. The inspectors determined there was no cross-cutting aspect associated with this finding because it did not reflect current performance due to the age of the performance deficiency. (Section 1R05.1)

Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50.55a(g)4 when a licensee vendor examiner failed to perform VT-3 visual examinations in accordance with the American Society of Mechanical Engineers (ASME) Code. Specifically the examiner failed to verify the adequacy of illumination following a snubber VT-3 examination. The licensee entered this issue into the corrective action program as IR 1208643 and, following an extent of condition evaluation, re-performed 18 VT-3 visual examinations.

The inspectors determined that the licensee examiner's failure to verify the adequacy of the illumination level following the examination of a snubber was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone 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). Absent NRC identification, the licensee would not have performed the ASME Code-required examinations for a number of components, which could have allowed a rejectable condition to go undetected. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 4, "Phase I - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone and answered "No" to the Mitigating Systems Cornerstone questions. Specifically, the issue did not result in the actual loss of the operability or functionality of a safety system. Therefore, the finding screened as having very low safety significance (Green). This finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area (H.4(b)) because the licensee did not effectively communicate expectations regarding procedural compliance and licensee personnel did not follow procedures. (Section 1R08.1)

Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," when licensee personnel failed to analyze whether the design of the auxiliary feedwater (AF) system ensured that air entrained into the system following a postulated seismic or tornado event did not prevent the system from performing its safety function. Specifically, licensee personnel failed to evaluate the failure of non-seismically qualified condensate storage tank suction piping during an earthquake or tornado that would cause the operating auxiliary feedwater pumps to draw air from the break location, potentially air-binding the pumps. The licensee entered this issue into their corrective action program as 1202772 to identify any required changes to the design of the system and performed an operability evaluation.

The inspectors determined that the failure to analyze whether air entrained into the AF system following a postulated seismic or tornado event would prevent the system from performing its safety function was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Protection Against External Events attribute of the Mitigating Systems Cornerstone 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 the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 4, "Phase I - Initial Screening and Characterization of Findings," Table 4a for the Mitigating Systems Cornerstone and answered "No" to the Mitigating Systems Cornerstone questions. Specifically, the issue did not result in the actual loss of the operability or functionality of a safety system. Therefore, the finding screened as having very low safety significance (Green). The inspectors determined that there was no cross-cutting aspect associated with this finding because it did not reflect current performance due to the age of the performance deficiency.
(Section 1R15.2)

Green. A finding of very low safety significance and an associated NCV of Technical Specification (TS) 5.4.1 was self-revealed on April 21, 2011, when licensee personnel failed to suspend a reactor vessel head lift after it became apparent that there was a large deviation between the crane's actual load cell indication and the expected indication. Immediate corrective actions for this issue included resetting the head on the reactor vessel flange, and resolving the load cell indication issue prior to lifting the head again. The licensee also entered this issue into the corrective action program as IR 1206020.

The inspectors determined that the failure to adhere to a station quality procedure was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Human Performance attribute of the Barrier Integrity Cornerstone and affected the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. The inspectors determined the finding could be evaluated in accordance with IMC 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process," and determined that the finding was Green since it did not require a Phase 2 or Phase 3 analysis. The inspectors determined that this finding had a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area (H.1(b))

because licensee personnel failed to use conservative assumptions in decision making after identifying a large deviation between actual and expected load cell indications during a head lift evolution. (Section 1R19.2)

Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” related to an inadequate quality review of temporarily constructed scaffolds installed throughout the plant. Specifically, the licensee failed to adhere to procedural requirements associated with installed temporary scaffolds prior to reaching 90 days in service. The procedural action required that the temporary scaffold be converted to a permanent scaffold or that a 10 CFR Part 50.59 evaluation be performed for the specific scaffold to ensure that the temporary scaffold did not adversely affect structures, system and components (SSCs) before reaching 90 days in service. Corrective actions included implementing the procedural requirements for the identified scaffolds and entering the issue into the corrective action program as IR 1206426.

The inspectors determined that the failure to adhere to the standards of a quality procedure was a performance deficiency. This issue was determined to be more that minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because the performance deficiency, if left uncorrected, would have the potential to become a more a more significant safety concern. Specifically, by not taking the actions prescribed by procedure, the temporary structures would not have an adequate qualification if left in the plant for greater than 90 days and may not meet all standards of the station’s licensing basis. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 4, “Phase I - Initial Screening and Characterization of findings,” Table 4a for the Mitigating Systems Cornerstone and answered “No” to the Mitigating Systems Cornerstone questions. Specifically, the issue did not result in the actual loss of the operability or functionality of a safety system. Therefore, the finding screened as having very low safety significance (Green). The inspectors determined that this finding had a cross-cutting aspect in the Corrective Action Program component of the Problem Identification and Resolution cross-cutting area (P.1(d)) because the licensee did not take appropriate correct actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. Specifically, the licensee did not take appropriate corrective actions to address a very similar issue identified as NRC inspection finding 05000456/2010004-01; 05000457/2010004-01, “Failure to Follow Procedures for Temporary Scaffolds.” (Section 4OA2.5)

Green. A finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed when incorrect equipment was used during an AF suction piping flush surveillance. Specifically, the use of an incorrect and unqualified drain hose resulted in the hose rupturing and spraying water onto nearby safety-related equipment, rendering the equipment inoperable until equipment tests could be performed. The licensee immediately terminated the flushing operation and entered this issue into the corrective action program as IR 1226235. The licensee also initiated a root cause evaluation to identify additional corrective actions.

The inspectors determined that the use of an improper hose during an AF suction piping flush surveillance was a performance deficiency. This issue was determined to be more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because it was

associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone 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 the finding could be evaluated using the SDP in accordance with IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings.” Using Table 2 of IMC 0609, Attachment 4, the inspectors determined that the finding affected the secondary short-term decay heat removal function of the Mitigating Systems Cornerstone. The inspectors answered “No” to all Mitigating Systems Cornerstone questions in Table 4a, “Characterization Worksheet for Initiating Events, Mitigating Systems, and Barrier Integrity Cornerstone,” and, as a result, the finding was determined to be of very low safety significance (Green). The inspectors determined that this finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area (H.4(a)) because when faced with the choice between two different hoses for a flushing activity, workers proceeded with the evolution in the face of uncertainty. (Section 4OA3.6)

B. Licensee-Identified Violations

Two violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power during the inspection period.

Unit 2 operated at or near full power from the beginning of the inspection period until April 17, when the unit was shut down for a scheduled refueling outage. The unit was started up on May 11, synchronized to the grid on May 12, and reached full power on May 19. On May 28, Unit 2 was reduced in power to approximately 19 percent and the turbine was taken off line and re-balanced to address elevated vibrations. The unit was re-synchronized to the grid that same day and reached full power on May 30. Unit 2 operated at or near full power for the remainder of the inspection period.

During the inspection period, the licensee identified that the reactor power calorimetric calculation for both units was indicating approximately 0.5 percent lower than actual power due to a non-conservative feedwater flow coefficient. As a result, on May 17, 2011, both units were de-rated to approximately 99.5 percent indicated power to ensure actual power remained below 100 percent. On June 16, the licensee implemented a corrected flow coefficient to the power calorimetric calculation, which removed the approximately 0.5 percent disparity between calculated and actual reactor power. Separately, the licensee installed a Leading Edge Flow Meter (LEFM) on Unit 2 as a more accurate method to measure reactor power. As a result, Unit 2 power was being limited to the more conservative value of the calorimetric calculation and the LEFM indication.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness of Offsite and Alternate Alternating Current Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee’s procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors’ review included:

- The coordination between the TSO and the plant during off-normal or emergency events;
- The explanations for the events;
- The estimates of when the offsite power system would be returned to a normal state; and
- The notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- The compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- A re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- The communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

.2 Readiness for Impending Adverse Weather – May 13, 2011, Tornado Warning and May 25, 2011, Tornado Watches

a. Inspection Scope

Since thunderstorms with potential tornadoes and high winds were forecast in the vicinity of the facility during this inspection period, the inspectors reviewed the licensee’s overall preparations and protection for the expected weather conditions. On May 13, 2011, and May 25, 2011, the inspectors reviewed the licensee’s response to a tornado warning and two tornado watches, respectively, for the local area. The inspectors evaluated the licensee’s implementation of site procedures and determined if licensee staff’s actions were adequate. During the inspection, the inspectors focused on plant-specific design features and procedures used to mitigate or respond to specified adverse weather conditions. The inspectors also toured the plant grounds following the tornado warning to identify any loose debris representing a potential missile hazard during a tornado. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. The inspectors also reviewed a sample of CAP items to verify that the licensee identified adverse weather issues at an appropriate

threshold and dispositioned them through the CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment.

This inspection constituted one readiness for impending adverse weather condition sample as defined in IP 71111.01-05.

b. Findings

Failure to Adhere to Procedural Standards Related to the Storage of Outside Material That Could Impact Offsite Power Availability

Introduction: The inspectors identified a finding of very low safety significance (Green) when licensee personnel failed to adhere to housekeeping and severe weather abnormal operating procedures to ensure specified materials were not stored in the vicinity of the station offsite power transformers. Specifically, the licensee failed to identify and remove or secure a section of plastic drain pipe, three boards, and a tarp utilized to cover sea-land containers in a prohibited area adjacent to offsite power transformers. This material had the potential to impact offsite power during severe weather conditions, such as a tornado or high winds.

Description: On May 25, 2011, the inspectors noted that the National Weather Service had issued a tornado watch (Tornado Watch 367) for Will County, Illinois, which was in effect from 7:18 a.m. until 1:00 p.m. Following the notification, the inspectors performed outside plant walkdowns and identified a tarp and three pieces of wood in an area marked as a "Secured Equipment Area" within a line-of-sight of the station's transmission lines. Additionally, the inspectors identified a plastic drain pipe in an area marked as a "Transformer Exclusion Area." These areas were specified in Housekeeping and Material Condition Program Procedure MA-AA-716-026, "Station Housekeeping/Material Control Program." The inspectors immediately notified the licensee of the material storage issue. Later that same day, the National Weather Service issued a second tornado watch for Will County, Illinois (Tornado Watch 373), which was in effect from 6:30 p.m. until 10 p.m.

On May 26, 2011, the inspectors performed an outside plant walkdown, similar to the walkdown that was performed on the previous day. The inspectors identified that although the roll of drain pipe had been removed, the tarp was still in the same location. The inspectors again immediately notified the licensee of the material storage issue. The licensee entered the issue into the CAP as Issue Report (IR) 1221226.

On May 27, 2011, the inspectors determined that the tarp was still in the same location as had been identified on May 25 and May 26, 2011. The inspectors discussed the issue with licensee senior management to ensure that senior management understood that standards were not being met, and to discuss why licensee staff had failed to address the problem after being notified by the inspectors. Following the conversation, the tarp was immediately removed and IR 1221435 was generated to document the issue.

The inspectors identified that the licensee had not adhered to housekeeping standards or the abnormal weather operating procedure through which the licensee should have identified the improper material storage following the two tornado watches on May 25, 2011. More specifically:

- Procedure MA-AA-716-026, “Station Housekeeping/Material Condition Program”, Revision 9, Attachment 1, Storage Practices, Note 5 stated that, “The areas around the Unit 1 and Unit 2 Unit Auxiliary Transformers, Station Auxiliary Transformers, and Main Power Transformers as determined by the vehicle barrier blocks and the walls of the Aux. Bldg., Turbine Bldg., and containments are the Transformers Material Exclusion Areas (NRC Commitment 456-180-98-SCAQ0003a-01).”

  In addition, a secured material zone had been established. Reference 5 to MA-AA-616-026 listed the areas the material was stored in as a secured material zone. This procedure stated that no material may be brought into or stored inside of or above the exclusion zone areas unless prior permission was received from the Shift Manager. Additionally, the procedure stated that material should be secured in the secured area zone to prevent damage in the excluded area in the event of adverse weather conditions.

- Step 3.b of quality procedure 0Bw0A ENV-1, “Adverse Weather Conditions Unit 0,” Revision 110, stated, “Secure or remove any loose material and equipment from around the plant exterior that could impact offsite power availability.” Preceding Step 3 was a note that stated, “The following types of materials may present a hazard during high winds: Plywood, plastic and cloth tarpaulins…” Therefore, through implementation of 0BwOA ENV-1, licensee personnel should have identified the prohibited material following the May 25, 2011 tornado watch notifications.

The inspectors identified that these standards were based upon corrective actions from a 1998 Braidwood Unit 1 loss-of-offsite-power (LOOP) event. The cause of the LOOP event was related to high winds blowing material (i.e., a braided cable) into an energized station auxiliary transformer.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to adequately control material in accordance with station procedures that could have affected offsite power availability during severe weather conditions was a performance deficiency.

The inspectors determined that the issue was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, “Issue Screening,” since the finding was associated with the Human Performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, controls and actions prescribed by station procedures to limit the likelihood of losing preferred offsite power during high wind conditions were not followed.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings.” Using IMC 0609, Attachment 4, and because this finding was associated with the Transient Initiator area of the Initiating Events Cornerstone and did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, the finding was determined to be of very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area since licensee personnel failed to provide oversight of work activities to ensure that nuclear safety was supported (H.4(c)).

Enforcement: No violation of regulatory requirements was identified. Because this finding did not involved a violation and had very low safety significance, it was identified as a Finding **(FIN 05000456/2011003-01; 05000457/2011003-01, Failure to Follow Procedural Standards Related to the Storage of Outside Material That Could Impact Offsite Power Availability)**

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Unit 2 125 Volts Direct Current (VDC), Division 21 Electrical System;
- Unit 1 and Unit 2 “B” Train Auxiliary Feedwater (AF) Systems;
- Unit 2 “A” Train Residual Heat Removal (RHR) System.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

On April 25, 2011, the inspectors performed a complete system alignment inspection of the spent fuel pool cooling system to verify the functional capability of the system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment during the refueling outage. The inspectors walked down the system to review mechanical and electrical equipment line-ups, electrical power availability, system pressure and temperature indications, component labeling, component lubrication, component and equipment cooling, hangers and supports, and the operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Unit 2 "B" Train Auxiliary Feedwater Pump Room, Fire Zone 11.4A-2;
- Unit 1 "A" Train Diesel Generator Day Tank Room, Fire Zone 9.2-1and 9.3-1;
- Auxiliary Building 401', Fire Zone 11.5;
- Auxiliary Building Laundry Room, Fire Zone 11.6C-0;
- Unit 1 "A" Train Emergency Diesel Generator Room, Fire Zone 9.2-0; and
- Fuel Handling Building, Fire Zone 12.1-1;

The inspectors reviewed these areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events (IPEEE) with later additional insights, their potential to impact equipment which could initiate or

mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that issues identified during the inspection were entered into the licensee's CAP.

These activities constituted six quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

Structural Steel Beam Missing Fire Proofing Materials

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated Non-Cited Violation (NCV) of Braidwood Operating License Condition 2.E when licensee personnel failed to fireproof a structural steel beam to achieve a required 3-hour fire rating. Specifically, the lack of fireproofing on the structural beam degraded a 3-hour rated fire barrier between the auxiliary building laundry room and the Unit 1 lower cable spreading room.

Description: On April 29, 2011, the inspectors performed a fire protection walkdown of the Auxiliary Building Laundry Room (Fire Zone 11.6C-0). The inspectors identified that a structural steel beam supporting the laundry room ceiling and Unit 1 lower cable spreading room floor was not fully fireproofed. The beam was inside a pipe chase that contained miscellaneous drain piping and ventilation ductwork. The inspectors determined that the structural steel beam carried a 3-hour fire rating in accordance with Section 2.3.11.56, "Auxiliary Building Laundry Room – Elevation 426' 0" (Fire Zone 11.6C-0)," of the Braidwood Fire Protection Report since there were safe shutdown cables between the laundry room and the Unit 1 lower cable spreading room. Specifically, Section 2.3.11.56 of the Fire Protection Report stated, in part, that "The [ceiling] slab is supported by steel beams and columns which are protected with a fire-resistant coating and carry a three-hour fire rating." Based on this arrangement, heat generated by a fire in the laundry room could weaken the structural steel beam to the point that the supported ceiling might collapse and cause damage to the lower cable spreading room above. Based on visual inspection and later confirmed by the licensee, this condition likely existed since initial plant construction. This issue was entered into the licensee's CAP as IR 1209808. The licensee implemented compensatory measures that included hourly fire watches.

Analysis: The inspectors determined that the failure to fireproof the structural steel beam in the auxiliary building laundry room as specified in the Braidwood Fire Protection Report was a performance deficiency.

The inspectors determined that the issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," since the issue was associated with the Protection Against External Events attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

The inspectors determined that the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, "Significance Determination Process," Appendix F, "Fire Protection Significance Determination Process," because it was associated with or involved impairment or degradation of a fire protection barrier. The inspectors performed a Phase 2 evaluation of the Fire Protection SDP because the finding was associated with a high degradation of fire confinement since fireproofing material was not present to protect the steel beam from heat generated by a fire. Based on a walkdown by the inspectors and the licensee's evaluation, it was determined that there were no fire ignition source scenarios that would have caused the structural steel beams to weaken to the point that the ceiling might collapse. Therefore, no potentially challenging fire scenarios existed.

From Table 2.2.1 of Step 2.2 of IMC 0609, Appendix F, the only Fire Damage State (FDS) scenarios that apply for degradations in "Fire Confinement" are FDS3 scenarios (i.e., scenarios in which fire damage extends to a fire area adjacent to the fire area of fire origin, in general, due to postulated fire spread through a degraded inter-area fire barrier element [e.g., a steel beam without fire proof material present]). However, per Question 2 in IMC 0609, Appendix F, Task 2.2.2, "Screening Assessment for FDS3 Scenarios," FDS3 scenarios are screened out if there is a non-degraded automatic gaseous room-flooding fire suppression system either in the exposed (i.e., auxiliary building laundry room) or the exposing (i.e., Unit 1 lower cable spreading room) fire area. Since the Unit 1 lower cable spreading room had an automatic total flooding carbon dioxide system, FDS3 scenarios screened out. Therefore, this finding was determined to be of very low safety significance (Green).

The inspectors determined there was no cross-cutting aspect associated with this finding because it did not reflect current performance due to the age of the performance deficiency.

Enforcement: Braidwood Station Operating License Condition 2.E stated, in part, that the licensee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the licensee's Fire Protection Report. Section 2.3.11.56, "Auxiliary Building Laundry Room – Elevation 426' 0" (Fire Zone 11.6C-0)," of the Fire Protection Report stated, in part, that "The [ceiling] slab is supported by steel beams and columns which are protected with a fire-resistant coating and carry a 3-hour fire rating." Contrary to the above, since initial construction, the structural steel beam in the auxiliary building laundry room was not fireproofed to a 3-hour fire rating as required by the Braidwood Station Fire Protection Report and Operating License Condition 2.E. As part of the licensee's immediate corrective actions, the licensee implemented compensatory measures that included hourly fire watches. Because this violation was of very low safety significance and because this issue was entered into the licensee's CAP as IR 120988, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000456/2011003-02, Structural Steel Beam Missing Fire Proofing Materials).

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of the Unit 2 component cooling water heat exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. The inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed are listed in the Attachment.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

From April 20, 2011, through May 6, 2011, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), steam generator tubes, emergency feedwater systems, risk significant piping and components, and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, R08.3, IR08.4, and 1R08.5 below constituted one inservice inspection sample as defined in IP 71111.08-05.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed and reviewed records for the following non-destructive examinations required by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements, and if any indications and/or defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Ultrasonic Testing (UT) of a risk-informed (R-A , R01.20), 4" Pipe-to-Elbow, Weld, 2RC-17-06, IMB Loop D;
- UT of a risk-informed (R-A , R01.20), 4" Elbow-to-Pipe, Weld, 2RC-17-07, IMB Loop D;
- UT of risk-informed (R-A, R01.11, R01.18), 16" Pipe-to-Elbow, Weld, 2FW-02-19, IMB Loop A, R41;

- UT of risk-informed (R-A, R01.11, R01.18), 16" Elbow-to-Pipe, Weld, 2FW-02-20, IMB Loop A, R41;
- UT of risk-informed (R-A, R01.11, R01.18), 16" Pipe-to-Elbow, Weld, 2FW-02-23, IMB Loop A, R42;
- VT-3 of ASME Class 1, Residual Heat Removal (RHR) Snubber, M-2RH02013S; and
- VT-3 of ASME Class 1, Chemical Volume and Control System (CVCS) Snubber, M-2CV02005S.

During the prior outage non-destructive surface and volumetric examinations, the licensee did not identify any relevant and/or recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems since the beginning of the last refueling outage to determine if the licensee applied the pre-service non-destructive examinations (NDE) and acceptance criteria required by the Construction Code and ASME Code, Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of the Construction Code and Section XI of the ASME Code.

- 2SI99G-1", 2" to 1" Reducer Coupling, Welds SW12 and SW13 (part of assembly of MOD for 1" Check Valve 2SI8969F), Code Class 2.

b. Findings

Failure to Perform Post VT-3 Examination Illumination Verification in Accordance with American Society of Mechanical Engineer's Code

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR 50.55a(g)4 when a licensee vendor examiner failed to perform VT-3 visual examinations in accordance with the ASME Code. Specifically the examiner failed to verify adequacy of illumination following a snubber VT-3 visual examination.

Description: On April 25, 2011, the inspectors identified through direct observation that an NDE examiner failed to perform a verification of the adequacy of illumination following VT-3 examinations of RHR system snubbers. The adequacy of illumination check was required by the VT-3 procedure the examiner was using and the 2001 Edition, through 2003 Addenda of ASME Code Section XI, Article IWA-2213(f), VT-3 Examination. Specifically, the ASME Code specified that "When a battery-powered light is used, the adequacy of the illumination level shall be checked before and after each examination or series of examinations, not to exceed four hours between checks." The Code required illumination checks provided assurance that an examiner is able to detect rejectable indications and/or conditions when performing visual examinations on safety-related components prior to returning those components to service.

When the inspectors asked the examiner why he did not perform the post examination illumination verification, he stated that he believed that while pre-examination verification checks were required, post-examination illumination checks were not. When asked by the inspectors if the Reference Use procedure being used required the verification, the

examiner stated he did not believe so. However, he did not refer to the procedure even though he had it available at the work location. Licensee procedure HU-AA-104-101, “Procedure Use and Adherence,” Revision 4, required, in part, that examiners read and understand Reference Use procedures prior to their use. Licensee Procedure ER-AA-335-016, “VT-3 Visual Examination of Component Supports, Attachments and Interiors of Reactor Vessels,” Revision 6 being used for the examination, the examiner’s VT-3 vendor procedure, and the licensee’s VT-1 procedure, all included the same illumination verification ASME Code requirements. The licensee’s subsequent extent of condition review showed that the examiner had performed essentially all of the VT-3 snubber examinations, but that he had not performed any VT-1 examinations at Braidwood during the refueling outage. The licensee subsequently re-performed 18 visual examinations and entered this issue into the CAP as IR 1208643.

Analysis: The inspectors determined that the licensee examiner’s failure to verify the adequacy of the illumination level following the examination of an RHR system snubber was contrary to the ASME Code Section XI, Article IWA-2213 and was a performance deficiency.

The finding was determined to be more than minor because the finding was associated with the Equipment Performance (Reliability) attribute of the Mitigating Systems Cornerstone 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). Absent NRC identification, the licensee would not have performed the ASME Code-required examinations for a number of components, which could have allowed a rejectable condition to go undetected. The licensee entered this issue into the CAP as IR 1208643 and, as a result of an extent of condition evaluation, re-performed 18 visual examinations.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 4, “Phase I - Initial Screening and Characterization of findings,” Table 4a for the Mitigating Systems Cornerstone and answered “No” to the Mitigating Systems Cornerstone questions. Specifically, the issue did not result in the actual loss of the operability or functionality of a safety system. Therefore, the finding screened as having very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area (H.4(b)) because the licensee did not effectively communicate expectations regarding procedural compliance and licensee personnel did not follow procedures.

Enforcement: Title 10 CFR 50.55a(g)4 required, in part, that throughout the service life of a pressurized water-cooled nuclear power facility, components (including supports) must meet the requirements set forth in the ASME Code Section XI.

The 2001 Edition, through 2003 Addenda of ASME Code Section XI, Article IWA-2213(f), VT-3 Examination, requires that when a battery-powered light is used, the adequacy of the illumination level shall be checked before and after each examination or series of examinations, not to exceed 4 hours between checks.

Contrary to the above, on April 25, 2011, while performing a VT-3 examination using procedure ER-AA-335-016, Revision 6, on an RHR system snubber (M-2RH02013S), the licensee examiner failed to check the illumination levels of the battery powered light after the VT-3 examination of the snubber as required by the 2001 Edition, through 2003 Addenda of ASME Code Section XI, Article IWA-2213(f). As a result of an extent of condition evaluation, the licensee re-performed 18 visual examinations.

Because this violation was of very low safety significance and because the issue was entered into the licensee's CAP as IR 1208643, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000457/2011003-03: Failure to Perform Post VT-3 Examination Illumination Verification in Accordance with ASME Code)**.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 2 reactor vessel head, a bare metal visual (BMV) examination and a non-visual examination were required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors observed a recording of the BMV examination conducted on the reactor vessel head at each of the penetration nozzles to determine if the activities were conducted in accordance with the requirements of ASME Code Case (CC) N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, this inspectors determined:

- If the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with the licensee procedures;
- If the licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and
- If indications of potential through-wall leakage were identified, that the licensee entered the condition into the CAP and implemented appropriate corrective actions.

The inspectors observed a number of non-visual examinations conducted on the reactor vessel head penetrations to determine if the activities were conducted in accordance with the requirements of ASME CC N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, the inspectors determined:

- If the required examination scope (volumetric and surface coverage) was achieved and limitations (if applicable) were recorded in accordance with licensee procedures;
- If the UT examination equipment and procedures used were demonstrated by blind demonstration testing;
- If indications or defects were identified, that the licensee documented the conditions in examination reports and/or entered this condition into the CAP and implemented appropriate corrective actions; and
- If indications were accepted for continued service, that the licensee evaluation and acceptance criteria were in accordance with the ASME Section XI Code, 10 CFR 50.55a(g)(6)(ii)(D), or an NRC-approved alternative.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage for Unit 2. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

On April 18, 2011, the inspectors observed the licensee staff performing visual testing examinations of the Unit 2 RCS within containment to determine if these examinations focused on locations where boric acid leaks could cause degradation of safety-significant components.

The inspectors reviewed the following licensee evaluations of RCS components with boric acid deposits to determine if degraded components were documented in the CAP. The inspectors also evaluated corrective actions for any degraded RCS components to determine if they met the component construction code, ASME Section XI Code, and/or NRC-approved alternative.

- Boric Acid Evaluation (BAE) 00981057; 2FE-0160, RCP 2B Hi Ring Leak Orifice Plate; October 18, 2009;
- BAE 0981011; 2FE-0162, RCP [Reactor Coolant Pump] 2B No. 1 Seal Bypass Flow Element; October 18, 2009;
- BAE 0861305; 2RC8037C, RC [Reactor Coolant] Loop 2C PMP Suction Leg Drain Isolation Valve Assembly; June 15, 2010; and
- BAE 0983679; 2RH607, RH HX [Heat Exchanger] 2B Flow Control Valve Assembly; March 23, 2010.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- IR 975127; Dry Boric Acid at 2SI8835 Packing; October 5, 2009;
- IR 976122; Boric Acid Leakage (Flange Connection 2PI-0601/2RH06AB-0.5); October 7, 2009; and
- IR 978933; Dry Boric Acid Body/Cover 2CV8378A; October 14, 2009;

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition of eddy current testing (ET) data, interviewed ET data personnel, and reviewed documentation related to the steam generator (SG) ISI program to determine if:

- In-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) Document 1014983, “Steam Generator In-Situ Pressure Test Guidelines,” and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- In-situ pressure test records demonstrated pressure and hold times consistent with EPRI Document 1014983, “In-situ Pressure Test Guidelines”;
- In-situ pressure test results were properly applied to SG tube integrity performance criteria identified in EPRI Document 1019038;
- The numbers and sizes of SG tube flaws/degradation identified was consistent with the licensee’s previous outage Operational Assessment predictions;
- The SG tube ET examination scope and expansion criteria were sufficient to meet the TSs, and EPRI Document 1013706, “Pressurized Water Reactor Steam Generator Examination Guidelines”;
- The SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- The licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- The licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- The licensee implemented an inappropriate “plug on detection” tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- The licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons-per-day or the detection threshold during the previous operating cycle;
- The ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known and/or expected types of SG tube degradation in accordance with Appendix H, “Performance Demonstration for Eddy Current Examination,” of EPRI Document 1013706, “Pressurized Water Reactor Steam Generator Examination Guidelines”;
- The licensee performed secondary side SG inspections for location and removal of foreign materials;
- The licensee implemented repairs for SG tubes damaged by foreign material; and
- Foreign objects were left within the secondary side of the SGs, and if so, that the licensee implemented evaluations which included the effects of foreign object migration and/or tube fretting damage.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no NRC review was completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if;

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

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. Documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On June 21, 2011, the inspectors observed Crew #3 respond to a simulated feedwater transient followed by a large break Loss-of-Coolant-Accident, then a station blackout (SBO) in the plant simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment.

This inspection constituted one quarterly licensed operator requalification program sample as defined in IP 71111.11.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Diesel and Motor Driven Fire Pumps; and
- Carbon Dioxide Fire Suppression System.

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

This inspection constituted two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Unit 2 “A” RHR System Train Work Window, Planned Yellow;
- Unit 2 Reduced Inventory Operations and Reactor Head Lift, Planned Yellow;
- DC [Direct Current] Bus 111 Cross-Tied to DC Bus 211, Planned Yellow;
- June 8, 2011, Essential Service Water Temporary Hose Rupture in Auxiliary Building during Backwashing Activity, Unplanned Orange; and
- Unit 1 “A” Feedwater Regulating Valve Card Replacement following Controlling Card Failure, Unplanned Operational Risk.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed are listed in the Attachment.

These maintenance risk assessments and emergent work control activities constituted five samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- High Energy Line Break Analysis Input Error, IR 1185015;
- Postulated Void in Auxiliary Feedwater from the Failure of Condensate Storage Tank (CST) Piping, IR 1202772;
- Generic Letter 96-06 Review, Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions, IR 1185022;

- Asiatic Clams Found In Stagnate Essential Service Water Supply to Auxiliary Feedwater Pumps, IR 1194353; and
- Calorimetric Flow Constant Error and Effect on the Power Range Nuclear Instruments, IR 1217217.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sample of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment.

These operability activities constituted five samples as defined in IP 71111.15-05.

b. <u>Findings</u>

<u>Failure to Ensure that the Design of the Auxiliary Feedwater Suction Piping was Adequate to Prevent Air Entrainment Following a Seismic or Tornado Event</u>

<u>Introduction</u>: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," when licensee personnel failed to analyze whether the design of the AF system ensured that air entrained into the system following a postulated seismic or tornado event did not prevent the system from performing its safety function.

<u>Description</u>: The function of the AF system is to provide adequate cooling water to the steam generators during certain abnormal or accident events. The AF pumps are normally aligned to take suction from the CST. Section 10.D.3.4, "NRC Recommendation GL-4," of the UFSAR, stated, in part:

> "To prevent air binding of the auxiliary feedwater pumps, switchover from the condensate storage tank supply to the essential service water system occurs when low pressure is detected on the suction side. Pressure switches are installed on all four auxiliary feedwater pumps. The switches function to: 1) alarm low AF pump suction pressure in the main control room, 2) switch the AF pump suction source from the CST to the essential service water [SX] system, and 3) trip the respective AF pump on low suction pressure to prevent damage to the pump."

Switchover from the CST to the SX system is automatically accomplished on low pressure (18.1 pound per square inch absolute (psia)) in the suction pipe to the AF pumps. The AF pumps will trip when the low-low pressure setpoint of 16.5 psia for longer than 2.5 seconds is reached.

The inspectors identified a scenario in which the AF switchover setpoint and pump trip logic used to prevent air binding had not been previously evaluated and was questionable. Specifically, the inspectors identified that if the non-seismically qualified portion of the CST suction piping catastrophically failed due to a tornado or seismic event, the AF suction pressure would likely decrease below the low pressure (suction switchover) and low-low pressure (pump trip) setpoints. The inspectors determined the pumps would remain running for 2.5 seconds with a flow velocity of about 11 feet per second and that this would potentially result in air being entrained into the AF pumps before the pumps tripped on low-low pressure. Then, as the switchover valves opened, the pump suction pressure would increase to 17 psia, the pump restart setpoint. However, because the motor-driven AF pump can accelerate to full speed in about 1 second, this pump start could result in suction pressure fluctuations causing pressure to decrease below the low-low pressure setpoint (pump trip) and then increase above the pump restart setpoint. In addition, as suction pressure decreases, the check valve in the seismically-qualified portion of the piping from the CST may open resulting in more air being introduced into the system. At some point, the switchover valves would open sufficiently to support continuous pump operation and maintain the suction piping pressurized such that the CST check valve would remain closed. The licensee indicated that the pumps were expected to trip and restart up to four times on a complete loss of CST head.

The licensee captured the inspectors' concerns in their CAP as IR 1202772, and performed an operability evaluation of the AF suction piping from the CST due to an impact from a seismic event or a tornado missile. The licensee's evaluation addressed the piping in the turbine and auxiliary buildings as well as the buried piping from the CST to the turbine building. The evaluation concluded that the piping was operable, but non-conforming. Specifically, the evaluation concluded that the piping would remain operable under a design basis seismic event and would not be adversely affected by the failure of other adjacent piping, equipment, or structures. The evaluation also concluded that the piping location and the surrounding structure, including concrete floors and walls, provided adequate protection from a potential tornado missile impact. The corrective actions that were being considered by the licensee at the end of this inspection were to determine the required changes to the design basis documentation and/or plant hardware to restore the design basis of the AF system.

<u>Analysis</u>: The inspectors determined that the failure to analyze whether air entrained into the AF system following a postulated seismic or tornado event would prevent the system from performing its safety function was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency.

The performance deficiency was determined to be more than minor because it was associated with the Protection Against External Events attribute of the Mitigating Systems Cornerstone 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). Specifically, the inspectors had reasonable doubt on the operability of the AF system because its design did not ensure that air would not enter the system following a seismic or tornado event. The failure of the AF design to ensure that the system will not experience significant air entrainment could result in air binding or degraded performance of the AF pumps and, therefore, did not ensure the availability, reliability, and capability of the AF system.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 4, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems Cornerstone and answered “No” to the Mitigating Systems Cornerstone questions. Specifically, the finding was determined to be of very low safety significance (Green) because the finding involved a design or qualification deficiency that did not result in a loss of operability or functionality since the piping would remain operable during a seismic event and was adequately protected from a tornado missile impact.

The inspectors determined that there was no cross-cutting aspect associated with this finding because it did not reflect current performance due to the age of the performance deficiency.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that design control measures shall provide for verifying or checking the adequacy of design.

Contrary to the above, as of April 7, 2011, the licensee's design control measures failed to verify the adequacy of the AF design. Specifically, licensee personnel failed to ensure that air entrained into the AF system as a result of failed non-seismically qualified condensate storage tank suction piping following a postulated design basis seismic or tornado event would not prevent the AF system from performing its safety function, as required. As part of the licensee’s immediate corrective actions, an operability evaluation was performed that concluded the AF system was operable, but non-conforming. Because this violation was of very low safety significance and it was entered into the licensee’s CAP as IR 1202772, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000456/2011003-04; NCV 05000457/2011003-04, Failure to Ensure that the Design of the Auxiliary Feedwater Suction Piping Was Adequate to Prevent Air Entrainment Following a Seismic or Tornado Event)

.2 Potential Design Control Violation Related to Safety-Related Door Impairment

Introduction: The inspectors identified an Unresolved Item (URI) related to the licensee’s control of barrier doors during activities that involved the transport of equipment between spaces. Specifically, the licensee’s barrier impairment program permitted barrier doors to be open for up to 30 minutes for the transport of station equipment without the performance of an evaluation. At the conclusion of this inspection period, the licensee had not provided a regulatory basis for the allowance.

Description: While reviewing IR 1185016, “Non-Conservatisms in the Turbine Building HELB [High Energy Line Break] Analyses,” the inspectors questioned the policy for moving equipment through a door that protects 10 CFR 50, Appendix B, safety-related systems and components from the effects of fire, flooding, and a high energy line break, or that provides a ventilation barrier needed to support a safety function.

Step C.10 of BwAP 1110-03, “Plant Barrier Impairment Program,” Revision 20, stated “Doors MAY be opened without a PBI (Plant Barrier Impairment) PERMIT during normal passage (30 minutes maximum) of personnel or equipment. The door SHOULD be closed at termination of attendance. If the door must be blocked or tied open, then a PBI PERMIT SHALL be required. Plant alarms or special controls SHOULD be considered before holding a door open.”

When actions were implemented to address the non-conservatism issues in IR 1185016, the licensee issued Revision 21 to BwAP 1110-03, which added the phrase “With the exception of HELB doors…” to the beginning of Step C.10, which had the unintended effect of prohibiting all passage through HELB doors without a PBI evaluation. This issue was identified by the inspectors and was entered into the licensee’s CAP.

Procedure CC-AA-201, “Plant Barrier Control Program,” Revision 6, defined an impaired barrier as, “A barrier that is inoperable such that it cannot fully perform its intended design function.” Regulatory Guideline 1.189, “Fire Protection for Nuclear Power Plants,” Revision 1, defined an impairment as, “The degradation of a fire protection system or feature that adversely affects the ability of the system or feature to perform its intended function.”

Additionally, the inspectors were not convinced that the licensee had established adequate measures to ensure design basis events (e.g., HELB, fire, flooding, etc.) would not impact safety-related or augmented quality structures, systems, and components in a unacceptable manner as required by 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” For instance, the licensee had not provided adequate assurance that a HELB door would shut, post event, if the licensee was moving a large piece of equipment through a door (e.g. breakers, scaffold poles, hoses, etc).

At the end of the inspection period, it was unclear if the licensee’s policy was an adequate design control measure with an acceptable exception. The scope of this issue was limited to the transport and passage of equipment, and did not apply to personnel.

This URI will remain open pending a more detailed review of the licensee’s policy, NRC regulatory requirements, and accepted standards and practices.
(URI 05000456/2011003-05; 05000457/2011003-05, Potential Design Control Violation Related to Safety-Related Door Impairment)

.3 Asiatic Clams Identified in the Essential Service Water System Supply to the Auxiliary Feedwater System

Introduction: The inspectors identified an URI related to the licensee’s discovery of small clam shells in a portion of the 2A AF system train. At the conclusion of the inspection period, the licensee was still investigating the root cause of the event.

Description: On March 30, 2011, and May 9, 2011, the licensee discovered Asiatic clam shells (Corbicula fluminea) in a portion of the 2A AF system train. The shells were removed from the system upon discovery, but resulted in an 8-hour reportable event (EN 46868) because there were periods when the 2B AF train was not operable concurrent with this condition existing in the 2A train. The licensee concluded that although a more detailed review of past operability was still in progress, the size and quantity of shells removed from the piping on May 9, 2011, was greater than could readily be evaluated as operable without a more detailed analysis.

The licensee identified that the shells were asiatic clams; common macro-fouling organisms in power station raw water systems. The licensee determined that clams of any species that are not commonly found in the SX system and their accumulation in the SX to AF crosstie suction piping was not consistent with other NRC Generic Letter (GL) 89-13 “Service Water System Problems Affecting Safety-Related Equipment” inspection results.

At the end of the inspection period, the licensee was in the process of completing a root cause evaluation. The licensee concluded that the shell relics identified in the 2A AF system could be traced back to specific SX and chemical feed system conditions that existed in the 1990's.

This URI will remain open pending an inspector review of the completed root cause evaluation, NRC Generic Letter 89-13 commitment review, and past operability and availability review. **(URI 05000456/2011003-06; 05000457/2011003-06, Asiatic Clams Identified in the Essential Service Water System Supply to the Auxiliary Feedwater System)**

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed the following post-maintenance testing (PMT) activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Reactor Building Drain Containment Isolation Valve Pilot Valve Replacement;
- Pressurizer Power Operated Relief Valve Air Operator Diaphragm Replacement;
- 2A Emergency Diesel Generator Maintenance;
- Unit 2 Containment Polar Crane Load Cell Replacement; and
- 1A and 2A Auxiliary Feedwater Pump Testing after becoming Wetted from an Essential Service Water Flushing Line Hose Failure.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with the post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment.

This inspection constituted five post-maintenance testing samples as defined in IP 71111.19-05.

b. Findings

Failure to Follow Procedure During Reactor Vessel Head Lift

Introduction: A finding of very low safety significance (Green) and an associated NCV of TS 5.4.1 was self-revealed on April 21, 2011, when licensee personnel failed to suspend a reactor vessel head lift after it became apparent that there was a large deviation between the crane's actual load cell indication and the expected indication. The failure to suspend the head lift was contrary to procedural requirements.

Description: On April 21, 2011, during the initial lift of the Unit 2 reactor vessel head to support refueling outage activities, the individual monitoring the load cell observed that the load cell increased as a negative number and continued to increase until the load cell stabilized at negative (-) 138,000 pounds, at which point the reactor vessel head was approximately 3 to 5 inches above the flange. The expected weight of the reactor head was positive (+) 380,000 pounds. The lift was temporarily stopped and the field supervisor contacted the site reactor services manager and informed him that the load cell was indicating a negative number. In addition, the control room unit supervisor was notified that there was an issue with the load cell.

The site reactor services manager and the technical director in charge of the lift then proceeded to discuss the load cell discrepancy to understand why the load cell indication was negative. They also discussed the guidance referenced in procedure BwMP 3100-009, "Reactor Vessel Closure Head Removal," Revision 20. This procedure included a caution statement that stated, "Any sudden increase in load would indicate binding or interference," and "During the removal of the reactor head, monitor the load cell reading. If there is a drastic change, stop lifting until the reason has been resolved." The licensee's staff viewed this caution as being applicable to the monitoring of the load for sudden changes that could indicate potential binding and not to apply to the negative value of the load cell.

Because the load cell appeared to be responding to load changes, but in the negative direction, the reactor services site director instructed the floor manager to continue the lift until the next holding point, which was at a head elevation of 18 inches from the flange, in order to perform scheduled flange inspections. The decision was based on the belief that the load cell could still be used to monitor for binding. The licensee also believed that it was prudent to not lower the load since without an accurate load cell, binding may inadvertently damage the head during the lowering process. In addition, the licensee concluded that by raising the head they could also monitor for binding with the use of cameras mounted on the floor of the refueling cavity. Once the procedure-required hold point of 18 inches was reached, licensee personnel agreed to contact the vendor to determine if the negative indication could be addressed. After the reactor vessel head was lifted from the 3 to 5 inch hold point to the 18 inch hold point, the shift manager was informed of the situation. The shift manager directed the lift to be suspended until further information was gathered and understood.

After the lift was put on hold, the licensee had discussions with the vendor regarding the load cell discrepancy. The vendor stated that based on the information provided by the licensee, the load cell was not reliable and did not recommend using the load cell as a way to determine if binding was occurring. Based on this information, the licensee decided that setting the reactor vessel head back onto the flange and replacing the load

cell was the safest path. The reactor head was subsequently set back onto the flange, the load cell was replaced, retested, and the lift was recommenced and performed without further incident.

The licensee's subsequent troubleshooting uncovered that the reason why the load cell was providing a negative indication was that a 5-foot cable was purchased as a replacement to the original 20-foot cable that was supplied by the vendor with the load cell. This modified cable length caused the negative load cell indication. The licensee subsequently tested the same load cell with the vendor-specified cable and it responded in the expected manner.

Analysis: The inspectors determined that the failure to adhere to BwMP 3100-009, "Reactor Vessel Closure Head Removal," Revision 20, during the Unit 2 reactor vessel head lift was a performance deficiency. The issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Human Performance attribute of the Barrier Integrity Cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, "Significance Determination Process," Appendix G, "Shutdown Operations Significance Determination Process." The inspectors used Checklist 3 contained in Attachment 1 and determined that the finding was Green since it did not require a Phase 2 or Phase 3 analysis.

The inspectors determined that this finding had a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area (H.1(b)) because licensee personnel failed to use conservative assumptions in decision making after identifying a large deviation between actual and expected load cell indications during a head lift evolution.

Enforcement: Technical Specification 5.4.1 requires that written procedures be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Step 9.d.6 of Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978, required that written procedures be established, implemented and maintained regarding the removal of the reactor vessel head. Licensee procedure HU-AA-104-101, "Procedure Use and Adherence," Revision 4, required procedure users to, "Follow the procedure exactly as written," and to, "Comply with all applicable Precautions, Limitations and Prerequisites." Procedure BwMP 3100-009, "Reactor Vessel Closure Head Removal," Revision 20, included a caution statement that stated, "During the removal of the reactor head, monitor the load cell reading, if there is a drastic change, stop lifting until the reason has been resolved."

Contrary to the above, on April 21, 2011, the licensee failed to adhere to BwMP 3100-009, Revision 20, when the load cell being used to lift the reactor vessel head indicated a drastic change between the actual and the expected reading. The licensee did temporarily stop the lifting activity, but then decided to continue until the next pre-established hold point in the procedure, which was 18 inches. This action was contrary to the procedure caution step that required suspending the lift until the issue was resolved. Corrective actions for this issue included suspending the lift,

resetting the head on the flange, and replacing the load cell. Because this violation was of very low safety significance and was entered into the CAP as IR 1206020, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000457/2011003-07, Failure to Follow Procedure During Reactor Vessel Head Lift)**

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan and contingency plans for the Unit 2 refueling outage, conducted April 17 – May 11, 2011, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed are listed in the Attachment.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the Outage Safety Plan for key safety functions and compliance with the applicable TS when taking equipment out of service;
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and Outage Safety Plan requirements were met;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity;
- Maintenance of secondary containment as required by TS;
- Refueling activities;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- Licensee identification and resolution of problems related to refueling outage activities.

This inspection constituted one refueling outage sample as defined in IP 71111.20-05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- BwMSR 3.7.1.1, Revision 2, Unit 2 Main Steam Safety Valve Trevi Testing; April 14 (Routine);
- 1B Emergency Diesel Generator Monthly Surveillance; April 13 (Routine);
- Unit 2 Containment Penetration Status Weekly Surveillance; May 4 (Containment Isolation Valve);
- 2BwOL 3.3.1, Resistance Temperature Detector 421 and 422, 18-Month Calibration (LIV); June 1 (Routine); and
- Essential Service Water to Fire Protection Cross-Tie Piping Flush; June 17 (Routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, ASME Code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;

- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment.

This inspection constituted four routine surveillance testing samples, and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted a partial sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

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

- reactor head disassembly and reassembly;
- scaffold outage activities;
- steam generator platform and bullpen setup and teardown/decontamination;
- lead shielding installation and removal; and
- seal table room outage activities.

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

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

- reactor head disassembly and reassembly;
- scaffold outage activities; and
- steam generators platform and bullpen setup and teardown/decontamination.

For these radiation work permits, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material), and lead shielding installation and removal were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm setpoints were in conformance with survey indications and plant policy.

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

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

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

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

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

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

- reactor head disassembly and reassembly;
- scaffold outage activities;
- steam generator platform and bullpen setup and teardown/decontamination;

- lead shielding installation and removal; and
- seal table room outage activities.

For these radiation work permits, the inspectors evaluated airborne radioactive controls and monitoring, including the potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entries into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

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

b. Findings

No findings were identified.

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.6 Radiation Worker Performance (02.07)

a. Inspection Scope

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

b. Findings

No findings were identified.

.7 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

b. Findings

No findings were identified.

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

This inspection constituted a partial sample as defined in IP 71124.02-05.

.1 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors reviewed the assumptions and basis (including dose rate and resource estimates) for the current annual collective exposure estimate for reasonable accuracy for select As-Low-As-Is-Reasonably-Achievable (ALARA) work packages. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

b. Findings

No findings were identified.

.2 Source Term Reduction and Control (02.04)

a. Inspection Scope

The inspectors used licensee records to determine the historical trends and current status of significant tracked plant source terms known to contribute to elevated facility aggregate exposure. The inspectors assessed whether the licensee had made allowances or developed contingency plans for expected changes in the source term as the result of changes in plant fuel performance issues or changes in plant primary chemistry.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

The inspection constitute one complete sample as defined in IP 71124.03-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the plant UFSAR to identify areas of the plant designed as potential airborne radiation areas and any associated ventilation systems or airborne monitoring instrumentation. Instrumentation review included continuous air monitors (continuous air monitors and particulate-iodine-noble-gas-type instruments) used to identify changing airborne radiological conditions such that actions to prevent an overexposure may be taken. The review included an overview of the respiratory protection program and a description of the types of devices used. The inspectors reviewed UFSAR, TS, and emergency planning documents to identify the location and quantity of respiratory protection devices stored for emergency use.

The inspectors reviewed the licensee’s procedures for maintenance, inspection, and use of respiratory protection equipment including self-contained breathing apparatus, as well as procedures for air quality maintenance.

The inspectors reviewed reported performance indicators to identify any related to unintended dose resulting from intakes of radioactive material.

b. Findings

No findings were identified.

.2 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed the licensee’s use of permanent and temporary ventilation to determine whether the licensee used ventilation systems as part of its engineering controls (in lieu of respiratory protection devices) to control airborne radioactivity. The inspectors reviewed procedural guidance for use of installed plant systems, such as containment purge, spent fuel pool ventilation, and auxiliary building ventilation, and assessed whether the systems were used, to the extent practicable, during high-risk activities (e.g., using containment purge during cavity flood-up).

The inspectors selected installed ventilation systems used to mitigate the potential for airborne radioactivity, and evaluated whether the ventilation airflow capacity, flow path (including the alignment of the suction and discharges), and filter/charcoal unit efficiencies, as appropriate, were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable.

The inspectors selected temporary ventilation system setups used to support work in contaminated areas. The inspectors assessed whether the use of these systems was consistent with licensee procedural guidance and ALARA concepts.

The inspectors reviewed airborne monitoring protocols by selecting installed systems used to monitor and warn of changing airborne concentrations in the plant and evaluating whether the alarms and set-points were sufficient to prompt licensee/worker action to ensure that doses were maintained within the limits of 10 CFR Part 20 and ALARA concepts.

The inspectors assessed whether the licensee had established trigger points (e.g., the Electric Power Research Institute's "Alpha Monitoring Guidelines for Operating Nuclear Power Stations") for evaluating levels of airborne beta-emitting (e.g., plutonium-241) and alpha-emitting radionuclides.

b. Findings

No findings were identified.

.3 Use of Respiratory Protection Devices (02.03)

a. Inspection Scope

For those situations where it was impractical to employ engineering controls to minimize airborne radioactivity, the inspectors assessed whether the licensee provided respiratory protective devices such that occupational doses were ALARA. The inspectors selected work activities where respiratory protection devices were used to limit the intake of radioactive materials, and assessed whether the licensee performed an evaluation concluding that further engineering controls were not practical and that the use of respirators was ALARA. The inspectors also evaluated whether the licensee had established means (such as routine bioassay) to determine if the level of protection (protection factor) provided by the respiratory protection devices during use was at least as good as that assumed in the licensee's work controls and dose assessment.

The inspectors assessed whether respiratory protection devices used to limit the intake of radioactive materials were certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration or had been approved by the NRC per 10 CFR 20.1703(b). The inspectors selected work activities where respiratory protection devices were used. The inspectors evaluated whether the devices were used consistent with National Institute for Occupational Safety and Health/Mine Safety and Health Administration certification or any conditions of their NRC approval.

The inspectors reviewed records of air testing for supplied-air devices and self-contained breathing apparatus bottles to assess whether the air used in these devices met or exceeded Grade D quality. The inspectors reviewed plant breathing air supply systems to determine whether they met the minimum pressure and airflow requirements for the devices in use.

The inspectors selected several individuals qualified to use respiratory protection devices, and assessed whether they had been deemed fit to use the devices by a physician.

The inspectors selected several individuals assigned to wear a respiratory protection device and observed them donning, doffing, and functionally checking the device as appropriate. Through interviews with these individuals, the inspectors evaluated

whether they knew how to safely use the device and how to properly respond to any device malfunction or unusual occurrence (loss of power, loss of air, etc.).

The inspectors chose multiple respiratory protection devices staged and ready for use in the plant or stocked for issuance for use. The inspectors assessed the physical condition of the device components (mask or hood, harnesses, air lines, regulators, air bottles, etc.) and reviewed records of routine inspection for each. The inspectors selected several of the devices and reviewed records of maintenance on the vital components (e.g., pressure regulators, inhalation valves, exhalation valves, hose couplings). The inspectors assessed whether onsite personnel assigned to repair vital components had received vendor-provided training.

b. Findings

No findings were identified.

.4 Self-Contained Breathing Apparatus for Emergency Use (02.04)

a. Inspection Scope

Based on the UFSAR, TS, and emergency operating procedure requirements, the inspectors reviewed the status and surveillance records of self-contained breathing apparatuses staged in-plant for use during emergencies. The inspectors reviewed the licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions.

The inspectors selected several individuals on control room shift crews and from designated departments currently assigned emergency duties (e.g., onsite search and rescue duties) to assess whether control room operators and other emergency response and radiation protection personnel (assigned in-plant search and rescue duties or as required by emergency operating procedures or the emergency plan) were trained and qualified in the use of self-contained breathing apparatuses (including personal bottle change-out). The inspectors evaluated whether personnel assigned to refill bottles were trained and qualified for that task.

The inspectors reviewed the past 2 years of maintenance records for select self-contained breathing apparatus units used to support operator activities during accident conditions and designated as "ready for service" to assess whether any maintenance or repairs on any self-contained breathing apparatus unit's vital components were performed by an individual, or individuals, certified by the manufacturer of the device to perform the work. The vital components typically included the pressure-demand air regulator and the low-pressure alarm. The inspectors reviewed the onsite maintenance procedures governing vital component work to determine any inconsistencies with the self-contained breathing apparatus manufacturer's recommended practices. For those self-contained breathing apparatuses designated as "ready for service," the inspectors determined whether the required, periodic air cylinder hydrostatic testing was documented and up to date, and the retest air cylinder markings required by the U.S. Department of Transportation were in place.

The inspectors determined whether appropriate mask sizes and types were available for use (i.e., in-field mask size and type match what was used in fit-testing). The inspectors determined whether on-shift operators had facial hair that would interfere with the sealing of the mask to the face and whether vision correction (e.g., glasses inserts or corrected lenses) was available as appropriate.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.05)

a. Inspection Scope

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed whether the corrective actions were appropriate for a selected sample of problems involving airborne radioactivity and were appropriately documented by the licensee.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

This inspection constituted one complete sample as defined in IP 71124.04-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the results of radiation protection program audits related to internal and external dosimetry (e.g., licensee's quality assurance audits, self-assessments, or other independent audits) to gain insights into overall licensee performance in the area of dose assessment and focus the inspection activities consistent with the principle of "smart sampling."

The inspectors reviewed the most recent National Voluntary Laboratory Accreditation Program (NVLAP) accreditation report on the vendor's most recent results to determine the status of the contractor's accreditation.

A review was conducted of the licensee procedures associated with dosimetry operations, including issuance/use of external dosimetry (routine, multi-badging, extremity, neutron, etc.), assessment of internal dose (operation of whole body counter, assignment of dose based on derived air concentration-hours, urinalysis, etc.), and evaluation of and dose assessment for radiological incidents (distributed contamination, hot particles, loss of dosimetry, etc.).

The inspectors evaluated whether the licensee had established procedural requirements for determining when external and internal dosimetry was required.

b. Findings

No findings were identified.

.2 External Dosimetry (02.02)

a. Inspection Scope

The inspectors evaluated whether the licensee's dosimetry vendor was NVLAP accredited and if the approved irradiation test categories for each type of personnel dosimeter used were consistent with the types and energies of the radiation present and the way the dosimeter was being used (e.g., to measure deep dose equivalent, shallow dose equivalent, or lens dose equivalent).

The inspectors evaluated the onsite storage of dosimeters before their issuance, during use, and before processing and/or reading. The inspectors also reviewed the guidance provided to radiation workers with respect to the care and storage of dosimeters.

The inspectors assessed the use of active dosimeters (electronic personal dosimeters) to determine if the licensee used a "correction factor" to address the response of the electronic personal dosimeter as compared to the passive dosimeter for situations when the electronic personal dosimeter must be used to assign dose and whether the correction factor was based on sound technical principles.

The inspectors reviewed dosimetry occurrence reports or CAP documents for adverse trends related to electronic personal dosimeters, such as interference from electromagnetic frequency, dropping or bumping, failure to hear alarms, etc. The inspectors assessed whether the licensee had identified any trends and implemented appropriate corrective actions.

b. Findings

No findings were identified.

.3 Internal Dosimetry (02.03)

(1) Routine Bioassay (In Vivo)

a. Inspection Scope

The inspectors reviewed procedures used to assess the dose from internally deposited nuclides using whole body counting equipment. The inspectors evaluated whether the procedures addressed methods for differentiating between internal and external contamination, the release of contaminated individuals, the route of intake and the assignment of dose.

The inspectors reviewed the whole body count process to determine if the frequency of measurements was consistent with the biological half-life of the nuclides available for intake.

The inspectors reviewed the licensee's evaluation for use of its portal radiation monitors as a passive monitoring system to determine if instrument minimum detectable activities

were adequate to determine the potential for internally deposited radionuclides sufficient to prompt additional investigation.

The inspectors selected several whole body counts and evaluated whether the counting system used had sufficient counting time and sufficiently low background to ensure appropriate sensitivity for the potential radionuclides of interest. The inspectors reviewed the radionuclide library used for the count system to determine its appropriateness. The inspectors evaluated whether anomalous count peaks and/or nuclides indicated in each output spectra received appropriate disposition. The inspectors reviewed the licensee's 10 CFR Part 61 data analyses to determine whether the nuclide libraries included appropriate gamma-emitting nuclides. The inspectors evaluated whether the licensee adequately accounted for hard-to-detect nuclides in the dose assessment.

b. Findings

No findings were identified.

(1) Special Bioassay (In Vitro)

a. Inspection Scope

There was no internal dose assessments obtained using in vitro monitoring for the inspectors to review. The inspectors reviewed and assessed the adequacy of the licensee's program for in vitro monitoring (i.e., urinalysis and fecal analysis) of radionuclides (tritium, fission products, and activation products), including collection and storage of samples. The inspectors reviewed the vendor laboratory quality assurance program and assessed whether the laboratory participated in an industry recognized cross-check program including whether out-of-tolerance results were resolved appropriately.

b. Findings

No findings were identified.

(1) Internal Dose Assessment – Airborne Monitoring

a. Inspection Scope

The inspectors reviewed the licensee's program for airborne radioactivity assessment and dose assessment, as applicable, based on airborne monitoring and calculations of derived air concentration. The inspectors determined whether flow rates and collection times for air sampling equipment were adequate to allow lower limits of detection to be obtained. The inspectors also reviewed the adequacy of procedural guidance to assess internal dose if respiratory protection was used.

b. Findings

No findings were identified.

(1) Internal Dose Assessment – Whole Body Count Analyses

a. Inspection Scope

The inspectors reviewed several dose assessments performed by the licensee using the results of whole body count analyses. The inspectors determined whether affected personnel were properly monitored with calibrated equipment and that internal exposures were assessed consistent with the licensee's procedures.

b. Findings

No findings were identified.

.4 Special Dosimetric Situations (02.04)

(1) Declared Pregnant Workers

a. Inspection Scope

The inspectors assessed whether the licensee informed workers, as appropriate, of the risks of radiation exposure to the embryo and fetus, the regulatory aspects of declaring a pregnancy, and the specific process to be used for (voluntarily) declaring a pregnancy.

The inspectors selected individuals who had a declared pregnancy during the current assessment period and evaluated whether the licensee's radiological monitoring program (internal and external) for declared pregnant workers was technically adequate to assess the dose to the embryo and fetus. The inspectors reviewed exposure results and monitoring controls employed by the licensee with respect to the requirements of 10 CFR Part 20.

b. Findings

No findings were identified.

(1) Dosimeter Placement and Assessment of Effective Dose Equivalent for External Exposures

a. Inspection Scope

The inspectors reviewed the licensee's methodology for monitoring external dose in non-uniform radiation fields or where large dose gradients existed. The inspectors evaluated the licensee's criteria for determining when alternate monitoring, such as use of multi-badging, was to be implemented.

The inspectors reviewed dose assessments performed using multi-badging to evaluate whether the assessment was performed consistent with licensee procedures and dosimetric standards.

b. Findings

No findings were identified.

(1) Shallow Dose Equivalent

a. Inspection Scope

The inspectors reviewed shallow dose equivalent dose assessments for adequacy. The inspectors evaluated the licensee's method (e.g., VARSKIN or similar code) for calculating shallow dose equivalent from distributed skin contamination or discrete radioactive particles.

b. Findings

No findings were identified.

(1) Neutron Dose Assessment

a. Inspection Scope

The inspectors evaluated the licensee's neutron dosimetry program, including dosimeter types and/or survey instrumentation.

The inspectors reviewed neutron exposure situations (e.g., independent spent fuel storage installation operations or at-power containment entries) and assessed whether: (a) dosimetry and/or instrumentation was appropriate for the expected neutron spectra; (b) there was sufficient sensitivity for low dose and/or dose rate measurement; and (c) neutron dosimetry was properly calibrated. The inspectors also assessed whether interference by gamma radiation had been accounted for in the calibration and whether time and motion evaluations were representative of actual neutron exposure events, as applicable.

b. Findings

No findings were identified.

(1) Assigning Dose of Record

a. Inspection Scope

For the special dosimetric situations reviewed in this section, the inspectors assessed how the licensee assigned dose of record for total effective dose equivalent, shallow dose equivalent, and lens dose equivalent. This included an assessment of external and internal monitoring results, supplementary information on individual exposures (e.g., radiation incident investigation reports and skin contamination reports), and radiation surveys and/or air monitoring results when dosimetry was based on these techniques.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.05)

a. Inspection Scope

The inspectors assessed whether problems associated with occupational dose assessment were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee involving occupational dose assessment.

b. Findings

No findings were identified.

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

This inspection constituted one complete sample as defined in IP 71124.06-05.

.1 Inspection Planning and Program Reviews (02.01)

(1) Event Report and Effluent Report Reviews

a. Inspection Scope

The inspectors reviewed the radiological effluent release reports issued since the last inspection to determine if the reports were submitted as required by the Offsite Dose Calculation Manual and/or TSs. The inspectors reviewed anomalous results, unexpected trends, or abnormal releases identified by the licensee for further inspection to determine if they were evaluated, were entered in the CAP, and were adequately resolved.

The inspectors identified radioactive effluent monitor operability issues reported by the licensee as provided in effluent release reports, to review these issues during the onsite inspection, as warranted, given their relative significance and determine if the issues were entered into the CAP and adequately resolved.

b. Findings

No findings were identified.

(1) Offsite Dose Calculation Manual and Updated Final Safety Analysis Report Review

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) descriptions of the radioactive effluent monitoring systems, treatment systems, and effluent flow paths for evaluation during inspection walkdowns.

The inspectors reviewed changes to the Offsite Dose Calculation Manual made since the last inspection against the guidance in NUREG-1301, 1302 and 0133, and Regulatory Guides 1.109, 1.21 and 4.1. When differences were identified, the inspectors reviewed the technical basis or evaluations of the change during the onsite

inspection to determine whether they were technically justified and maintain effluent releases ALARA.

The inspectors reviewed licensee documentation to determine if the licensee had identified any non-radioactive systems that had become contaminated as disclosed either through an event report or the Offsite Dose Calculation Manual since the last inspection. This review provided an intelligent sample list for the onsite inspection of any 10 CFR 50.59 evaluations and allowed a determination if any newly contaminated systems had an unmonitored effluent discharge path to the environment, whether any required Offsite Dose Calculation Manual revisions were made to incorporate these new pathways and whether the associated effluents were reported in accordance with Regulatory Guide 1.21.

b. Findings

No findings were identified.

(1) Groundwater Protection Initiative Program

a. Inspection Scope

The inspectors reviewed reported groundwater monitoring results and changes to the licensee's written program for identifying and controlling contaminated spills and/or leaks to groundwater.

b. Findings

No findings were identified.

(1) Procedures, Special Reports, and Other Documents

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (LERs), event reports, and/or special reports related to the effluent program issued since the previous inspection to identify any additional focus areas for the inspection based on the scope and breadth of problems described in these reports.

The inspectors reviewed effluent program implementing procedures, particularly those associated with effluent sampling, effluent monitor setpoint determinations, and dose calculations.

The inspectors reviewed copies of licensee and third party (independent) evaluation reports of the effluent monitoring program since the last inspection to gather insights into the licensee's program and aid in selecting areas for inspection review (smart sampling).

b. Findings

No findings were identified.

.2 Walkdowns and Observations (02.02)

a. Inspection Scope

The inspectors performed walkdowns on selected components of the gaseous and liquid discharge systems to evaluate whether equipment configuration and flow paths were consistent with the documents reviewed. The inspectors also assessed equipment material condition associated with the gaseous and liquid discharge systems. Special attention was made to identify potential unmonitored release points (such as open roof vents in boiling water reactor turbine decks, temporary structures butted against turbine, auxiliary or containment buildings); building alterations which could impact airborne, or liquid, effluent controls; and ventilation system leakage that communicated directly with the environment.

For equipment or areas associated with the systems selected for review that were not readily accessible due to radiological conditions, the inspectors reviewed the licensee's material condition surveillance records, as applicable.

The inspectors walked down filtered ventilation systems to verify that there were no conditions, such as degraded high-efficiency particulate air and/or charcoal banks, improper alignment, or system installation issues, that would impact the performance, or the effluent monitoring capability, of the effluent system.

As available, the inspectors observed selected portions of the routine processing and discharge of radioactive gaseous effluents (including sample collection and analysis) to verify that appropriate treatment equipment was used and the processing activities aligned with discharge permits.

The inspectors determined if the licensee had made significant changes to their effluent release points (e.g. changes subject to a 10 CFR 50.59 review or require NRC approval of alternate discharge points).

As available, the inspectors observed selected portions of the routine processing and discharge of liquid waste (including sample collection and analysis) to verify that appropriate effluent treatment equipment was being used and that radioactive liquid waste was being processed and discharged in accordance with procedure requirements and discharge permits.

b. Findings

No findings were identified.

.3 Sampling and Analyses (02.03)

a. Inspection Scope

The inspectors selected effluent sampling activities, consistent with smart sampling, and assessed whether adequate controls had been implemented to ensure representative samples were obtained (e.g., provisions for sample line flushing, vessel recirculation, composite samplers, etc.)

The inspectors selected effluent discharges made with inoperable (declared out-of-service) effluent radiation monitors to verify that controls were in place to ensure compensatory sampling was performed consistent with the radiological effluent TSs and Offsite Dose Calculation Manual and that those controls were adequate to prevent the release of unmonitored liquid and gaseous effluents.

The inspectors determined whether the facility was routinely relying on the use of compensatory sampling in lieu of adequate system maintenance, based on the frequency of compensatory sampling since the last inspection.

The inspectors reviewed the results of the inter-laboratory comparison program and assessed the quality of the radioactive effluent sample analyses and assessed whether the inter-laboratory comparison program included hard-to-detect isotopes, as appropriate.

b. Findings

No findings were identified.

.4 Instrumentation and Equipment (02.04)

(1) Effluent Flow Measuring Instruments

a. Inspection Scope

The inspectors reviewed the methodology the licensee used to determine the effluent stack and vent flow rates to verify that the flow rates were consistent with radiological effluent T Ss and Offsite Dose Calculation Manual or UFSAR values, and that differences between assumed and actual stack and vent flow rates did not affect the results of the projected public doses.

b. Findings

No findings were identified.

(1) Air Cleaning Systems

a. Inspection Scope

The inspectors assessed whether surveillance test results since the previous inspection for TS required ventilation effluent discharge systems (high-efficiency particulate air and charcoal filtration), such as the standby gas treatment system and the containment and auxiliary building ventilation system, met TS acceptance criteria.

b. Findings

No findings were identified.

.5 Dose Calculations (02.05)

a. Inspection Scope

The inspectors reviewed all significant changes in reported dose values compared to the previous radiological effluent release report (e.g., a factor of 5, or increases that approach Appendix I Criteria) to evaluate the factors which may have resulted in the change.

The inspectors reviewed radioactive liquid and gaseous waste discharge permits and verified that the projected doses to members of the public were accurate and based on representative samples of the discharge path.

The inspectors evaluated the methods used to determine the isotopes that were included in the source term to ensure all applicable radionuclides were included, within detectability standards. The review included the current Part 61 analyses to ensure hard-to-detect radionuclides were included in the source term.

The inspectors reviewed changes in the licensee's offsite dose calculations since the last inspection to verify the changes were consistent with the Offsite Dose Calculation Manual and Regulatory Guide 1.109. The inspectors reviewed meteorological dispersion and deposition factors used in the Offsite Dose Calculation Manual and effluent dose calculations to ensure appropriate factors were being used for public dose calculations.

The inspectors reviewed the latest Land Use Census to verify that changes (e.g., significant increases or decreases to population in the plant environs, changes in critical exposure pathways, the location of nearest member of the public, or critical receptor, etc.) had been factored into the dose calculations.

For the releases reviewed above, the inspectors evaluated whether the calculated doses (monthly, quarterly, and annual dose) were within the 10 CFR Part 50, Appendix I and TS dose criteria.

The inspectors reviewed, as available, records of any abnormal gaseous or liquid tank discharges (e.g., discharges resulting from misaligned valves, valve leak-by, etc.) to ensure the abnormal discharge was monitored by the discharge point effluent monitor. Discharges made with inoperable effluent radiation monitors, or unmonitored leakages were reviewed to ensure that an evaluation was made of the discharge to satisfy 10 CFR 20.1501 so as to account for the source term and projected doses to the public.

b. Findings

No findings were identified.

.6 Groundwater Protection Initiative Implementation (02.06)

a. Inspection Scope

The inspectors reviewed monitoring results of the Groundwater Protection Initiative to determine if the licensee had implemented the program as intended, and to identify any

anomalous results. For anomalous results or missed samples, the inspectors assessed whether the licensee had identified and addressed deficiencies through its CAP.

The inspectors reviewed identified leakage or spill events and entries made into 10 CFR 50.75(g) records. The inspectors reviewed evaluations of leaks and spills, and reviewed any remediation actions taken. The inspectors reviewed onsite contamination events involving contamination of ground water and assessed whether the source of the leak or spill was identified and mitigated.

For unmonitored spills, leaks, or unexpected liquid or gaseous discharges, the inspectors assessed whether an evaluation was performed to determine the type and amount of radioactive material that was discharged by:

- Assessing whether sufficient radiological surveys were performed to evaluate the extent of the contamination and the radiological source term and assessing whether a survey and evaluation had been performed to include consideration of hard-to-detect radionuclides.
- Determining whether the licensee completed offsite notifications, as provided in its Groundwater Protection Initiative implementing procedures.

The inspectors reviewed the evaluation of discharges from onsite surface water bodies that contain or potentially contain radioactivity, and the potential for groundwater leakage from these onsite surface water bodies. The inspectors assessed whether the licensee was properly accounting for discharges from these surface water bodies as part of their effluent release reports.

The inspectors assessed whether onsite ground water sample results and a description of any significant onsite leaks or spills into groundwater for each calendar year were documented in the Annual Radiological Environmental Operating Report for the radiological environmental monitoring program or the Annual Radiological Effluent Release Report for the radiological effluent TSs.

For significant, new effluent discharge points (such as significant or continuing leakage to ground water that continues to impact the environment if not remediated), the inspectors evaluated whether the offsite dose calculation manual was updated to include the new release point(s).

b. Findings

No findings were identified.

.7 Problem Identification and Resolution (02.07)

a. Inspection Scope

The inspectors assessed whether problems associated with the effluent monitoring and control program were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee CAP. In addition, they inspectors evaluated the appropriateness of the corrective actions for a selected sample of problems documented by the licensee involving radiation monitoring and exposure controls.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System Specific Activity Performance Indicator (PI) for Braidwood Station for the period from the first quarter of 2010 through the fourth quarter of 2010. The inspectors used PI definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee’s reactor coolant system chemistry samples, TS requirements, issue reports, event reports, and NRC Integrated Inspection Reports for the period from the first quarter of 2010 through the fourth quarter of 2010 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s IR database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment.

This inspection constituted two reactor coolant system specific activity samples as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/Offsite Dose Calculation Manual Radiological Effluent Occurrences PI for the period from the first quarter of 2010 through the first quarter of 2011. The inspectors used PI definitions and guidance contained in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee’s issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite

dose calculations for selected dates between the first quarter of 2010 through the first quarter of 2011, to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment.

This inspection constituted one radiological effluent TS/Offsite Dose Calculation Manual radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Physical Protection

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for followup, the inspectors performed a daily screening of

items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6 month period of January 2011 through June 2011, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit and/or surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semiannual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors evaluated the licensee's implementation of their process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds (OWAs) on system availability; and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment were reviewed to accomplish the objectives of the inspection

procedure. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into their CAP, and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an initiating event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of mitigating systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified OWAs.

This review constituted one operator workaround annual inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.5 Selected Issue for Followup Inspection: Corrective Actions Related to NRC Identified Issues Related to the Control of Station Temporarily Constructed Scaffolds

a. Scope

The inspectors reviewed the corrective actions associated with NRC NCV 05000456/2010004-01; 05000457/2010004-01, “Failure to Follow Procedure for Temporary Scaffolds.” This Green finding included an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” The inspectors had identified numerous examples in which temporary constructed scaffolds had remained in the plant for over 90 days and a procedurally required qualification action had not been performed. The inspectors concluded that the corrective actions associated with the population of scaffolds identified were not consistent with the scaffold installation, modification, and removal request process. Specifically, this program required that temporary scaffolds that remained in place for greater than 90 days be made permanent or that a specific 10 CFR 50.59 review be conducted for the individual scaffold.

This review was specifically conducted based upon the discovery that the licensee had not adequately addressed a programmatic aspect of the finding based on discovery of numerous temporary scaffolds that remained in place for greater than the allowable 90-day limit without the actions prescribed by the procedure.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05.

b. Findings

Inadequate Quality Review of Temporary Constructed Scaffolds Installed Throughout the Plant

Introduction: The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," related to an inadequate quality review of temporarily constructed scaffolds installed throughout the plant. Specifically, the licensee failed to follow procedural requirements for installed temporary scaffolds prior to reaching 90 days in service. The procedural action required that the temporary scaffold be converted to a permanent scaffold or that a specific 10 CFR Part 50.59 evaluation be performed for the specific scaffold to ensure that the temporary scaffold did not adversely affect structures, system and components (SSCs) before reaching 90 days in service.

Description: On December 15, 2010, a temporary scaffold was constructed in the Unit 1 diesel-driven AF pump room to support a quarterly surveillance on an emergency lighting unit. On April 1, 2011, the inspectors identified and subsequently notified the licensee that although the scaffold had been in place for greater than 90 days, actions to either convert the temporary scaffold to a permanent scaffold, or perform a 10 CFR Part 50.59 evaluation for the specific scaffold to ensure that SSCs had not been affected was not performed, as required.

The scaffold was removed on April 14, 2011. Subsequently, the licensee identified several other temporary scaffolds that had remained in place for greater than 90 days without further evaluation, and entered these deficiencies into the CAP as IRs 1200851, 1201358, and 1201890. Also, the licensee documented in IR 1206426 the untimely response related to the disposition of temporary scaffolds.

The licensee had numerous opportunities to identify that the scaffold was required to be evaluated prior to the expiration of the 90 day time limit. On March 3, 2011, Nuclear Oversight identified an adverse trend with the implementation of the scaffold control program in IR 1183271 and on March 29, 2011, the scaffold coordinator identified 30 scaffolds that would exceed 90 days in the following month. The scaffold identified by the inspectors was not on the list identified in IR 1193841 (which was an action assignment from the monthly check documented in IR 1190300.) The inspectors reviewed the IRs and noted that although these scaffolds had been entered into the CAP, the corrective action assignments for these IRs did not align with the required actions established in the applicable station procedure.

Specifically, Step 2.14 of quality procedure MA-AA-716-025, "Scaffold Installation, Modification, and Removal Request Process," Revision 8, defined a temporary scaffold as follows:

- "Non-Permanent Scaffold – Temporary access or support structures utilizing scaffold material erected in support of Maintenance or Operations activities that are to be removed at the completion of the activities. These temporary access structures are not intended to be left in place for more than ninety days of at power plant operations."

Additionally, Step 3.6 of the same procedure required the following:

- Scaffold Coordinator/Designee – Is responsible for the coordination of erection and removal of all scaffolds on site. Maintaining a log or electronic equivalent of the status of all scaffolds, and reviewing the log to ensure that any Scaffolds approaching their 90 day limit are removed or converted to a Permanent Scaffold or requesting that an individual 10 CFR 50.59 Review be performed for the individual Scaffolds required to be left in place beyond the 90 days."

The inspectors previously documented a similar issue involving a different temporary scaffold as NCV 05000456/201004-01, "Failure to Follow Procedure for Temporary Scaffolds." At that time, the licensee's view was that there was no specific requirement that temporary scaffolds be disassembled prior to exceeding an inservice life of 90 days. A generic 10 CFR 50.59 evaluation had previously been completed that was applicable to all temporary scaffolds. However, IR 1122116, originated by the Exelon corporate office on September 5, 2010, documented that this interpretation was not correct and that in accordance with MA-AA-716-025, temporary scaffolds required a specific analysis in order to remain in place beyond 90 days.

The licensee assigned three corrective actions in IR 1095900 to address NCV 05000456/2010004-01. The first action was to "Complete the procedure review and meet with the NRC and Illinois Emergency Management Agency (IEMA) to further discuss the procedure and the requirements for additional 10 CFR 50.59 reviews and document the conclusions of the meetings." That action was closed with the statement, "Based on the latest discussions, scaffolds that exceed the 90-day duration will either require a specific 10 CFR 50.59 review or will be processed in order to accept the scaffold as permanent. The 10 CFR 50.59 evaluation performed for the scaffold procedure will not be used as the 10 CFR 50.59 review for scaffolds that exceed the 90-day duration." The action closure statement did not indicate that this action was discussed with the NRC or with IEMA, and would not have been acceptable because the actions must be completed prior to exceeding the 90-day limit, not after exceeding it.

The second action was to "Evaluate the scaffold procedure MA-AA-716-025 for required revision to clarify the need/expectations for 10 CFR 50.59 of non-permanent scaffolds installed beyond 90 days. Provide proposed wording to maintenance and generate additional followup actions to track any recommended change." This action was open at the end of the inspection period with a due date of June 29, 2011.

The third action was to "Review scaffold procedure and 10 CFR 50.59 for inspections requirements by the Fire Marshall. Revised documents as needed to ensure alignment between the procedure and 10 CFR 50.59." This new assignment was from ATI 1122116-02, which was an assignment from Exelon corporate office that had been closed to this task. This action was open at the end of the inspection period with a due date of June 22, 2011.

The inspectors concluded that, despite the completed action assignment to perform 10 CFR 50.59 reviews for scaffolds at the 90-day limit, the station continued to fail to adhere to Step 3.6 of procedure MA-AA-716-025, and also continued to not meet the intent of Step 2.14. The inspectors determined that this was not an administrative issue based on a detailed review of the procedures and discussions with the licensee staff. This finding was entered into the licensee's CAP as IR 1206426. Corrective actions

included the performance of an extent of condition review and the restoration to compliance of all known deficiencies.

Analysis: The inspectors determined that the failure of licensee personnel to adhere to quality procedure, MA-AA-716-025 as it related to the control of temporary scaffolding was a performance deficiency.

The issue was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the issue, if left uncorrected, would have the potential to become a more significant safety concern. Specifically, by not taking the actions prescribed in the scaffold control procedures, the temporary structures would not have an adequate qualification if left in the plant for greater than 90 days and may not meet all standards of the station's licensing basis. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 4, "Phase I - Initial Screening and Characterization of findings," Table 4a for the Mitigating Systems Cornerstone and answered "No" to the Mitigating Systems Cornerstone questions. Specifically, the issue did not result in the actual loss of the operability or functionality of a safety system. Therefore, the finding screened as having very low safety significance (Green).

The inspectors determined that this finding had a cross-cutting aspect in the CAP component of the Problem Identification and Resolution cross-cutting area (P.1(d)) because the licensee did not take appropriate correct actions to address safety issues and adverse trends in a timely manner, commensurate with their safety significance and complexity. Specifically, the licensee did not take appropriate corrective actions to address a very similar issue identified as NRC inspection finding 05000456/2010004-01; 05000457/2010004-01, "Failure to Follow Procedures for Temporary Scaffolds."

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be prescribed by instructions, procedures, or drawings of a type appropriate to the circumstance and shall be accomplished in accordance with these instructions, procedures, or drawings. Step 3.6 of quality procedure MA-AA-716-025, "Scaffold Installation, Modification, and Removal Request Process," Revision 7B, required that temporarily constructed scaffolds be removed or converted to a permanent scaffold or an individual 10 CFR 50.59 review be performed for the individual scaffold required to be left in place beyond 90 days.

Contrary to the above, from October 29, 2008, to March 12, 2011, the licensee failed to adhere to Step 3.6 of quality procedure, MA-AA-716-025, in a number of instances. In each instance, the temporarily constructed scaffold was left in place for greater than 90 days without conversion to a permanent scaffold or an individual 10 CFR 50.59 review being performed. Corrective actions included the performance of an extent of condition review and the restoration to compliance of all known deficiencies.

Because this violation was of very low safety significance and was entered into the licensee's CAP as IR 1206426, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000456/2011003-08; 05000457/2011003-08, Inadequate Quality Review of Temporary Constructed Scaffolds Installed Throughout the Plant)**

4OA3 Followup of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report 05000456/2010-004-00, “Reactor Trip Due to Performance of a Channel Calibration With a Coincident Bistable in Half-Trip Condition”

a. Scope

On September 20, 2010, Braidwood Unit 1 began a TS surveillance calibration for a steam generator water level channel. Upon placing a bistable for the channel in the tripped condition consistent with the procedural instructions, an automatic turbine trip occurred. The turbine trip was promptly followed by a reactor trip.

The licensee investigated the cause and determined that a universal logic card in the solid state protection system had failed and resulted in an unknown half trip condition. During the performance of the channel calibration when the coincident loop bistable was tripped, the 2-of-4 logic was met, which caused an immediate turbine trip.

The licensee reported this Licensee Event Report (LER) pursuant to 10 CFR 50.73(a)(2)(iv)(A) due to the actuation of the reactor protection system.

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations. Additionally, the inspectors reviewed the root cause report associated with this issue and reviewed the response of control room operators in response to the reactor trip. No findings were identified and no violation of NRC requirements occurred. Documents reviewed are listed in the Attachment. This LER is closed.

This event followup review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.2 (Closed) Licensee Event Report 05000456/2010-005-00; 05000457/2010-005-00, “Incorrect Methodology Used in Calculations in 1999 Resulted in Non-Conservative Control Room Outside Air Intake Monitor Alarm Setpoints”

a. Scope

On November 12, 2010, Braidwood Station submitted an LER identifying incorrect alarm/ actuation setpoints for the control room noble gas channels (0PR31B, 0PR32B, 0PR33B and 0PR34B). The licensee discovered the error during an extent of condition investigation of the licensee’s calculation methodology for alarm/actuation setpoints. Consequently, the licensee declared the control room ventilation system inoperable, entered a limiting condition for operation (LCO), and placed the control room ventilation filtration system in the emergency mode until the setpoints were revised for the outside air intake noble gas channels. The LCO was exited 9 hours later.

The licensee determined that the incorrect alarm setpoints were an oversight during the noble gas channel detector replacement in December 1999. The licensee found that the old noble gas setpoints were set approximately 45 percent higher (non-conservative) than the recalculated alarm setpoints. Specifically, the setpoints used to initiate the

automatic control room emergency ventilation system were entered as 2.9 millirem/hr in December 1999 as compared to the revised setpoint of 2.0 millirem/hr. The licensee calculated that the radiation exposure consequence from noble gas to the control room from the higher trip setpoints was less than 0.5 millirem whole body dose to the control room personnel. The licensee concluded that the setpoint preparer did not have sufficiently detailed instructions and individual judgment was applied to determine the calculation methodology. In addition, the calculation reviewer also did not have guidance for the correct methodology.

The inspectors reviewed this LER and the associated requirements. The licensee revised the applicable radiation monitor procedures that perform setpoint calculations for the process monitors including instructions for the basis of the review criteria. The enforcement aspects of this licensee-identified violation are discussed in Section 4OA7 of this report. This LER is closed.

This event followup review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.3 (Closed) Licensee Event Report 05000456/2010-007-00; 05000457/2010-007-00, "Potential Loss of Residual Heat Removal System Safety Function in Mode 4 When Aligned for Shutdown Cooling Due to Potential for Flashing or Voiding of Coolant During a Shutdown Loss-of-Coolant Accident"

a. Scope

NRC inspection report 05000456/457/2010005 documented a performance deficiency and an associated NCV related to the licensee's failure to report conditions which could have prevented the fulfillment of the RHR emergency core cooling function in accordance with 10 CFR 50.72(a)(2)(v) (NCV 05000456/2010005-02; 05000457/2010005-02, "Failure to Submit Licensee Event Report per 10 CFR 50.73(a)(2)(v)"). Specifically, the licensee identified a number of historical instances that represented a loss of RHR function when both RHR trains were utilized for normal shutdown cooling above 200 ºF during a past operability review. This operability review, however, was not reviewed against the 10 CFR 50.73 reporting requirements.

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations with the exception of the previously documented NCV. Documents reviewed are listed in the Attachment. No additional findings were identified and no additional violations of NRC requirements occurred. This LER is closed.

This event followup review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.4 (Closed) Licensee Event Report 05000456/2011-002-00, "Loss of Unit 1 Train A Bus 141 Degraded Undervoltage Function"

a. Scope

On October 11, 2010, during the performance of undervoltage relay surveillance, the Unit 1 Train A degraded undervoltage relay for a 4.16 kilovolt (kV) engineered safety feature (ESF) Bus 141 was found out of tolerance. The relay was replaced following the surveillance.

The purpose of undervoltage protection was to detect when a loss of voltage or a degraded voltage condition occurs. Two relays were provided in each 4.16 kV bus for detecting an undervoltage condition and two additional relays were provided for detecting a loss of bus voltage. Each pair of relays was used in a two-out-of-two logic scheme to generate a Loss of Power signal to start the diesel generators for an undervoltage or loss of voltage condition. Due to the undervoltage relay being out of tolerance, the degraded voltage protection was unavailable. However, loss of voltage protection remained available. In addition, Train B had both functions available, therefore the system was capable of performing its safety function.

The licensee performed an evaluation to determine the cause of the out-of-tolerance condition. This evaluation identified that the undervoltage relay had a manufacturing defect that a polarity sensitive capacitor was installed with the polarity reversed. The effect of the capacitor being installed with the polarity reversed was an increase in leakage current that could cause a dip in voltage and could lead to a drift in calibration over time.

Based on a review of past trends and surveillance history, the licensee determined that there was not enough evidence to determine the specific time when the relay became out-of-tolerance and the relay would have failed to sense an undervoltage condition. However, the licensee stated that even though there was insufficient evidence to determine when the relay became out-of-tolerance, it might have existed for longer that the period allowed by TS LCO 3.3.5, "Loss of Power Diesel Generator Start Instrumentation," Condition A, which required the channel to be placed in a tripped condition within 1 hour of a channel becoming inoperable.

The licensee reported this condition under 10 CFR 50.73(a)(2)(i)(B), "Any condition that is prohibited by the plant's TS." The inspectors reviewed this LER and determined that it was conservatively completed in accordance with NRC regulations. Documents reviewed are listed in the Attachment. No findings were identified and no violation of NRC requirements occurred. This LER is closed.

This event followup review constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

.5 (Closed) Unresolved Item 05000456/2010004-02; 05000457/2010004-02, Temporary Scaffold Quality Control Process

a. Scope

Unresolved Item 05000456/2010004-02; 05000457/2010004-02 was opened in NRC Inspection Report 05000456/2010004; 05000457/2010004 to determine whether the licensee's procedures for constructing temporary scaffolding provided an adequate level of quality to ensure the station's licensing basis was maintained. This issue arose based on the licensee's position that temporary scaffolds could be installed in the plant greater than 90 days without the performance of a specific 10 CFR 50.59 evaluation or an action to make the scaffold permanent in accordance with standards involved in that process.

During this inspection period, the licensee revised this position and revised associated processes and procedures to prescribe additional reviews if a temporary scaffold was intended to be left in place for greater than 90 days.

b. Findings

Non-Cited Violation 05000456/2010004-01; 05000457/2010004-01, "Failure to Follow Procedure for Temporary Scaffolds" and NCV 05000456/2011003-08; 05000457/2011003-08, "Inadequate Quality Review of Temporary Constructed Scaffolds Installed throughout the Plant" documented the inspectors' review of the URI and the associated regulatory aspects. This URI is closed.

.6 Personnel Performance during the June 8, 2011, Auxiliary Building Flooding Event and Notice of Unusual Event Declaration

a. Inspection Scope

On June 8, 2011, a temporarily connected 4-inch hose failed during the performance of an 18-month SX supply to AF system flushing activity. The hose split in a manner that caused water to be sprayed upon the 1A and 2A AF pumps, motors, and electrical junction boxes, the unit common CC water pump electrical bus, and safety-related motor control center (MCC) 132X1. The licensee declared a Notice of Unusual Event (NOUE) based on flooding that had the potential to result in inoperable safety-related equipment.

The inspectors evaluated operator performance for this unplanned occurrence to determine if the response was adequate and in accordance with station procedures. The inspectors responded to both the control room and conducted independent plant walkdowns to ensure that the licensee had adequately isolated the leak, performed adequate system walkdowns of affected equipment, and to verify that the diesel driven AF pumps were operable since they were being relied upon to satisfy a safety function. Additionally, the inspectors verified that the station NOUE declaration was appropriate and timely.

b. Findings

No findings were identified related to licensee's performance in response to the event. However, a self-revealed finding was identified related to the cause of the event.

Incorrect Equipment Used During an Auxiliary Feedwater Suction Piping Flush Surveillance

Introduction: A finding of very low safety significance (Green) and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when incorrect equipment was used during an AF suction piping flush surveillance. Specifically, the use of an incorrect and unqualified drain hose resulted in the hose rupturing and spraying water onto nearby safety-related equipment, rendering the equipment inoperable until equipment tests could be performed.

Description: On June 8, 2011, the licensee performed a scheduled 18-month surveillance to flush the essential service water supply piping to the 2A AF pump in accordance with 2BwOS SX-1, "Unit Two Auxiliary Feedwater Pump Essential Service Water Suction Line Flush 18 Month Surveillance." This procedure specified that a 4-inch hose rated for 150 pounds per square inch gauge (psig) be connected to flush valve 2SX253. At 10:04 a.m., the licensee placed the 2A AF pump in pull-to-lock to perform the suction pipe flush. At 10:11 a.m., the main control room received a "Post Loss of Cooling Accident H2 Monitor Trouble" alarm and shortly thereafter received a field report that after the 2SX253 valve was opened to commence the flush, the drain hose ruptured and sprayed an estimated 300 gallons of water in the area. The rupture was isolated locally within approximately 45 seconds. Initial field reports indicated that the following equipment was wetted or otherwise impacted:

- 1A AF Pump Electrical Junction Box;
- 2A AF Pump Electrical Junction Box;
- Cable Trays on 364' Elevation;
- Unit 0 Component Cooling Water Pump Bus; and
- MCC 132X1.

The 2A AF pump was already in pull-to-lock due to the flush surveillance. The licensee placed the control switch for the 1A AF pump and all Unit "0" CC equipment switches in pull-to-lock due to concerns about the equipment impact of being wetted. As a result, online risk for Unit 1 and Unit 2 was revised from Yellow to Orange. The licensee entered abnormal procedure 0BwOA PRI-8, "Auxiliary Building Flooding Unit 0," and declared an NOUE at 10:26 a.m., in accordance with Emergency Action Level HU5 due to flooding with the potential to impact safety-related equipment. The licensee terminated the NOUE at 11:22 a.m., after the leak was isolated.

Following the event, the licensee evaluated the condition of potentially affected equipment. The licensee noted some water intrusion into MCC 132X1. However, there was no impact on MCC operation. The licensee also found mud and water inside the Unit 0 component cooling bus and determined that it should not be used until cleaning and additional functionality checks were performed. The licensee performed an electrical check (megger test) of the 1A and 2A AF pump motors and found no issues from the water spray.

The licensee's preliminary investigation into the event revealed that an incorrect hose was used for the flushing evolution. Surveillance procedure 2BwOS SX-1 included a prerequisite statement to use a 4-inch diameter hose rated for 150 psig. In addition, during the pre-job brief the Operations field supervisor advised the equipment operators performing the surveillance to expect that a black hose would already be installed.

However, the work package being used by the individuals installing the temporary hose did not include guidance on hose color or pressure rating. The work instructions included a step to install the hose provided by Operations; however, in this case, the licensee personnel installing the hose were directed to obtain the hose from the hose locker. The hose locker contained black cam-lock hoses rated for 150 psig and green cam-lock hoses rated for 65 psig. When the personnel installing the hose for the surveillance went to the locker to retrieve the hose, they noted that the cam-lock flange fit the green hose and, therefore, selected the (incorrect) green hose rather than the (correct) black hose.

Analysis: The inspectors determined that the use of an improper hose during the 2A auxiliary feedwater suction line flushing evolution, which ruptured and wetted safety-related equipment, was a performance deficiency.

The issue was determined to be more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because it was associated with the Equipment Performance attribute of the Mitigating Systems Cornerstone and adversely 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). Specifically, the availability and reliability of safety-related equipment was potentially adversely affected due to being sprayed with water.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, Attachment 4, “Phase 1 – Initial Screening and Characterization of Findings.” Using Table 2 of IMC 0609, Attachment 4, the inspectors determined that the finding affected the secondary short-term decay heat removal function of the Mitigating Systems Cornerstone. The inspectors answered “No” to all Mitigating Systems Cornerstone questions in Table 4a, “Characterization Worksheet for Initiating Events, Mitigating System, and Barrier Integrity Cornerstone,” and, as a result, the finding was determined to be of very low safety significance (Green).

The inspectors determined that the finding had a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area (H.4(a)) because when faced with the choice between two different hoses for a flushing activity, workers proceeded with the evolution in the face of uncertainty.

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, the work package directing installation of a drain hose prior to flushing the 2A AF pump essential service water suction piping on June 8, 2011, did not provide clear guidance on the appropriate hose to install. As a result, the installed hose was not rated to a sufficient pressure and ruptured during the flushing activity, rendering several pieces of safety-related equipment inoperable due to being wetted. Corrective actions included reinforcing the use of the correct hose for the surveillance and the need to stop and involve supervision when faced with uncertain conditions. The licensee has also initiated a Root Cause Evaluation to identify additional corrective actions. Because this violation was of very low safety significance and because this issue was entered into the licensee’s CAP as IR 1226235, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy.

(NCV 0500456/2011003-09; 05000457/2011003-09, Incorrect Equipment Used During an Auxiliary Feedwater Suction Piping Flush Surveillance)

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in Severe Accident Management Guidelines (SAMGs), and as required by 10 CFR 50.54(hh); (2) an assessment of the licensee's capability to mitigate station blackout conditions, as required by 10 CFR 50.63 and station licensing bases; (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases; and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flooding events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

Inspection Report 05000456/2011011; 05000457/2011011 (ML111320261), dated May 13, 2011, documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues. This Temporary Instruction (TI) is closed.

.2 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines"

On May 19, 2011, the inspectors completed a review of the licensee's Severe Accident Management Guidelines (SAMGs), implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Dai Ichi fuel damage event in Japan. Plant-specific results for the Braidwood Station were provided as an Enclosure to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated June 1, 2011 (ML111520396). This TI is closed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 6, 2011, the inspectors presented the inspection results to Mr. M. Kanavos and other members of the licensee staff. The licensee acknowledged the information

presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- In-Plant Airborne Radioactivity Control and Mitigation, Performance Indicator Verification, Occupational Dose Assessment, and close-out of LER 05000456/2010-005-00 with Mr. M. Kanavos on April 1, 2011.
- Inservice Inspection with Mr. D. Enright on May 4, 2011.
- Completion of TI-183, "Followup to the Fukushima Dai Ichi Nuclear Station Fuel Damage Event," on April 21, 2011.
- Completion of TI-184, "Availability and Readiness Inspection of SAMGs," on May 19, 2011.
- Radiological Hazard and Exposure Control, and ALARA Planning and Control, with Mr. M. Kanavos on May 27, 2011.
- Radioactive Gaseous and Liquid Effluent Treatment with Mr. M. Kanavos on May 27, 2011.

4OA7 Licensee-Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee, and are violations of NRC requirements, which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being disposition as NCVs.

Cornerstone: Barrier Integrity, Occupational Radiation Safety

- License Condition 2.C.(1) states, in part, that the licensee is authorized to operate both units at reactor core power levels not to exceed 3586.6 megawatts thermal. Contrary to this, both units exceeded their license thermal power limits since original construction by approximately 0.5 percent. The licensee identified that the flow coefficient utilized in the reactor power calorimetric calculation was not conservative during a post-maintenance calibration of a new flow instrument. The finding was determined to have very low safety significance because it only involved the potential to affect the fuel barrier. The licensee entered this issue into the CAP as IR 1217236 and implemented the correct flow coefficients.

- While performing extent of condition activities as a result of a prior NRC inspection, the licensee identified a violation of TS 3.3.7, which requires the actuation instrumentation for the control room ventilation system to be operable. Non-conservative actuation setpoints were used for the control room outside air intake noble gas channels, which rendered the monitors inoperable. Specifically, in December 1999, the setpoints for the control room outside air intake noble gas channels (0PR31B, 0PR32B, 0PR33B and 0PR34B) were entered as 2.9 millirem/hr as compared to the correct setpoint of 2.0 millirem/hr. This issue was entered into the licensee's CAP as IR 1106414. This finding was of very low safety significance

(Green) because the finding did not involve: (1) an ALARA finding; (2) an overexposure; (3) a substantial potential for overexposure; or (4) an impaired ability to assess doses.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Enright, Site Vice President
M. Kanavos, Plant Manager
S. Butler, Corrective Action Program Manager
P. Daly, Radiation Protection Manager
A. Ferko, Engineering Director
M. Marchionda-Palmer, Operations Director
J. Moser, Radiation Protection Manager
R. Radulovich, Nuclear Oversight Manager
J. Rappeport, Chemistry Manager
C. VanDenburg, Regulatory Assurance Manager

Nuclear Regulatory Commission

E. Duncan, Chief, Reactor Projects Branch 3
B. Dickson, PST, Branch Chief, DRS/RIII
A. M. Stone, Branch Chief, DRS/RIII

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2011003-01; 05000457/2011003-01	FIN	Failure to Follow Procedural Standards Related to the Storage of Outside Material that could Impact Offsite Power Availability (Section 1R01.2)
05000456/2011003-02;	NCV	Structural Steel Beam Missing Fire Proofing Materials (Section 1R05.1)
05000457/2011003-03	NCV	Failure to Perform Post VT-3 Examination Illumination Verification in Accordance with ASME Code (Section 1R08.1)
05000456/2011003-04; 05000457/2011003-04	NCV	Failure to Ensure that the Design of the Auxiliary Feedwater Suction Piping was Adequate to Prevent Air Entrainment Following a Seismic or Tornado Event (Section 1R15.1)
05000456/2011003-05; 05000457/2011003-05	URI	Potential Design Control Violation Related to Safety-Related Door Impairment (Section 1R15.2)
05000456/2011003-06; 05000457/2011003-06	URI	Asiatic Clams Identified in the Essential Service Water System Supply to the Auxiliary Feedwater System (Section 1R15.3)
05000456/2011003-07; 05000457/2011003-07	NCV	Failure to Follow Procedure During Reactor Vessel Head Lift (Section 1R19.1)
05000456/2011003-08; 05000457/2011003-08	NCV	Inadequate Quality Review of Temporary Constructed Scaffolds Installed Throughout the Plant (Section 4OA2.5)
05000456/2011003-09; 05000457/2011003-09	NCV	Incorrect Equipment Used During an Auxiliary Feedwater Suction Piping Flush Surveillance (Section 4OA3.6)

Closed

05000456/2011003-01; 05000457/2011003-01	FIN	Failure to Follow Procedural Standards Related to the Storage of Outside Material that could Impact Offsite Power Availability (Section 1R01.2)
05000456/2011003-02;	NCV	Structural Steel Beam Missing Fire Proofing Materials (Section 1R05.1)
05000457/2011003-03	NCV	Failure to Perform Post VT-3 Examination Illumination Verification in Accordance with ASME Code (Section 1R08.1)
05000456/2011003-04; 05000457/2011003-04	NCV	Failure to Ensure that the Design of the Auxiliary Feedwater Suction Piping was Adequate to Prevent Air Entrainment Following a Seismic or Tornado Event (Section 1R15.1)
05000456/2011003-07; 05000457/2011003-07	NCV	Failure to Follow Procedure During Reactor Vessel Head Lift (Section 1R19.1)
05000456/2011003-08; 05000457/2011003-08	NCV	Inadequate Quality Review Of Temporary Constructed Scaffolds Installed Throughout the Plant (Section 4OA2.5)
05000456/2011003-09; 05000457/2011003-09	NCV	Incorrect Equipment Used During an Auxiliary Feedwater Suction Piping Flush Surveillance (Section 4OA3.6)
05000456/2010004-02; 05000457/2010004-02	URI	Temporary Scaffold Quality Control Process (Section 40A3. 5)

05000456/2010-004-00	LER	Reactor Trip Due to Performance of a Channel Calibration with a Coincident Bistable in a Half-Trip Condition (Section 4OA3.1)
05000456/2010-005-00; 05000457/2010-005-00	LER	Incorrect Methodology Used in Calculations in 1999 Resulted in Non-Conservative Control Room Outside Air Intake Monitor Alarm Setpoints (Section 4OA3.2)
05000456/2010-007-00; 05000457/2010-007-00	LER	Potential Loss of Residual Heat Removal System Safety Function in Mode 4 When Aligned for Shutdown Cooling Due to Potential for Flashing or Voiding of Coolant During a Shutdown Loss-of-Coolant-Accident (Section 4OA3.3)
05000456/2011-002-00	LER	Loss of Unit 1 Train A Bus 141 Degraded Undervoltage Function (Section 4OA3.4)

Discussed

05000456/2008003-01; 05000457/2008003-01	NCV	Failure to Implement Material Control Procedures (Section 1R01.2)
05000456/2009003-01	FIN	Failure to Control and Secure material Adjacent to Unit 1 Transformer Yard which could Become Potential Missiles (Section 1R01.2)
05000456/201004-01; 05000457/201004-01	NCV	Failure to Follow Procedure for Temporary Scaffolds (Section 4OA2.5)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- 0BwOA ENV-1; Adverse Weather Condition Unit 0; Revision 110
- 1BwOA ENV-1; Adverse Weather Condition Unit 1; Revision 5
- OP-AA-108-107; Switchyard Control; Revision 2
- OP-AA-108-107-1001; Station Response to Grid Capacity Conditions; Revision 3
- OP-AA-108-107-1002; Interface Procedure Between Comed/PECO and Exelon Generation for Transmission Operations, Revision 4
- 0Bw-OA ELEC-1; Abnormal Grid Conditions; Revision 7
- IR 1215755; NOS ID Severe Weather Action Procedure Not Followed; May 13, 2011
- IR 1215778; Entered 0/1/2 BwOA ENV-1 Due to Tornado Warning; May 13, 2011
- 2011 – Braidwood Certification Letter for Summer Readiness

1R04Q Equipment Alignment

- IR 1232307; Safety – NAOH Stalactites in 2B RH Pump Room; June 23, 2011
- BwOP DC-E5; Electrical Lineup – Unit 2 Operating – 125V DC Division 21; Revision 9
- BwOP FW-M1, Operating Mechanical Lineup; Revision 13
- BWOP RH-M3, Operating Mechanical Lineup; Revision 8

1R04S Equipment Alignment

- BwOP FC-E1, Electrical Lineup – Unit 1; Revision 1
- BwOP FC-M1; Operating Mechanical Lineup Unit 1; Revision 8
- BwOP FC-M2; Operating Mechanical Lineup Unit 2; Revision 7
- BwOP FC-1; Fuel Pool Cooling System Startup; Revision 25

1R05 Fire Protection

- IR 1011644; 1FP5100G Valve Handle Needs to be Replaced; January 3, 2010
- IR 1019831; Weekly Work Package Closure Review – 1 Item Identified; September 10, 2009
- IR 1021793; A Better Way to do Business was Discovered; January 26, 2010
- IR 1023358; The DDFP Prerun Cooling Water Temp is Lower than 120 Degree F; January 29, 2010
- IR 1023497; 2FP507 Will Not Close Preventing System Reset; January 29, 2010
- IR 1030462; AR Tag Dated February 11, 1999 Still Hanging in Field (Still Valid); February 15, 2010
- IR 1033210; Operations Identified Issues; February 20, 2010
- IR 1035746; Operations – DDFP Draw Down Volts Low; February 25, 2010
- IR 1048525; Revise BwAP 1100-16 to Add Caution for Water in UCSR; March 26, 2010
- IR 1049419; Procedure Enhancements Needed for 0BwOS FP.e.e.E-12; March 29, 2010
- IR 1052200; FP Valve Handwheel Damaged; April 4, 2010
- IR 1056179; Unit 1 – Status of UCSR Floor Gypsum Fire Seal Caulking; April 13, 2010
- IR 1056181; Unit 2 – Status of UCSR Floor Gypsum Fire Seal Caulking; April 13, 2010

- IR 1058059; 1FP5062 Has a Broken Handwheel; April 18, 2010
- IR 1058338; 0BwOS FP-Q5 Failed Acceptance Criteria; April 19, 2010
- IR 1063981; MRULE Expert Panel Should Consider a New MRULE Function; April 30, 2010
- IR 1066926; UCSR Fire System Surveillance Concern; May 7, 2010
- IR 1069241; 0FP25J Failure; May 14, 2010
- IR 1069759; Diesel Driven Fire Pump had Low Discharge Pressure; May 15, 2010
- IR 1070599; Fire Pump Does Not Have Screen and Missing Metal; May 18, 2010
- IR 1077556; UCSR Halon Inoperable for 5 Weeks; June 6, 2010
- IR 1081770; DDFP Relief Valve Potentially Need Setpoint Adjustment; June 17, 2010
- IR 1083896; Halon Manual Actuators Replacement; June 5, 2010
- IR 1084940; 0B Fire PP GOCAR BwAP 1110-1A2 Extension Beyond 7 Days; June 27, 2010
- IR 1102674; Leak from Diesel Driven Fire Pump (DUP); August 16, 2010
- IR 1104545; 0B Fire Pump BwAP 1110-1A2 Extension Past 7 Days; August 22, 2010
- IR 1117428; FP Diesel Jacket Water Too Hot to Sample; September 24, 2010
- IR 1132822; GCOAR Exceeds 5 Weeks for Zone 2D-17; October 29, 2010
- IR 1134351; 0B Fire Pump F&P Test Not Performed as Scheduled; November 2, 2010
- IR 1137149; Byron NRC FP Triennial Inspection Concern – 4kV Bus Restart; November 8, 2010
- IR 1139099; 0B Fire Pump Failed Surveillance; November 11, 2010
- IR 1150018; 0FP225 Frozen or Clogged – Test Header Will Not Drain; December 8, 2010
- IR 1155513; FP Underground Gate Valve 0FP829 Access Hole Full of Water; December 23, 2010
- IR 1155514; Frozen Mud in the Access Hole for 0FP977; December 23, 2010
- IR 1155516; 0FP817 Cover Frozen, Hole Full of Water and Ice; December 23, 2010
- IR 1155518; 0FP816 Cover Frozen, Hole Found Full of Water and Ice; December 23, 2010
- IR 1155803; 2FP5062 Handwheel Broken; December 25, 2010
- IR 1155804; 2FP028 Handwheel Broken; December 26, 2010
- IR 1166736; OPS ID; Adverse Trend Identified in Fire Protection; January 25, 2011
- IR 1169972; PMT Test Failed; February 2, 2011
- IR 1171547; Missing Handwheel on 1FP5028; February 5, 2011
- IR 1171898; Long Standing Issue on FP Piping; November 1, 2006
- IR 1177582; 1FP5097A Valve Handle is Broken; February 20, 2011
- IR 1179659; Unidentifiable Noise from 1FP09J; February 24, 2011
- IR 1182004; 0B Fire Pump Draw Down Voltage Unsatisfactory During 0BwOS FP2.2.M-2; March 1, 2011
- IR 1184825; 0B Fire Pump Failed Draw Down Surveillance; March 8, 2011
- IR 1188556; 1FP251B Burps Water on Fire Pump Starts; March 17, 2011
- IR 1192662; 0B Fire Pump Has an Internal Coolant Leak; March 26, 2011
- IR 1199209; NER 11-009 Identified Vulnerability CO2, Halon and Foam Systems; April 7, 2011
- IR 1199270; Engine Coolant Leak – Maint Rule Functional Failure; April 7, 2011
- IR 1204920; NER 11-009 Walkdown HS-173 Installation Not Consistent; April 8, 2011
- IR 1208194; CO2 Not Restored Within Required 72 Hours (Zone 2S-43); April 27, 2011
- IR 1208195; CO2 for Electric Cable Tunnel (2S-47) Not Restored in 72 Hrs; April 27, 2011
- IR 1209232; QV Identified Fire Seal Issue; April 28, 2011
- IR 1209808; NRC Question - Should Beam in AB Have Fireproofing On It; April 29, 2011 (IR Generated as a Result of the Inspection)
- IR 1211019; Fire Zone 11.6C-0 to be Included in Surveillance Procedure; May 3, 2011
- IR 1211303; 0FP03PB Request Procurement Engineering Evaluate Heads; May 3, 2011
- IR 1211321; 0FP03PB New Heads Have Flaws and Holes Partial Bored; May 3, 2011
- IR 1212786; Missing Seal in 2 Hour Wall; May 6, 2011

- Braidwood Pre-Fire Plan #90; DG 401’ Diesel Generator Room 1A and Day Tank Room; Revision 0
- Braidwood Pre-Fire Plan #138; AB 383’ Unit 2 Auxiliary Feedwater Pump Diesel Room; Revision 0
- Braidwood Pre-Fire Plan #146; AB 401’ Unit 2 Aux Bldg General Area; ; Fire Zone 11.5-0 Center
- Braidwood Pre-Fire Plan #147; AB 401’ Unit 1 Aux Bldg General Area; Fire Zone 11.5-0 North
- Braidwood Pre-Fire Plan #148; AB 401’ Aux Bldg General Area; Fire Zone 11.5-0 South
- Pre-Fire Plan #165; AB 426’ Aux. Building Laundry Room; Revision 0
- Pre-Fire Plan #178; FH 401’ Fuel Handling Building; Revision 0
- Pre-Fire Plan #179; FH 426’ Fuel Handling Building; Revision 1
- BwMS FP.7.1.E-1; Fire Rated Assemblies Visual Inspection; Revision 4
- BwMS FP.7.1.E-1A0; Fire Rated Assemblies Visual Inspection Accessible Area Data Sheets; Revision 1
- Drawing A-253; Auxiliary Building Mezzanine Floor Plan Area 2; Revision CM
- Drawing S-1302; Auxiliary Building Floor Framing Plan E. 439’-0” Area 2; Revision DJ
- EC 384357; Fire Protection Evaluation for Fire Zone 11.6C-0 and 3.2B-1 Boundaries to Demonstrate Separation Equivalent to BTP CMEB 9.5-1, C5.b(1); Revision 0

<u>1R07 Heat Sink Performance</u>

- IR 1206757; 2CC01A: U2 CC HX West Channel Flange Degraded; April 23, 2011
- WO 00773926-01; MM-Repair of Corroded U2 CC HX Inlet Flange 2CC01A; May 1, 2011
- EC 382983; 2CC01A Repair of Corroded U2 CC HX Inlet flange, Aux. El. 364’ M & 20

<u>1R08 Inservice Inspection Activities</u>

- ER-AA-335-016; VT-3 Visual Examination of Component Supports, Attachments and Interiors of Reactor Vessels; Revision 6
- HU-AA-104-101; Procedure Use and Adherence; Revision 4
- WDI-STD-1041; Reactor Vessel Head Penetration Ultrasonic Examination Analysis; Revision 4
- EXE-PDI-UT-1; Ultrasonic Examination of Ferritic Pipe Welds in Accordance with PDI-UT-1; Revision 5
- EXE-PDI-UT-2; Ultrasonic Examination of Austenitic Pipe Welds in Accordance with PDI-UT-2; Revision 5
- EXE-ISI-8; Visual VT-1 and VT-3 Examination at Exelon; Revision 1
- STD-400-173; Checkout and Operation of the Steam Generator Tube Standard In Situ Pressure Test System; Revision 13
- A2R14 COMA; Braidwood Unit 2 A2R14 Steam Generator Condition Monitoring Operational Assessment Report; Revision 0
- N-533-1; Alternative Requirements for VT-2 Visual Examination of Class 1,2, and 3 Insulated Pressure-Retaining Bolted Connections Section XI, Division 1; February 26, 1999
- AR01208707; A2R15 CRDM Inspection ID’D 3 Penetrations that Require Disposition; April 28, 2011
- IR 1206632; Unit 2 RPV CRDM Pen #23 Special Interest Indications; April 23, 2011
- IR 0979896; NRC Identified 2A RH HX Weld Indication Not Dispositioned; October 15, 2009
- IR 0980924; Less Than Expected UT Exam Results for Augmented Areas in U2; October 18, 2009
- IR 0981582; SX Valve Bolts Rusty (2SX021 and 25 A, B, C, D); October 20, 2009

- IR 0985759; A2R15 Replace Snubber 2FW05011S Due to Marginal Test Results; October 29, 2009
- IR 0978989; U-2 Containment Liner Plate Metal Reduction Exceeding 10 percent; October 14, 2009
- IR 0983713; 2SI121A Boric Acid Accumulation; October 24, 2009
- WPS 8-8-GTSM; GTAW/SMAW P-8 to P-8 Material; Revision 2
- PQR 1-51A; GTAW/SMAW Manual P-8 to P-8; December 28; 1983
- PQR 4-51A; GTAW/SMAW Manual P-8 to P-8; December 28; September 12, 1986
- PQR A-003; GTAW/SMAW Manual P-8 to P-8; February 8, 2000
- PQR A-004; GTAW/SMAW Manual P-8 to P-8; February 8, 2000
- WO 1033803-01; Modify RWST Level Transmitter Drain Line 2SI99G-1" per EC#365548; March 26, 2009
- WO 01413529; Dry Boric Acid Residue on 1SI03AA; February 25, 2011
- WO1293251; M-2RH02013S, VT-3 Examination Record for Component Supports/Attachments; April 26, 2011
- WO1293251; M-2CV02005S, VT-3 Examination Record for Component Supports/Attachments; April 26, 2011

1R11 Licensed Operator Requalification Program

- LORT Simulator Exam; July 21, 2011

1R12 Maintenance Effectiveness

- IR 962281; CO2 Hi Pressure Alarm Came in Early; September 5, 2009
- IR 996016, 2CO181 – 2BAF CO2 Fuse Blow; November 19, 2009
- IR 1023358; The DDFP Prerun Cooling Water Temp is Lower Than 120 Degree F; January 29, 2010
- IR 1035746; OPS – DDFP Drawdown Volts Low; February 25, 2010
- IR 1049432; Small Oil Leak on 0FP03PB, 0B Diesel Fire Pump; March 29, 2010
- IR 1069759; Diesel Driven Fire Pump Had Low Discharge Pressure; May 15, 2010
- IR 1083719; Check Valve 0FP8788B Failed As Found Inspection; June 23, 2010
- IR 1102674; Leak from Diesel Driven Fire Pump; August 17, 2010
- IR 1104545; 0B Fire Pump BwAP 1110-1A2 Extension Past 7 Days; August 22, 2010
- IR 1108238; 0COO14 Relief Lifted during CO Tank Fill; August 31, 2010
- IR 1111602; Need Eng to Eval Operability of 1A D/G CO2 Relay; September 10, 2010
- IR 1112544; UCSR CO2 System Issue; September 13, 2010
- IR 1120807; Potential NCV Identified by NRC for an Ineffective PMT; October 1, 2010
- IR 1124572; CO2 Compressor Running and Very Noisy; October 10, 2010
- IR 1142620; 1CO02J Fuse Blown; November 18, 2010
- IR 1139099; 0B Fire Pump Failed Surveillance (Battery Drawdown Current); November 11, 2010
- IR 1182004; 0B Fire Pump Drawdown Volt Unsat During 0BwOS FP.2.2.M-2; March 1, 2011
- IR 1184825; 0B Fire Pump Failed Drawdown Surveillance 0FP03EB; March 8, 2011
- IR 1188556; 1FP251B Burps Water on Fire Pump Starts; March 17, 2011
- IR 1192662; 0B Fire Pump has an Internal Coolant Leak – 0FP03PB; march 26, 2011
- IR 1196457; Unable to Restore the 0B Fire Pump Within 7 Day GOCAR; April 2, 2011
- IR 1203975; CO2 Pipe Leak While Filling; April 8, 2011
- BwAP 1110-1; Fire Protection Program System Requirements; Revision 30
- BwAP 1110-1A2; Fire Suppression Water Supply Required Compensatory Measures Action Response Cover Sheet; Revision 7

- 0BwOS FP.3.3.E-12; 0B Fire Pump NFPA Test; Revision 8
- Scoping Risk Significance Detailed Report; DC-07 Provide Power, Control, Protection, Indication and Alarms for the Diesel Driven Fire Pump

1R13 Maintenance Risk Assessments and Emergent Work Control

- IR 1193841; Scaffolding Needs the Removal Task Scheduled; March 29, 2011
- IR 1207126; Loose Impeller Nut on 2RH01PA; April 24, 2011
- IR 1207692; Excessive 2A RH Pump Impeller-to-Shaft Clearance Fit Reading; April 25, 2011
- IR 1208649; Resolution Required for Out-of-Spec Readings for 2RH01PA; April 27, 2011
- Prompt Investigation of IR 1208649; Resolution Required for Out-of-Spec Readings for 2RH01PA
- IR 1210405; Engineering Evaluation of 2RH01PB Test Data; May 1, 2011
- IR 1209962; Develop New IST Group Acceptance Ranges for 2RH01PA; April 30, 2011
- IR 1209838; Determine if Impeller from the 2A RH Pump can be Reused; April 29, 2011
- IR 1233017; Failure of 1A FRV Controller and Entry into 1BwOA PRI-16; June 16, 2011
- IR 1233131; 4.0 Critique for 1A FRV Failure; June 26, 2011
- ER-AA-660-1043; Shutdown Risk Management; Revision 5
- 2BwOS 5.5.8.SI-11; Comprehensive Inservice Testing (IST) Requirements for Unit 2 Safety Injection Pumps and Safety Injection System Check Valve Stroke Test; Revision 2
- MTG: PORC 11-012; PORC Review – Freeze Seal
- Night Shift Log; Unexpected Steam Generator 1A Flow Mismatch Steam Flow Alarm Received; June 26, 2011
- OU-AP-104; Shutdown Safety Management Program Byron/Braidwood Annex; Revision 14
- OU-AA-108-117; Protected Equipment Program; Revision 1
- Protected Equipment List; Bus 242; April 19; 2011
- Protected Equipment List; 2A RH Protected for SD Cooling; April 19, 2011
- Protected Equipment List; U2 RWST Makeup Source; April 19, 2011
- Protected Equipment List; 2B CV Protected for Inventory Control; April 19, 2011
- Protected Equipment List; 125 VDC Bus 111 Crosstie to 211; April 19, 2011
- Protected Equipment List; ACB 11-14 and U2 SATs; April 19, 2011
- Protected Equipment List; ACB 14-15 and U2 SATs; April 19, 2011
- Protected Equipment List; 211/213 CVT; April 19, 2011
- Protected Equipment List; 0FP03PB Pump; April 19, 2011
- Shutdown Safety Equipment Status Checklist; April 18, 2011
- Shutdown Safety Equipment Status Checklist; April 19, 2011

1R15 Operability Evaluations

- IR 1184508; Drain Line Plugged (2WF55AA); March 7, 2011l
- IR 1185016; Non-Conservatisms in the Turbine Building HELB Analysis; March 8, 2011
- IR 1185022; Generic Letter 96-06 Analysis Non-Conservatisms; March 8, 2011
- IR 1194353; AF Tell Tale Drain Line Plugged – 2AF03AA; March 20, 2011
- IR 1194703, HELB Concerns With ESF SWGR RM Will Prevent Breaker Swaps, March 30, 2011
- IR 1199223; HELB Past Operability Review; April 12, 2011
- IR 1201265; Deficiencies Noted During Breaker Swap – 1AP04EG; April 7, 2011
- IR 1202772; NRC Questions On Auxiliary Feedwater Pump Suction Piping; April 14, 2011
- IR 1213669; 2AF018A and Downstream Line Blocked with Shells; May 9, 2011
- IR 1217217; Unaccounted for Factor in FW Venturi Discharge Coefficient; May 17, 2011
- IR 1217799; Request WO to Perform Additional Flushing of 2A AF Suction; May 9, 2011

- IR 1217624; NRC Questions Admin Power Limit of 99.5 Percent on Unit 1; May 18, 2011
- IR 1217803; Request Actions to Support Flushing of 2A AF Pump Suction; May 19, 2011
- IR 1217934; NRC Questions Unit 2 Venturi Power Limit; May 19, 2011
- IR 1217999; Unaccounted for Factor in FW Venturi Discharge Coefficient; May 19, 2011
- IR 1218004; Reactor Service Ace Pulled from MRC Review Before MRC Review; May 19, 2011
- IR 1219577; NRC Question on RX Trip Setpoints VS LEFM Derate; May 23, 2011
- Op Eval 11-006; HELB Analysis Input Errors, Revision 0
- Op Eval 11-007; Generic Letter 96-06 Analyses Non-Conservatisms; Revision 0
- Op Eval 11-010; Postulated Void in Auxiliary Feedwater from the Failure of Condensate Storage Tank (CST) Piping; Revision 0
- BwAP 1110-3; Plant Barrier Impairment; Revision 20
- BwAP 1110-3; Plant Barrier Impairment; Revision 21
- 2BwOSR 3.3.1.2-1; Power Range High Flux Setpoint Daily Channel Calibration (Computer Calorimetric); Revision 13
- 1BwOS SX-1; AF Pump SX Suction Line Flush 18 Month Surveillance; Revision 002
- CC-AA-102; Unit 2, Install Cameron LEFM CheckPlus System; EC 377976 Revision 002
- CC-AA-201; Plant Barrier Control Program; Revision 8
- EC 384067; Op Eval 11-007, AF Pump Suction Concerns; April 20, 2011
- EC 384507; Clamshells in Piping between MOV 2AF006A and 2AF017A; Revision 00
- FAI/11-364; AFW Air Intrusion Analysis; April 6, 2011
- OP_AA-102-104; Unit 1 Calorimetric Power Limit; Revision 1
- WR 00360954; Drain Line Plugged (2WF55AA; March 9, 2011
- UFSAR Table 3.11-13
- Generic Letter 89-13; Service Water System Problems Affecting Safety-Related Equipment; July 18, 1989
- Calc. No. VRW-97-1038-M; Required Relief Capacity of Isolated Pipe Segments Needed to Meet GL 96-06 Concerns; Revision 0
- Calc. No. BYR-99-027/BRW-99-0050-M; Thermal Heat Up of Isolated Pipe Through Penetrations P-5, P-8, P-37, P-55, P-70, and P-71; Revision 0
- Calc. No. BYR-99-073/BRW-99-0228-M,; Evaluation of Penetration Areas for Thermal Over-Pressurization Following a LOCA (GL 96-06); Revision 0
- Calc No. BYR-99-076/BRW-99-0219-M; Redistribution of Piping Loads due to GL 96-06 Thermal Over-Pressurization; Revision 0
- Calc. No. MWMECH-03-001; Evaluation of the Braidwood and Byron RCFC Thermal Hydraulic Loads During LOOP/LOCA Using the EPRI Methodology for GL 96-06; Revision 0
- Calc. No. PSA-B-98-13; Thermal Hydraulic Behavior of RCFC System During LOCA/LOOP for Byron and Braidwood Stations; Revision 0
- Sargent & Lundy Analysis; Updated Containment Profile Impact on Reactor Containment Fan Coolers (RCFCs) Void Collapse; February 10, 2011
- Westinghouse Calc. No. CN-CRA-00-37; Byron/Braidwood Containment Response to SLB for Units 1 and 2 Uprating; Revision 0

1R19 Post-Maintenance Testing

- EC 384103; Document Replacement of Polar Crane Load Cell Design Summary
- IR 1203359; Polar Crane Load Cell Needs Evaluation/Drawing Updates; March 17, 2011
- IR 1206020; Polar Crane Load Cell Inaccurate Reading During RX Head Lift; April 21, 2011
- IR 1206053; CV LLRT Configuration Control Level 3 Event; April 20, 2011
- IR 1206466; A2R15LL Reactor Cavity Inventory Too Low for Head Lift; April 21, 2011

- IR 1208004; Reactor Service Ace Pulled from MRC Review Before MRC Review; May 19, 2011
- BwAP 340-1; Use of Procedures for Operating Department; Revision 24
- BwMP 3100-009; Reactor Vessel Closure Head Removal; Revision 20
- 1BwOSR 3.4.11.2; Pressurizer System PORV Valve Stroke Surveillance; Revision 5
- 2BwOSR 3.6.3.5.RF-1; Reactor Building Floor Drain Containment Isolation Valve Stroke Quarterly Surveillance; Revision 2
- 2BwOSR 5.5.8.RY-2; Pressurizer System Isolation Valve Indication Surveillance; Revision 2
- HU-AA-104-101; Procedure Use and Adherence; Revision 4
- Reg Guide 1.33; QA Program Requirements (Operation); Revision 2; February 1978
- WO 1360260 03; OP PMT 2FSV-RF027; Stroke Test and Verify No Air Leaks; May 2, 2011
- WO 1270567 04; OP PMT – 2RY456 – Stroke Time Per 2BwOSR 3.4.11.2; May 2, 2011
- WO 1270567 05; OP PMT – 2RY456 – Ind Test Per 2BwOSR 5.5.8.RY-2; May 2, 2011
- WO 1270567 06; OP PMT – 2RY456 – Actuator Leak Check; May 2, 2011

1R20 Refueling and Other Outage Activities

- IR 1202770; OPEX Byron IR – Potential Unborated Water Sources; April 13, 2011
- IR 1203987; 2SI8900A Check Valve Back Leakage; April 18, 2011
- IR 1206940; Unable to Contact U2 Containment Closure Contact; April 24, 2011
- IR 1207429; Failure of 2A SX Strainer Functionality Evaluation; April 25, 2011
- IR 1207433; A2R15LL – RX Head Hoist Potential Electrical Safety Hazard; April 25, 2011
- IR 1208707; A2R15 CRDM Inspector Identified 3 Penetrations That Req Disposition; April 27, 2011
- IR 1208769; Loss of FME Integrity for 2SI8880; April 28, 2011
- IR 1208471; Axial ODSCC Found in the 2D SG During A2R15; April 26, 2011
- IR 1208516; Unqualified Coating Not Listed in Containment Coating Log; April 27, 2011
- IR 1208777; 2A CV Pump Curve Analysis with 2FE-0917 Out-of-Tolerance; April 22, 2011
- IR 1209287; Unit 2 Startup Feedwater Pump Motor Heater Breaker Bumped; April 28, 2011
- IR 1209943; Stripped Screw Causes Inadvertent Breaker Operation; April 30, 2011
- IR 1210274; D Construction Worker Taken Off Site for Medical Treatment; April 30, 2011
- IR 1210282; Axial End Play Measurement Out-of-Tolerance: 2RC01PB; April 30, 2011
- IR 1210386; OPEX – Byron IR 1207568 – Over Power Trip Setpoint Issue; April 25, 2011
- IR 1210418; U2 RX Cavity Sump Flow Indication Failed High; May 2, 2011
- IR 1210517; 2PI-SI116 Reads Incorrectly During Surveillance Testing; May 2, 2011
- IR 1210876; Unplanned ED Dose Rate Alarm Received; May 2, 2011
- IR 1210895; Inadequate IR Resolution; May 2, 2011
- IR 1211011; Operator Injury While Moving Nitrogen Bottles; May 3, 2011
- IR 1211357; 2B CV Pump Shaft Hard to Turn; May 3, 2011
- IR 1211413; 2SI8875A-I/A Found Closed; May 3, 2011
- IR 1211461; NRC Concern TCC Not Procedurally Controlled on 2A AF System; May 3, 2011
- IR 1211597; 2MS001C Failed to Partially Stroke; May 4, 2011
- IR 1211189; FME Found in Valve Internals of 2SI8822B; May 3, 2011
- IR 1211752; Fatigue Assessment Results Satisfactory; May 4, 2011
- IR 1211799; 2B CV Pump Room Supply Damper 0VA305Y Failed Closed; May 4, 2011
- IR 1211879; NRC Questions Pipe Attached to Beam in VA Chiller Room; May 5, 2011
- IR 1211890; NRC Identified 3 Items of Concern to the WCC; May 4, 2011
- IR 1214302; 2F-RF010 Loop Indicates Max Flow; May 10, 2011
- IR 1215470; 2FW009A Will Not Open After Repeated Attempts; May 13, 2011
- IR 1217223; RSH CW M/U Pumps Tripped with No Apparent Cause; May 17, 2011

- CAP102 Report BRW MRC CR Review CR-1208987; A2R15LL – 2CV01PA Alignment Issue; April 28, 2011
- 1BwGP 100-5T1; Flowchart – Continuous Use; Revision 16
- 2BwGP 100-5; Plant Shutdown and Cooldown; Revision 37
- BwOP DG-11T1; Diesel Generator Start/Stop Log; Revision 7
- BwOP DG-11T2; Diesel Generator Operating Log; Revision 22
- BwOP RC-4; Reactor Coolant System Drain
- 2BwOSR 3.3.2.8-611A; Unit Two ESFAS Instrumentation Slave Relay Surveillance (Train A Automatic Safety Injection – K611); Revision 8
- 2BwOSR 3.8.1.2-1; 2A Diesel Generator Operability Surveillance; Revision 31
- BwVS 500-6; Low Power Physics Test Program; Revision 29
- Status View Report; Model: U2 A2R15; Base Data Modified March 23, 2011
- ACB 11-14 and U2 SATs; SAT 242-1 and SAT 242-2; ESF Power Supply
- ACB 14-15 and U2 SATs; SAT 242-1 and SAT 242-2; ESF Power Supply
- 2B CV Protected for Inventory Control; 2B CV Pump and Train Designated Protected for Inventory Control
- EC 383301; Install Vent Valves in Pipe 2AF03AA-6 Between Valves 2AF006A and 2AF017A; Revision 000
- EC 383328; Vacuum Filling the Pipe Segment Between the Two SX Crosstie Valves to the AF Pumps; Revision 0
- ER-AA-600-1043; Shutdown Risk Management; Revision 5
- ER-AP-331; Boric Acid Corrosion control (BACC) Program; Revision 5
- NF-AP-537; Beacon Shutdown Margin (SDM) Calculation; Revision 7
- WO 1439814 01; IST-2A DG Operability Monthly; June 2, 2011

<u>1R22 Surveillance Testing</u>

- IR 1223180; Work Delayed 8 Hours; May 31, 2011
- IR 2123669; 2AF018A and Downstream Line Blocked with Metal Shavings; May 9, 2011
- BwAR 1-2-A2; SX Pump Discharge Header Pressure Low; Revision 7a
- BwMSR 3.7.1.1; Main Steam Safety Valves Operability Test (Setpoint Verification Using the Furmanite Trevitest System); Revision 2
- 2BwOL 3.3.1; LCOAR Reactor Trip System (RTS) Instrumentation Tech Spec LCO 3.3.1; Revision 8
- 0BwOS SX-SA1; Essential Service Water – Fire Protection Systems Crosstie Flush; Revision 6
- 1/2BwOSR 3.3.2.8-611A/B; Pre Job Brief for DG Slave Start Surveillance
- 2BwOS SX-1; U2 AF Pump SX Suction Line Flush – 18 Month Surveillance; Revision 003
- 2BwOS XPC-W1; U2 Containment Penetration Status Weekly Surveillance; Revision 17
- WC-AA-104; 2T-0421/0422 18 Month Cal 2B Delta T/TAVE & Card Replacement; Revision 17
- WO 01253710 01; 2T-0421/0422 LOOP; January 22, 2011
- WO 01288008 01; IST-RT-2MS013S/014S/015S/016S/017S-MSSV Operability Test; April 15, 2011
- WO 01404389 01; U1 TRN B Relay Surveillance K611; April 13, 2011
- WO 01419253 01; IST-1B DG Operability Monthly; April 13, 2011
- EC 384321; 2A Train of AF System – Finding Clam Shells in Piping; Revision 00
- Drawing M-122; Diagram of Auxiliary Feed Water – Unit 2
- Drawing M-126; Diagram of Essential Service Water - Unit 2
- Drawing 2A-AF-31; Auxiliary Feedwater – Unit 2
- Drawing 2A-SX-89; Essential Service Water – Unit 2
- A2R15 Containment Closure Plan

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- RWP 10011817; A2R15 Remove and Reinstall Reactor Head and Upper Internals; Revision 0
- RWP 10011816; Reactor Head Component Disassembly and Reassembly Included Lift Preparation; Revision 0
- RWP 10011819; Seal Table Room Outage Activities; Revision 0
- RWP 10012559; Install Reactor Cavity Elevator; Revision 1
- RWP 10011790; A2R15; Lead Shielding Install/Maintain/Remove at Aux and Containment Buildings; Revision 0
- RWP 10011791; A2R15; ISI Activities Includes Weld Prep and Exams at Aux and Containment Buildings; Revision 0
- RWP 10011782; A2R15; FHD Outage Support Activities (No Fuel Moves/Tri-Nuke Work); Revision 2
- AR 01205144; Two PCEs Received on Removal of Sandbox Covers; April 14, 2011
- AR 01205772; Electronic Dose Rate Alarm Received during In-Service-Inspection Activities; April 21, 2011
- AR 01205138; Two Personnel Contamination Events were Received on Reactor Cavity Work Boot Seal Work; April 19, 2011
- RP-AA-441; Evaluation and Selection Process for Radiological Respirator Use; Revision 4
- RP-AA-300; Radiological Survey Program; Revision 7
- RP-AA-220; Revision 6; Attachment 2; Intake Investigation Form; Unit-1 Containment Entries where the Tritium Greater than 0.3 DAC; July 06, 2010
- RP-AA-301; Radiological Air Sampling Program; Revision 4
- RP-AA-460; Controls for High and Locked High Radiation Areas; Revision 20

2RS2 ALARA Planning and Controls (71124.02)

- RP-AA-400; ALARA Program; Revision 7
- RP-AA-401; Operational ALARA Planning and Control; Revision 12
- A2R15 Force Oxidation ALARA Guidelines; April 16, 2011
- ALARA 10011817; A2R15 Remove and Reinstall Reactor Head and Upper Internals; Revision 0
- ALARA 10011816; Reactor Head Component Disassembly and Reassembly Included Lift Preparation; Revision 0
- ALARA 10011819; Seal Table Room Outage Activities; Revision 0
- ALARA 10012559; Install Reactor Cavity Elevator; Revision 1
- ALARA 10011790; A2R15; Lead Shielding Install/Maintain/Remove at Aux and Containment Buildings; Revision 0
- ALARA 10011791; A2R15; ISI Activities Includes Weld Prep and Exams at Aux and Containment Buildings; Revision 0
- Braidwood Station A2R15 Elevated Dose Rate Contingency action Plan

2RS3 In-Plant Airborne Radioactivity Control and Mitigation

- Audit NOSA-BRW-09-06 (IR 851243); Radiation Protection Audit Report; Braidwood Station; August 13, 2009
- Audit NOSA-BRW-07-06 (IR 571106); Radiation Protection Audit Report; Braidwood Station; August 16, 2007
- D12-THP-6010-RPP-009; Emergency Equipment Inventory; Revision 27
- Annual Re-evaluation of Braidwood Nuclear Power Station's Prospective Evaluation for the Year 2009; September 1, 2010

- RP-BR-980; Attachment 1; Containment Release Form; Release No. G-11-054; March 29, 2011
- RP-BR-980; Attachment 1; Containment Release Form; Release No. G-11-054; March 28, 2011
- RP-BR-980; Attachment 1; Containment Release Form; Release No. G-11-054; March 29, 2011
- Email from James Gerrity; List of Names for the New SCBA Eye Glasses Inserts; April 12, 2011
- IR 00871887; Lack of Time Validation for SCBA Use by Licensed Operator; January 26, 2009
- IR 1106607; Respirator Medical Evaluations Not Entered Into PADS; August 24, 2010
- IR 1196255; NRC Inspection Identified No MCR Storage Area for SCBA Mask Inserts; April 1, 2011
- IR 1198986; Enhancement Identified by NRC during SCBA Inspection; April 1, 2011
- IR 1136359; Nuclear Oversight Services ID ISI SCBA Clutter in the MUDS; November 5, 2010
- IR-1207180; SCBA Kits Enhancement Identified during RP NRC Inspection; April 25, 2011
- RP-AA-220; Revision 6; Attachment 2; Intake Investigation Form; Unit-1 Containment Entries where the Tritium Greater than 0.3 DAC; December 9, 2010
- RP-AA-220; Revision 6; Attachment 2; Intake Investigation Form; Unit-1 Containment Entries where the Tritium Greater than 0.3 DAC; July 6, 2010
- PSI 0047937; Quarterly Service Air and Self Contained Breathing Apparatus Performed between March 11, 2010 and March 24, 2011
- RP-AA-825; Maintenance, Care and Inspection of Respiratory Protective Equipment; Revision 3
- RP-AA-825-1035; Issue and Control of Respirators; Revision 0
- RP-BR-827; Operation, Use and Inspection of Self Contained Breathing Apparatus
- RP-BR-828; Charging of Air Cylinder for SCBA

2RS4 Occupational Dose Assessment

- RP-AA-203-1001; Sample Personnel Exposure Investigation; Revision 6
- RP-AA-203-1001; Personnel Exposure Investigation; Revision 6
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; EDE Results on an Individual; November 18, 2010 Whole Body Counter; Revision 3
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; EDE Results on an Individual; November 23, 2010
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; EDE Results on an Individual; May 19, 2010
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; EDE Results on an Individual; November 18, 2010
- RP-AA-220; Revision 6; Intake Investigation Form; RWP 10009632; Intake Less Than 10 mrem; March 26, 2009
- RP-AA-270; Declared of Pregnancy; January 13, 2011
- NVLAP Certification of Accreditation to ISO/IES 17025:2005; Mirion Technologies (GDS), Inc.; July 7, 2010
- RP-AA-220; Revision 6; Intake Investigation Form; RWP 10011355; November 22, 2010
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; ED and TLD Dose Discrepancy; d February 11, 2011
- RP-AA-203-1001; Personnel Exposure Investigation: Attachment 1; ED and TLD Dose Discrepancy; January 25, 2010
- IR 1189298; Individual Splashed with Small Amount of Process Water; March 17, 2011
- IR 0983053; Worker Receives Unplanned ED Dose Alarm; December 2, 2009

- RP-AA-222; Methods for Estimating Internal Exposure from In Vivo and In Vitro Bioassay Data; Revision 3
- IR 11244049; ED Alarm Received in U-1 Containment 401; October 6, 2010
- IR 1122591; ED Dose Rate Alarm Received During Unit-1 RCS Sampling; October 9, 2010
- IR 1125938; ED Alarm Received in 401 Auxiliary HRSS Room; October 8, 2010
- IR 1124564; ED Alarm Received in 383 Auxiliary CV Valve Aisle; October 7, 2010
- Calibration of the Canberra Fastscan A1 WBC System; September 3, 2010

2RS6 Radioactive Gaseous and Liquid Effluent Treatment (71124.06)

- Braidwood Station Unit 1 and 2; Facility Operating License NPF-72 and NPF-77; 2010 Radioactive Effluent Release Report; April 29, 2011
- BwOP WX-526T1; Liquid Release Tank 0WX26T Release Tank WM; Revision 60
- Braidwood Radiological Spills and Unusual Occurrences per 10 CFR 50.75(g) and 10 CFR 72.30(d); June 29, 2010
- Sampling and Reporting Requirement for the North Oil Separator and Main Drainage Ditch; Revision 8
- Braidwood Nuclear Plant (PWR) Chemistry Report; May 24, 2011
- RP-BR-928; Unit 1 and 2 RE-PR028J Radiation Monitor Air Sampling; Revision 2
- RP-BR-655; Alternate Sampling Methods from General Atomic Process Radiation Monitors; Revision 0; Sampling Date May 25, 2011
- RP-AA-228; Record for 10 CFR 50.75(g) or 10 CFR 72.30 (d); Revision 0
- OF-VA020; Calibration of Aux. LDG EXH; Revision 1; February 6; 2010
- EN-AA-408; Radiological Groundwater Protection Program; Revision 0
- 12-EHP-6040-028-111; Containment Pressure Relief System Performance Test for Unit 2; Revision 04; March 25, 2010
- DB-SS4045-001; High Efficiency Particulate Air Filters and Charcoal Absorbers Test; for Containment Purge Exhaust Charcoal Filters; July 19, 2010
- EN-AA-408-4000; Radiological Groundwater Protection Program Implementation; Revision 0; May 23, 2011
- LS-AA-126-1005; Self-Assessment; Radioactive Gaseous and Liquid Effluent Treatment, and NRC Performance Indicator Verification; April 23, 2011
- Liquid Permit Status Summary Report; Gaseous Permit Status Summary Report; RETDAS Version 3.6.3; May 23, 2011
- BwRP-6110-13T1; Revision 18; Containment Release Form; Release Number G-09-107
- RP-BR-980; Revision 2; Containment Release Form; Release Number G-10-90
- RP-BR-980; Revision 2; Containment Release Form; Release Number G-10-160
- BwOP-WX-501T1; Liquid Release Tank 0W01T Release Form; Revision 56
- BwOP-WX-526T1; Liquid Release Tank 0WX26T Release Form; Revision 59
- IR 1000563; Exelon Pond ODCM Composite: No Sample One Week; December 21, 2009
- IR 0996564; 0PR01J High Alarm While Performing Liquid Radwaste Release; November 20, 2009
- IR 0880761; No ODCM Composite Sample Due to Cold Weather; February 14, 2009
- IR 1067027; No Exelon Pond Weekly ODCM Sample; May 8, 2010
- IR 1092717; Excessive Water Found in AB Ventilation Plenum; July 21, 2010
- IR 1115528; 0B Fuel Handling Building Charcoal Booster Fan Starting Reason was Unknown; September 20, 2010
- IR 1221439; Fe-55 Found in 2010 Second Quarter But Not Included in 2010 Annual Radioactive effluent Release Report; May 27, 2011
- IR 1221220; Difference in Byron/Braidwood use of RETDAS Program for Hard to Detect Nuclides; May 26, 2011

- DB-HP-01312; Testing of Portable HEPA Filtered Equipment; Revision 2
- 0BwOS RETS 2.1-1a; ODCM Radioactive Effluent Technical Standards Instrumentation – Liquid Effluent Monitoring RETS Operability Requirement 12.2.1A; Revision 6
- BwRP 6110-6T3; Continuous Radioactive Liquid Effluents Quarterly Compliance Verification; Applicable to Third and Fourth Quarter 2010; Revision 2

4OA1 Performance Indicator Verification

- PI Summary of Braidwood Station; Reactor Coolant System Activity; Between January 2010 and December 2010
- PI Summary of Braidwood Station; Liquid Release and Dose/Gaseous Release and Dose; Between January 2010 and April 2011
- LS-AA-2090: Monthly Data Elements for NRC Reactor Coolant System (RCS) Specific Activity from First Quarter of 2010 through Fourth Quarter 2010; In Microcuries Per Gram Dose Equivalent Iodine-131
- LS-AA-2150-RW5; NRC RETS/ODCM Radiological Effluent Occurrences from the First Quarter 2010 Through First Quarter 2011

4OA2 Identification and Resolutions of Problems

- IR 1095900; Questions Regarding 50.59 Requirements for Scaffolding; July 29, 2010
- IR 1174385; 2MS037C Steam Leak; February 12, 2011
- IR 1122116; Incorrect Interpretation of MA-AA-716-025; October 4, 2010
- IR 1183271; NOS ID Scaffold Program Adverse Trend; March 3, 2011
- IR 1184825; 0B Fire Pump Failed Drawdown Surveillance 0FP03EB; March 8, 2011
- IR 1192662; 0B Fire Pump Has an Internal Coolant Leak – 0FP03PB; March 26, 2011
- IR 1193841; Scaffolding Needs the Removal Task Scheduled; March 29, 2011
- IR 1196457; Unable to Restore the 0B Fire Pump Within 7 Day GOCAR; April 2, 2011
- IR 1199270; Engine Coolant Leak – Maintenance Rule Functional Failure; April 7, 2011
- IR 1200851; Action Requested on IR1118546 Dated September 27, 2010; April 11, 2011
- IR 1201338; NER 11-009, No Removal Date for Scaffold Around CO^2 Compress; April 12, 2011
- IR 1201890; Temporary Scaffold Not Removed; April 12, 2011
- IR 1204477; NER 11-009 – Deficiency/Enhancement: IEMA Question; April 19, 2011
- IR 1204481; NER 11-009 – Deficiency/Enhancement: Flange Cart; April 19, 2011
- IR 1204629; IRS Initiated With Incorrect Event/Discovery Dates; April 19, 2011
- IR 1204875; B.5.B Pump Run Documentation of Results; April 19, 2011
- IR 1204922; Water Leaking Into U2 VCT Valve Aisle onto 2CV112B; April 19, 2011
- IR 1204989; Need B.5.B Fuel Moves Completed following A2R15 Onload; April 20, 2011
- IR 1205005; Discrepancy in Procedures for Letdown Line Pressure Limit; April 20, 2011
- IR 1205012; NOS Identifies Fire Hazards Not Accounted for by Shaw Pipefitters; April 20, 2011
- IR 1205022; Shaw Employee Leaves Security Door 218 Unsecured; April 20, 2011
- IR 1205031; Scaffolds Will Reach 90 Days Next Month; April 20, 2011
- IR 1210519; Flow Change During 2BwOSR 5.5.8.SI-11; May 2, 2011
- IR 1206426; Untimely Response Related Disposition of Temporary Scaffolds; April 22, 2011
- IR 1207275; Draft LER Rejected at PORC; April 21, 2011
- IR 1208100; Cooling Lake Level Reaches Trigger of EN-BR-402-0005; April 26, 2011
- IR 1208605; I-131 Detected in REMP Milk Sample from Location BD-17; April 26, 2011
- IR 1208681; Sewage Treatment Effluent Elevated BOD – National Polution Discharge Elimination System; April 27, 2011

- IR 1208682; 2AP56E-C5 Thermal Overload Reset Rod Broken; April 27, 2011
- IR 1209057; Elevated Tritium in H Ditch North Oil Separator; April 28, 2011
- IR 1210519; Flow Change During 2BwOSR 5.5.8.SI-11; May 2, 2011
- IR 1210947; Loss of FME Integrity for 2SI8822D; May 2, 2011
- IR 1211303; 0FP03PB Request Procurement Engineering Evaluate Heads; May 3, 2011
- IR 1211321; 0FP03PB New Heads Have Flaws and Holes Partial Bored; May 3, 2011
- IR 1211461; NRC Concern TCC Not Procedurally Controlled on 2A AF System; May 3, 2011
- IR 1211517; Foreign Material Found in Kerotest Check Valve 2SI8968; May 4, 2011
- IR 1214400; 1MP01E West Bus Duct Cover Loose; May 11, 2011
- IR 1214454; NRC Identified Packing Leak 2MS009B; May 11, 2011
- IR 1215475; PCRA Needed to Revise 2BwOSR 3.3.1.14-1; May 11, 2011
- IR 1216584; NRC Identified Issues with SAMG's; May 16, 2011
- IR 1217150; UT Examination Results on 1SI06BB; May 17, 2011
- IR 1217246; TSs Elevated in MUDS Composite; May 17, 2011
- IR 1217217; Unaccounted for Factor in FW Venturi Discharge Coefficient; May 17, 2011
- IR 1217311; 1A SI Pump Discharge Header Needs Ultrasonic Testing; May 18, 2011
- IR 1219630; NOS ID: OPS Non-Job Reading Material; Professional Attire; May 23, 2011
- IR 1223080; Rods Failed to Move When Demanded in Manual; May 31, 2011
- IR 1223177; Breaker 1562 Replacement Approaching Crit Date HELB Concern; June 1, 2011
- IR 1223422; Resident NRC Questions on PBI Program; May 31, 2011
- IR 1223444; 1A AF Suction: Need WO for Flushing per Troubleshooting Plan; June 1, 2911
- IR 1223463; NOS IDs Unauthorized Temporary Configuration Change (TCC); June 1, 2011
- IR 1225208; LEFM Implementation and PPC Calorimetric Issue; June 6, 2011
- IR 1226383; 4.0 Critique for SX Flush Line Failure; June 8, 2011
- IR 1227373; AF Pumps Start/Trip Not Addressed in Key Design Basis Docs; June 6, 2011
- IR 1227522; 1B DG Starting Air Comp. Not Starting at Proper Pressure; June 11, 2011
- IR 1227553; 1B FW Pump Speed Oscillations; June 11, 2011
- IR 1227591; Overdue Engineering Department Evaluation; June 6, 2011
- IR 1227683; Received Spike of the 1PR28J Gas Channel (Low Range); June 12, 2011
- IR 1227829; Spurious Rod Deviation Power Range Tilt Alarm; June 13, 2011
- IR 1227869; 1A DG 1R Fuel Pump Weep-Leak; June 11, 2011
- IR 1227935; PRA Changes for June 6, 2011 Work Week; June 13, 2011
- IR 1227988; 0B VA Supply Fan Diff Press High alarm – 0VA01CB; June 13, 2011
- IR 1228056; Incorrect CA Closure (Engineering Programs Group); June 13, 2011
- IR 1228123; 0PR60J in Alert Alarm (0PA269 Particulate Channel); June 13, 2011
- IR 1228215; OPS ID: Rounds Issues June 13, 2011 Days; June 13, 2011
- IR 1228270; Present Lake Status to PHC-Lake Root Cause; June 14, 2011
- IR 1228511; 2A DG Jacket Water Heater is Not Working – 2DG01KA-D; June 14, 2011
- IR 1228621; NOS ID Yellow Man Unprofessional; June 14, 2011
- IR 1228655; As Found of 2C CW Bay for Bryzoa Criteria – 2CW01PC; June 14, 2011
- IR 1228694; SX Makeup to CC Approved as Fast Track Project; March 30, 2011
- IR 1228881; SGTR-MTO PORV AF005 Accum Project Installation; June 15, 2011
- IR 1228888; SGTR-MTO PORV Trim Project Installation; June 15, 2011
- IR 1228999; NOS ID No Urgency in Component Failure Determination; June 15, 2011
- IR 1229038; Learning Programs Deep Dive Gap #1 (CAP – SOC/MRC); June 15, 2011
- IR 1230109; Unit 2 Operator Calorimetric Standing Order Need Changed; June 17, 2011
- IR 1230248; Summer Readiness – Missed Opportunity 2+ Years 0VT08J; June 18, 2011
- IR 1230250; Summer Readiness – Missed Opportunity 0VT09J; June 18, 2011
- IR 1230260; Unexpected Annunciator 2-1-B-5; June 17, 2011
- IR 1230287; Unclear Procedural Direction – BwOP CW-9; June 18, 2011
- IR 1230303; 1A FW PP L.O. Filter High DP; June 18, 2011

- IR 1230314; 2C MSIV Active Side Accumulator Pressure Low Alarm; June 18, 2011
- IR 1230376; 1A DG Jacket Water Leak; June 18, 2011
- IR 1230383; U1 SVAG Switch Indication Issue; November 3, 2010
- IR 1230405; New SX Valve Stroke Surveillance Does Not Work Effectively; June 19, 2011
- IR 1230421; Shift Staffing Temporarily Below BwAP 320-1 Desired Level; June 19, 2011
- IR 1230460; Procedure Enhancements 1/2BwOSR 3.4.14.1; June 19, 2011
- IR 1230461; Floor Drain Backup on 383' level; June 19, 2011
- IR 1230472; Weakness in OP-AA-201-010-1001 Incorporation Discovered; June 19, 2011
- IR 1230474; Enhancement on OP-AA-201-010-1001 Incorporation Discovered; June 19, 2011
- IR 1230485; Corrosion of VA/WO Coil Valve Bonnets; June 19, 2011
- IR 1230509; Monthly Temp Scaffold 90-day Review – KT&R 989925; June 20, 2011
- IR 1233849; 1B Forebay Diver Inspection Findings; June 28, 2011
- IR 1235115; 2A Forebay Diver Inspection Findings; June 30, 2011
- IR 1235324; MS/FW Tunnel Pen Cooling Low Flow; June 30, 2011
- IR 1235362; Failure to Schedule Work Accurately Again; June 29, 2011
- MA-AA-716-025; Scaffold Installation, Modification, and Removal Request Process; Revision 8
- WO 1379395; MM OLLO Scaffold to Safely Reach Emergency Ltg. Unit; December 16, 2010
- 90 Day Scaffold List as of May 12, 2011

4OA3 Followup of Events & Notices of Enforcement Discretion

- IR 1210777; Freon Leak from 0WO03CC Box Valve #29 Packing; May 2, 2011
- IR 1226959; Inadvertent Siren Activation for Braidwood/Dresden – BD11; June 9, 2011
- Unusual Event Declared; IR 1226235 - Temporary Hose Used for Maintenance Work Ruptured; June 8, 2011
- Event Summary Report; Termination from an Unusual Event; June 8, 2011

4OA5 Other Activities

- IR 1202772; NRC Questions on AF Pump Suction Piping; April 14, 2011
- SAMG; PWR Sever Accident Management Administrative Guidance; Revision 0
- EC384067; OpEval 11-010, AF Pump Suction concerns; April 20, 2011
- FAI/11-364; AF Air Intrusion analysis; April 6, 2011

4OA7 Licensee-Identified Violations

- IR 1106414: Non-Conservative Liquid Discharge Alarm Set-Points; August 26, 2010

IR 1082508; NRC IDENTIFIED INCONSISTENT GAMMA ENERGY SENSITIVITY USED WITHIN THE DOCUMENT FOR PROCESS MONITOR 0PR01J; JUNE 6, 2010

LIST OF ACRONYMS USED

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AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AF	Auxiliary Feedwater
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BAE	Boric Acid Evaluation
BMV	Bare Metal Visual
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
FDS	Fire Damage State
HELB	High Energy Line Break
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
ISI	Inservice Inspection
kV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOOP	Loss of Offsite Power
MCC	Motor Control Center
NCV	Non-Cited Violation
NDE	Non Destructive Exam
NRC	U.S. Nuclear Regulatory Commission
NVLAP	National Voluntary Laboratory Accreditation Program
OWA	Operator Workaround
PARS	Publicly Available Records System
PBI	Plant Barrier Impairment
PI	Performance Indicator
psia	Pounds Per Square Inch Absolute
psig	Pounds Per Square Inch Gauge
RCS	Reactor Coolant System
RHR	Residual Heat Removal
SAMGs	Severe Accident Management Guidelines
SDP	Significance Determination Process
SG	Steam Generator
SSCs	Structures, Systems, and Components
TS	Technical Specification
TSO	Transmission System Operation
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Testing
VT-3	VT-3 Visual Examination
WO	Work Order

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Sincerely,

/RA/

Eric R. Duncan, Chief
Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000456/2011003; 05000457/2011003
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Letter to M. Pacilio from E. Duncan dated August 9, 2011.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, NUCLEAR REGULATORY COMMISSION INTEGRATED INSPECTION REPORT 05000456/2011003; 05000457/2011003

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