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

February 8, 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, NRC INTEGRATED INSPECTION REPORT 05000456/2010005; 05000457/2010005

Dear Mr. Pacilio:

This refers to the inspection completed on December 31, 2010, at your Braidwood Station, Units 1 and 2. The enclosed report presents the results of this inspection which were discussed on January 5, 2011, with Mr. A. Shahkarami, and other members of your staff.

During this inspection, the NRC staff examined activities conducted under your license as they relate to public health and safety to confirm compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has determined that one Severity Level IV violation of NRC requirements occurred. The NRC has also identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has determined that two violations are associated with these issues. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report. Additionally, a licensee-identified violation is 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 the 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 NRC Senior Resident Inspector at the Braidwood Station.

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

Sincerely,

/RA/

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 05000456/2010005; 05000457/2010005
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:	05000456/2010005; 05000457/20100005
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	October 1 through December 31, 2010
Inspectors:	J. Benjamin, Senior Resident Inspector D. Betancourt-Roldan, Resident Inspector J. Dalzell-Bishop, Emergency Response Specialist J. Robbins, Byron Resident Inspector N. Feliz-Adorno, Reactor Inspector T. Go, Health Physics Inspector M. Holmberg, Reactor Inspector R. Jickling, Senior Emergency Preparedness Inspector D. Jones, Reactor Inspector R. Ng, Project Engineer B. Palagi, Senior Operations Engineer M. Thorpe-Kavanaugh, Reactor Engineer M. Perry, Resident Inspector Illinois Emergency Management Agency B. Metrow, American Society of Mechanical Engineers Inspector, Illinois Emergency Management Agency
Approved by:	E. Duncan, Chief Branch 3 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000456/2010005, 05000457/2010005; 10/01/2010 - 12/31/2010; Braidwood Station, Units 1 & 2; Fire Protection; Identification and Resolution of Problems; Followup of Events

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. This report contains one NRC-identified Green finding, one NRC-identified Severity Level IV violation, and one self-revealed Green finding. Two of these issues 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

Cornerstone: Initiating Events

- Green. A finding of very low safety significance was identified by the inspectors when licensee personnel failed to adequately utilize operating experience that ultimately contributed to an August 16, 2010, Unit 2 reactor trip. Specifically, the licensee did not properly evaluate received operating experience as documented in Issue Report (IR) 259836, “OPEX Review: Isophase Bus Ground Faults.” A portion of this document emphasized the need to consider re-evaluating the associated preventative maintenance frequency for deionizer grids, louvers, and dampers if the isophase air flow through these devices had been raised since the last inspection. The station had occasionally raised air flow since 2002 and no actions were taken to address this portion of the IR. On August 16, 2010, pieces of an isophase crossover damper broke off and caused a phase to ground short, resulting in a turbine trip and automatic reactor trip. The licensee’s root cause evaluation determined that not properly evaluating this portion of the IR was a missed opportunity and likely contributed to the cause of the trip. The licensee entered this issue into their corrective action program (CAP) as IR 1101855. Corrective actions for this issue included reevaluating the operating experience and revising the preventative maintenance schedule to ensure crossover dampers are inspected and/or replaced prior to failure, with the scheduled periodicity to be based upon a thorough engineering analysis. The maintenance procedure for the isophase bus duct was also revised to include inspection criteria for the crossover dampers.

  The inspectors determined that the failure to adequately evaluate readily available industry operating experience in accordance with station procedure LS-AA-115, “Operating Experience Program,” was a performance deficiency. Specifically, the station concluded the operating experience was not applicable to Braidwood station even though air flow through the dampers had been raised occasionally since 2002 and no actions to reevaluate the preventive maintenance frequency were taken. The finding was determined to be more than minor because it was associated with the Procedure Adequacy 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 performance deficiency contributed to the cause of the August 16, 2010, Unit 2 reactor trip. The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events cornerstone. The finding screened as having very low safety significance (Green) because it was determined not to contribute to both a plant trip and the likelihood that mitigating system equipment or functions would not be available. The inspectors did not identify a cross-cutting aspect associated with this finding since it was not considered to reflect current performance. (Section 4OA3.5)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance and an associated NCV of License Condition 2.E was identified by the inspectors when licensee personnel failed to maintain a fire seal between Unit 2 Fire Zone 11.6-2 on the 426′ elevation and Unit 2 Fire Zone 11.5A-2 on the 414′ elevation of the auxiliary building and adjacent to the containment structure in accordance with the approved Fire Protection Program. This issue was entered into the licensee's corrective action program as IR 1126534. Corrective actions consisted of implementing a fire watch for this area until the seal was repaired. In addition, the licensee performed an extent of condition review and entered additional related deficiencies into the correction action program.

  The inspectors determined that the failure to identify and implement corrective actions for a degraded fire seal between two fire areas was contrary to the approved Fire Protection Plan and was a performance deficiency. The degraded fire seal was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of Protection Against External Factors (Fire) 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, fire seals are designed to confine a fire within an area for a time to allow for mitigating actions. A degraded fire seal would not assure this confinement function would be met for the designed and expected duration. The inspectors determined that the finding was of very low safety significance (Green) in accordance with IMC 0609, Appendix F, "Fire Protection Significance Determination Process." The inspectors identified that this issue had a cross-cutting aspect in the Problem Identification and Resolution area because licensee personnel failed to identify and therefore assess this issue completely, accurately, and in a timely manner within the station's CAP (P.1(a)). (Section 1R05.1)

- Severity Level IV: A Severity Level IV NCV of 10 CFR 50.73(a)(2)(v) was identified by the inspectors when licensee personnel failed to report known conditions that could have prevented the fulfillment of the Residual Heat Removal (RHR) system to perform its designed emergency core cooling safety function while operating in the shutdown cooling mode of operation, within 60 days of discovery. Specifically, upon receipt of Westinghouse Nuclear Safety Advisory Letter (NSAL) 0904, "Presence of Vapor in Emergency Core Cooling System/Residual Heat Removal System in Modes 3 or 4 Loss-of-Coolant Accident Conditions," the licensee determined that a loss of RHR system safety function occurred when both trains of the RHR system were placed into the shutdown cooling mode of operation above 200 degrees

Fahrenheit (°F). The station identified four instances in which both trains of RHR were operated in the shutdown cooling mode of operation above 200°F over the previous 3 year period. The licensee, however, failed to report to the NRC within 60 days that the RHR safety function had been lost. The station entered this issue into the CAP as IR 1155372. Corrective actions included the issuance of Licensee Event Report (LER) 05000456/457/2010-007-00 on January 18, 2010.

The inspectors determined that the failure to report this LER in accordance with NRC regulations was a performance deficiency since this issue had the potential to impact the regulatory process. Therefore, this violation was dispositioned through the traditional enforcement process. The inspectors determined that this issue was a Severity Level IV violation based on a similar example referenced in NRC Enforcement Policy Supplement I, Example D.4. The inspectors evaluated this issue under the Reactor Oversight Process (ROP) and did not identify a performance deficiency that could be assessed under the SDP. (Section 4OA2.2)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 operated at or near full power until October 4 when the unit was shut down to commence a scheduled refueling outage. On November 3, the reactor became critical and the main generator was placed online on November 6. The unit achieved full power operation on November 12. The unit operated at or near full power for the remainder of the inspection period.

Unit 2 operated at or near full power for the duration of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors performed a detailed review of the licensee's procedures and preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. 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. Cold weather protection, such as heat tracing and area heaters, was verified to be in operation where applicable and required. 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. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors' reviews focused specifically on the following plant systems due to their risk significance or susceptibility to cold weather issues:

- Cooling Water Intake Structure and Frazil Icing Condition Susceptibility Review;
- Condensate Storage Water Tanks; and
- Refueling Water Storage Tanks (RWSTs).

This inspection constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings of significance were identified.

.2 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the UFSAR for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined whether barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also walked down underground bunkers/manholes subject to flooding that contained multiple trains or multiple function risk-significant cables. The inspectors also reviewed the abnormal operating procedure for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding sample as defined in IP 71111.01-05.

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 “B” Auxiliary Feedwater System;
- Unit 1 Reactor Core System Level Instrumentation Systems During Reduced Inventory Conditions;
- Unit 1 Spent Fuel Pool Cooling System; and
- Unit 1 Essential Service Water 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 to this report.

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

b. Findings

No findings of significance were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

On October 7, 2010, the inspectors performed a complete system alignment inspection of the Unit 1 residual heat removal (RHR) system during a period of Unit 1 reduced inventory operations to verify that the system was both functionally available and operable. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance 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:

- Fire Zone 5.5-2, Unit 2 Auxiliary Electrical Equipment Room and Main Control Room;

- Fire Zone 11.2A-1, Unit 1 “A” RHR Pump Room;
- Fire Zone 11.2D-1, Unit 1 “B” RHR Pump Room;
- Fire Zone 11.6-2, Unit 2 Division 22 Auxiliary Building (AB) Elevation 426’ Electrical Penetration Area;
- Fire Zone 11.36-2, Unit 2 AB Elevation 364’ General Area;
- Fire Zone 1.1-1, Unit 1 Containment Missile Barrier Area Elevation 377’; and
- Fire Zone 12.1-1, Common Fuel Handling Building.

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, 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 minor issues identified during the inspection were entered into the licensee’s CAP. Documents reviewed are listed in the Attachment to this report.

These activities constitute seven quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

Degraded Fire Seal Between Two Fire Zones

Introduction: A finding of very low safety significance (Green) and an associated Non-Cited Violation (NCV) of License Condition 2.E was identified by the inspectors when licensee personnel failed to maintain a fire seal between Unit 2 Fire Zone 11.6-2 on the 426′ elevation and Unit 2 Fire Zone 11.5A-2 on the 414′ elevation of the AB and adjacent to the containment structure in accordance with the approved Fire Protection Program. This area of the plant is commonly referred to as the Unit 2 Electrical Penetration Area.

Description: On October 14, 2010, the inspectors performed a walk down of selected fire zones, including Fire Zone 11.6-2 on the 426’ elevation, and observed that the seal that filled the seismic gap between the AB and containment, two safety-related structures, was deformed and pulled away from the containment wall. This seal served as a fire barrier as described in Section 2.3.11.45 of the Fire Protection Report. This seal also contributed to the separation between two safety-related divisions of electrical power.

After the inspectors notified the licensee that the seal was degraded, the licensee promptly responded to the area to characterize the deformed seal. The Fire Marshal determined that the seal had separated from the wall for its entire height as evidenced by visible light being seen from the 414′ elevation. This evaluation was documented in IR 1126534. Additionally, the Fire Marshal performed a prompt extent of condition review and determined that a similar, but much less significant, condition existed on Unit 1. The assessment for Unit 1 was documented in IR 1126594. Corrective actions included implementing a compensatory fire watch for Unit 2 until the fire seal was repaired.

<u>Analysis</u>: The inspectors determined that the failure to maintain a fire seal between two fire areas was contrary to the approved Fire Protection Plan and was a performance deficiency. The degraded fire seal issue was determined to be more than minor because it was associated with the Protection Against External Factors (Fire) 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, fire seals are designed to confine a fire within an area for a time to allow for mitigating actions. A degraded fire seal would not assure the confinement function would be met for the designed and expected duration.

In accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 3b, the inspectors determined the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, “Fire Protection Significance Determination Process,” was required.

The inspectors used the Phase 1 Flow Chart located in Appendix F of Inspection Manual Chapter (IMC) 0609 during the evaluation process. The inspectors determined that the degraded seal should be categorized under the heading of Fire Confinement with its associated element; fire barriers that separate one fire area from another. The inspectors conservatively assigned a degradation rating of Moderate. The inspectors took note of the level of degradation of the fire seal and concluded that it was reasonable to expect the barrier would provide a minimum of 20 minutes of protection even in its degraded state. The inspectors took note of the fixed equipment and *in situ* combustibles and concluded that the location of the degraded seal was such that it would not be subjected to direct flame impingement. Therefore, per the Supplemental Screening for Fire Confinement Findings in IMC 0609, Appendix F, this finding was screened as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Problem Identification and Resolution (P.1(a)) because the licensee failed to identify and therefore assess this issue completely, accurately, and in a timely manner commensurate with its safety significance. Specifically, the view of the seal in question was unobstructed and therefore clearly visible to licensee staff that would be in the area for other activities. The degraded condition of the seal surface should have been identified and additional assessment provided by licensee staff. A review of recent CAP documents did not identify any previous actions associated with this seal.

<u>Enforcement</u>: License Condition 2.E requires that the licensee implements and maintains in effect all provisions of the approved Fire Protection Program as described in the UFSAR and as approved through Safety Evaluation Reports, dated December 2008. Braidwood’s Fire Protection Report, Section 2.3.11.50, states that the floor penetrations for Fire Zone 11.6-1 and 11.6-2 are provided with 3-hour rated fire seals.

Contrary to the above, on October 14, 2010, NRC Inspectors identified that a Unit 2 fire seal described in the Fire Protection Report had degraded and had not been previously identified or corrected. Specifically, the licensee failed to maintain a fire seal separating Unit 2 Fire Zone 11.6-2 on the 426' elevation and Fire Zone 11.5A-2 on the 414' elevation such that its 3-hour rating was assured. Because this violation was of very low safety significance and was entered into the licensee's CAP as IR 1126534, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000457/2010005-01; Degraded Fire Seal Between Two Fire Zones)**

1R06 Flooding (71111.06)

.1 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors observed if the cables were submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the CAP to verify the adequacy of the corrective actions. The inspectors performed a review of the following underground bunkers/manholes subject to flooding:

- Cable vaults 2E, 2F, 2H, and 2H

This inspection constituted one underground vaults sample as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing for the Unit 1 "A" safety injection pump cubicle cooler heat exchangers 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. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08P)

From October 6 through October 15, 2010, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection Program for monitoring degradation of the Unit 1 reactor coolant system, 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, 1R08.4, and 1R08.5 below count as one inspection sample as defined by IP 71111.08–05. Documents reviewed are listed in the Attachment.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors observed the following nondestructive examinations required by the American Society of Mechanical Engineers (ASME), Section XI, Code and/or 10 CFR 50.55a, to evaluate compliance with the ASME Code Section XI applicable ASME Code Case and Section V requirements and if any indications were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- Ultrasonic Examination of the A Main Steam Isolation Valve Weld (1MS-04-34); Report No. A1R15-UT-006;
- Ultrasonic Examination of the A Main Steam Isolation Valve Weld (1MS-04-35), Report No. A1R15-UT-007;
- Ultrasonic Examination of the A Main Steam Isolation Valve Weld (1MS-04-36), Report No. A1R15-UT-008;
- Ultrasonic Examination of the A Main Steam Isolation Valve Weld (1MS-04-37), Report No. A1R15-UT-009;
- Ultrasonic Examination of the A Main Steam Isolation Valve Weld (1MS-04-38), Report No. A1R15-UT-010; and
- Bare Metal Visual Examination of the Reactor Pressure Vessel Bottom Head penetrations (Under Vessel Exam), Work Order Surveillance No. 01243265.

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC approved alternative.

- Indication Assessment of Main Steam Welded Attachment (Lug) (1MS-07-SW08), and
- Indication Assessment of Main Steam Welded Attachments (Lugs) (1MS-05-PG1, 2, 3, and 4).

The inspectors reviewed the following pressure boundary welds completed for risk significant systems during the Unit 1 refueling outage to determine if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by the construction code, and a NRC approved Code Case N-416. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedures were qualified in accordance with the requirements of construction code and the ASME Code Section IX.

- Welds (FW-6.1, 7.1, and 8.1) Fabricated during Replacement of Kerotest Valve with new Anchor Darling Valve on Line 1RC8042C (WO 01067695).

b. Findings

No findings of significance were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the Unit 1 vessel head, no examination was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D). However, the licensee had previously committed to perform a bare metal visual examination of vessel head Penetration 74 pursuant to an exemption request to NRC Order EA-03-009 (reference NRC approval letter dated September 26, 2007, ADAMS Accession No. ML0724304520).

Therefore, the inspectors reviewed records (WO ATI 560371-03, Visual Examination Non-Destructive Examination Report, October 9, 2010) of the visual examination conducted on penetration 74 to determine if the activities were performed in accordance with the licensee’s commitments to NRC Order EA–03–009, and if any indications were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement. The inspectors also reviewed the vessel head visual examination procedure (ER-AP-335-001, Revision 1) to determine if the procedure incorporated the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D).

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

On October 3, 2010, the inspectors observed the licensee staff performing visual examinations of the Unit 1 reactor coolant system and emergency core cooling system (ECCS) within containment to determine if these visual examinations focused on locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of reactor core cooling system components with boric acid deposits to determine if degraded components were documented in the corrective action system. The inspectors also evaluated corrective actions for any degraded reactor coolant system components to determine if they met the ASME Section XI Code.

- IR 01120716, Dry Boric Acid Residue on 1A RHR Pump Seal at Cooler Lines;
- IR 01120762, Dried Boric Acid Residue at 2B RHR HX Bolts No. 1, No. 9 and No. 48; and
- IR 01120765, Clean Minor Boric Acid Residue/Rust from Bolting on 2A RHR heat exchanger.

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:

- Boric Acid on Valve Stem - 1CV8369B (IR 01085250); and
- 1SI8889C (Boric Acid at Packing) Clean A1R15 (IR 01104389)

b. Findings

No findings of significance were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

For the Unit 1 steam generators, no examination was required pursuant to the TSs during this refueling outage. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of inservice inspection/steam generator 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 inservice inspection/steam generator 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 inservice inspection and pressure boundary integrity.

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

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On November 16, 2010, the inspectors observed a crew of licensed operators in the plant's 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 to this report.

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

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results (71111.11B)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the individual Job Performance Measure operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered in 2010, as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and IP 71111.11, "Licensed Operator Requalification Program." The documents reviewed during this inspection are listed in the Attachment.

Completion of this section constituted one biennial licensed operator requalification inspection sample as defined in IP 71111.11B.

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:

- Station Annunciator System;
- Unit 2, "B" Train Essential Service Water System; and
- Unit 1, Solid State Protection System Cards.

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 to this report.

In addition to the three systems listed above, the inspectors performed an additional sample by reviewing the licensee's periodic 10 CFR 50.65(a)(3) evaluation to ensure that the licensee was effectively following the rule.

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

b. Findings

No findings of significance 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:

- Planned Unit 1 Refueling Outage (A1R15) Comprehensive Risk Assessment;
- Planned Unit 1 Reactor Coolant System Drain Down and Reduced Inventory Operations Yellow Risk Configuration;
- Planned Unit 1 Reactor Head Reassembly Operation;
- Emergent Work Activity Associated with a Unit 1 RHR System Pressure Boundary Leak; and
- Emergent Work Activity Associated with a Degraded Common Unit 1 and Unit 2 Essential Service Water System Discharge Valve Actuator (0SX165A).

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.

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- IR 1155373, Additional Information Regarding the Single Failure Criterion During a Postulated Steam Generator Tube Rupture Event;
- IR 1132131, Unit 2 “C” Channel Narrow Range Reactor Coolant System Hot Leg Resistance Temperature Detector Failing Low;

- IR 1137603, Unit 1 and Unit 2 Common Degraded Essential Service Water System Valve Actuator (0SX165A);
- IR 1100061, Unit 1 and Unit 2 Common Component Cooling Water System "0" Pump Alignment Concern;
- IR 1136544, Failure of Unit 1 Main Steam Safety System Valve (1MS009D) to Close During the August 16, 2010 Unit 1 Reactor Trip; and
- IR 1128892, Unit 1 "B" Emergency Diesel Generator System Kilowatt Load Difference Between Instrumentation.

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 sampling 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 to this report.

This operability inspection constituted six samples as defined in IP 71111.15-05.

b. <u>Findings</u>

No findings of significance were identified.

1R18 <u>Plant Modifications</u> (71111.18)

.1 <u>Temporary Plant Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed the following temporary modification(s):

- Engineering Change 382328, Unit 1, "A" Channel Reactor Coolant System Hot Leg Resistance Temperature Device Averaging Modification; and
- Engineering Change 381875, Unit 2, "C" Channel Reactor Coolant System Hot Leg Resistance Temperature Device Averaging Modification.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, UFSAR, and TS, as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications were installed as directed; the modifications operated as expected; modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted two temporary modification samples as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The following engineering design package was reviewed and selected aspects were discussed with engineering personnel:

- Vacuum Breakers 2-11 Elimination on Circulating Water Blowdown Line.

This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed completed work activities to verify that installation was consistent with the design control documents. The modification removed ten vacuum breakers that have been a potential source of leakage in past years. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one permanent plant modification samples as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- WO 900870, Unit 1 Reactor Containment Fan Cooler Fan/Motor Shaft and Bearing Work;

- WO 1128337, Unit 1 “D” Reactor Coolant Pump 10-year Inspection/Axial End Play Out of Specification;
- WO 1260575, Unit 1 Safety Injection System High Point Vent Installation;
- WO 1362930, Unit 1 133V 480 Volt Substantiation Transformer Replacement;
- WO 1259438, Unit 1 Solid State Protection System Universal Logic Card 1PA10J Replacement;
- WO 1294196, Unit 1 “B” Auxiliary Feedwater Pump Rocker Cover Replacement;
- WO 1217999, Unit 1 Containment Spray System Check Valve 1CS008 Disassembly and Inspection, and Reinstallation;
- WO 1243734, Unit 1 Essential Service Water System Valve 1SX174 Disassembly, Inspection, and Reinstallation; and
- WO 0347481, Unit 1 “A” Steam Generator System Upper Lateral Support Shim Package Bolting Repair.

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 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 to this report.

This inspection constituted nine post-maintenance testing sample as defined in IP 71111.19-05.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the Unit 1 refueling outage (RFO), conducted October 4 through November 3, 2010, 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 RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense in-depth commensurate with the OSP 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 OSP requirements were met, and controls over switchyard activities.
- 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 the TSs.
- Refueling activities.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of primary containment to verify that debris had not been left which could block Emergency Core Cooling System (ECCS) suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

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

b. Findings

No findings of significance 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:

- Unit 1, 18-Month Residual Heat Removal System Valve Stroke Surveillance (Inservice Testing);
- Unit 2, Quarterly “A” ASME Charging Pump and Discharge Check Valve Surveillance (Inservice Testing);
- Unit 2, Quarterly “B” ASME Charging Pump and Discharge Check Valve Surveillance (Inservice Testing);

- Unit 1, Reactor Coolant System Unidentified Leak Rate Determination Surveillance (Reactor Coolant System Leakage); and
- Unit 1, 18-Month Chemical and Volume Control System Containment Isolation Valve Stroke Surveillance (Isolation Valve).

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 were 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
- problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three inservice testing samples, one reactor coolant system leak detection inspection sample, and one containment isolation valve sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

.1 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

Since the last NRC inspection of this program area, emergency action level and Emergency Plan changes were implemented based on the licensee's determination, in accordance with 10 CFR 50.54(q), that the changes resulted in no decrease in effectiveness of the Plan, and that the revised Plan as changed continues to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50. Revisions to the emergency action levels and Emergency Plan were reviewed by the inspectors in the Exelon Nuclear Radiological Emergency Plan Annex for Braidwood Station, Revisions 23, 24, and 25. The inspectors conducted a sampling review of the Emergency Plan changes and a review of the emergency action level changes to evaluate for potential decreases in effectiveness of the Plan. However, this review does not constitute formal NRC approval of the changes. Therefore, these changes remain subject to future NRC inspection in their entirety. This emergency action level and emergency plan changes inspection constituted one sample as defined in IP 71114.04-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted one complete sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators for the occupational exposure cornerstone for followup. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings of significance were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

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

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

- Unit 1 Refuel Floor;
- Unit 1 Containment Activities; and
- AB 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 of significance were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

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

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 Component Disassembly and Reassembly including Reactor Head Preparation;
- Fuel Moves and Trinuclear Work Activities; and
- All Incore Sump Entries and Inspections.

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

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the 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 of significance were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

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

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

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

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

b. Findings

No findings of significance were identified.

.5 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 is properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

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

- Reactor Head Component Disassembly and Reassembly including Reactor Head Preparation;
- Fuel Moves and Trinuclear Work Activities; and
- All Incore Sump Entries and Inspections.

For these radiation work permits, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, and entry 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 licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

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

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

b. Findings

No findings of significance were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

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

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings of significance were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings of significance were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control 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 that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings of significance 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 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's three year rolling average collective exposure.

The inspectors reviewed the site--specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power Research Institute TR-108737, "BWR Iron Control Monitoring Interim Report," issued December 1998, and/or plant historical data, when available).

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures As-Low-As-Is-Reasonably-Achievable (ALARA), which included a review of processes used to estimate and track exposures from specific work activities.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance.

- Reactor Head Component Disassembly and Reassembly including Reactor Head Preparation;
- Diving Operation in Reactor Cavity;
- Fuel Moves and Trinuclear Work Activities; and
- All Incore Sump Entries and Inspections.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated whether the licensee's ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee's work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and radiation work permit documents.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee's ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the radiation protection group with the actual work activity time requirements, and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity, failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated whether the licensee had established measures to track, trend, and if necessary to reduce, occupational doses for ongoing work activities. The inspectors assessed whether trigger points or criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

The inspectors evaluated the licensee's method of adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered. The inspectors assessed whether adjustments to exposure estimates (intended dose) were based on sound radiation protection and ALARA principles or if they were just adjusted to account for failures to control the work. The inspectors evaluated whether the frequency of these adjustments called into question the adequacy of the original ALARA planning process.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers are familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedure compliance issues (e.g., workers are not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Barrier Integrity

.1 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the Braidwood Unit 1 and Unit 2 reactor coolant system leakage performance indicator for the period from the third quarter 2009 to the third quarter 2010 to determine the accuracy of the performance indicator data reported during those periods. Performance indicator definitions and guidance contained in the Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator logs, reactor coolant system leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of July 1, 2009, through September 30, 2010, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance 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 CAP

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. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

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 of significance were identified.

.2 Daily CAP 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

Failure to Submit Licensee Event Report Per 10 CFR 50.73(a)(2)(v)

Introduction: A Severity Level IV NCV of 10 CFR 50.73(a)(2)(v) was identified by the inspectors when licensee personnel failed to report an event or condition that could have prevented the fulfillment of the RHR system to perform its designed emergency core cooling safety function while operating in the shutdown cooling mode of operation.

Description: On May 26, 2010, the licensee completed a formal operating experience review regarding information in Westinghouse Nuclear Safety Advisory Letter (NSAL) 09-08, "Presence of Vapor in Emergency Core Cooling System/Residual Heat Removal System in Modes 3 or 4 Loss-of-Coolant Accident Conditions." The licensee documented their technical review in Engineering Change 379707. This evaluation concluded that the 260°F temperature limit that allowed the RHR system to operate in the shutdown cooling mode of operation was acceptable if the system was called upon to swap over to the emergency core coolant injection mode of operation when aligned to the RWST. However, when the RHR system was swapped from the injection mode of operation to the recirculation mode of operation (via the containment sump), flashing of liquids in the hot leg suction lines could occur and render both trains of RHR inoperable. This potential adverse consequence existed due to the elevation differences between the RHR piping at the containment penetrations and the expected containment sump level, combined with the postulated containment pressure at the established 260°F limit. The evaluation concluded that the Reactor Coolant System (RCS) temperature must be reduced to 200°F prior to placing both RHR trains in the shutdown cooling mode of operation in order to eliminate the potential for flashing of water within the isolated hot leg suction piping during transfer to the containment sump.

Based on the findings of the evaluation, the licensee implemented changes to the associated operating and emergency procedures to reflect the more restrictive 200°F RCS temperature limit.

The licensee also conducted a past operability review to determine if Braidwood had operated with both RHR trains in the shutdown cooling mode above 200°F (Issue Report (IR) 1073616, Assignment 4), which represented a loss of RHR function. The licensee's review revealed four occurrences. Specifically:

- October 12, 2009, during refueling outage A2R14, for 5 minutes before reaching Mode 5;
- March 30, 2009, during refueling outage A1R14, for 17 minutes before reaching Mode 5;
- April 21, 2008, during refueling outage A2R13, for 20 minutes before reaching Mode 5;
- October 1, 2007, during refueling outage A1R13, for 28 minutes before reaching Mode 5.

The licensee completed this evaluation on July 12, 2010. As of September 12, 2010, the licensee had not submitted a Licensee Event Report (LER). In accordance with 10 CFR 50.73(a)(2)(v), a licensee must report any event or condition that could have prevented the fulfillment of the safety function of structures within 60 days after the discovery of a reportable event.

Analysis: The inspectors determined that the failure to report the conditions which could have prevented the fulfillment of the RHR emergency core cooling function in accordance with 10 CFR 50.73(a)(2)(v) was a performance deficiency. This violation had the potential to impact the regulatory process based upon the generic communication that LERs serve, the required Reactor Oversight Process reviews that the NRC performs on all LERs, and the potential for these event(s) to impact the station ROP performance indicator metrics. Therefore, this violation was dispositioned through the traditional enforcement process.

The inspectors also evaluated this issue under the ROP. In this case, the licensee effectively managed the operating experience from Westinghouse and revised the station's procedures to address the concern. Therefore, the inspectors did not identify a performance deficiency that could be assessed under the Significance Determination Process (SDP).

The inspectors determined that this issue was a Severity Level IV violation based on a similar example referenced in the NRC Enforcement Policy. Specifically, Supplement I, Example D.4, stated that "a failure to make a required LER is categorized as a Severity Level IV violation."

ROP cross-cutting aspects do not apply to traditional enforcement issues, therefore, none was identified.

Enforcement: 10 CFR 50.73(a)(2)(v) requires, in part, that a licensee report any event or condition that could have prevented the fulfillment of the safety function of structures, systems, and components that are needed to (A) Shutdown the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material; or (D) Mitigate the consequences of an accident, within 60 days of event discovery. Contrary to this requirement, as of September 12, 2010, the licensee had not reported a condition that could have prevented the fulfillment of both the Unit 1 and Unit 2 RHR system to remove decay heat during a design basis event, which was identified on July 12, 2010. This exceeded the 60-day reporting requirement. Because this violation was not repetitive or willful, and was entered into the licensee's CAP, it is being treated as a Severity Level IV NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. Corrective actions included the issuance of LER 05000456(457)/2010-007-00 on January 18, 2011. **(NCV 05000456/2010005-02, 05000457/2010005-02, Failure to Submit Licensee Event Report per 10 CFR 50.73(a)(2)(v))**

.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 July 1, 2010 through December 31, 2010, 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 reworks maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule Program 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 a single semiannual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings of significance 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 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 operator workarounds. 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 operator workarounds.

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

b. Findings

No findings of significance were identified.

.5 Annual Sample: Conservative Decision Making Assessment

a. Inspection Scope

The inspectors performed this sample to review and evaluate the licensee's safety culture involving the station's ability to make conservative decisions at an acceptable level. This sample was performed to provide specific insight in assessing if the licensee had demonstrated sustained improvement following the 2009 reactor oversight process end-of-cycle assessment and subsequent 4OA2 sample performed mid-year of 2010.

The inspectors routinely observed decisions made at the station at various levels of management and supervision as well as across multiple disciplines (e.g. Operations, Maintenance, and Engineering). These observations included issues in which a particular decision was made through a planned process and issues that were emergent in nature. The inspectors focused on decisions that had the potential to impact safety and adversely impact the station's current licensing basis. Additionally, the inspectors routinely observed 'softer' aspects of this safety culture element by observing if the station was routinely committed to ensuring aspects of a particular issue were thoroughly understood to ensure an appropriate decision could be made.

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

b. Findings and Observations

No findings of significance were identified.

The inspectors observed several recent noteworthy examples in which the licensee made significant decisions commensurate with the safety significance of the particular instance and with regards to the station's current licensing basis. The inspectors noted that the licensee's ability to make conservative decisions at an appropriate level had been more integrated into the station's safety culture than previously observed and documented in the 2010 second quarterly inspection report (05000456/457)/2010003). Although these samples represented an extremely small population of the number of decisions implemented at the station on an annual basis, the inspectors observed these samples to be representative of the standard to which the licensee had set to maintain. These types of examples provided the inspectors with continued confidence regarding the licensee's ability to continue to improve with regards to this aspect of safety culture as demonstrated by a reduction in documented findings sharing this cross-cutting aspect and by routine observations.

- On October 7, 2010, following the Unit 1 "D" reactor coolant pump 10-year planned overhaul, the licensee performed a motor 'bump' test as part of the post maintenance testing activities. During the bump test the licensee identified that the breakaway torque was significantly higher than expected (350 ft-lbs vice 90 ft-lbs). Although this test indicated a higher than normal torque to drive the motor, it did not represent a testing failure since the tested value was below the acceptance criteria of 750 ft-lbs. The inspectors observed that the licensee implement a conservative approach to understand the reason behind the increase torque and logically worked through a troubleshooting plan to determine the cause of the higher than anticipated value before placing the pump back in service. On October 28, 2010, the licensee identified the cause - foreign material on the upper thrust bearing assembly.

- On October 28, 2010, the licensee identified that the one of the three Unit 2 "C" loop resistance temperature detector's output dropped unexpectedly in response to a small change in containment temperature. This particular resistance temperature detector had previously been identified slowly degrading based on trending data over the last several months. The licensee performed the TS channel check surveillance by comparing it to the other channels. This surveillance met the acceptance criterion; however, the licensee conservatively declared the channel inoperable based on its known degradation and unexpected and unanticipated interaction with containment temperature.

- During the Unit 1 refueling outage, the licensee identified an indication on the reactor vessel flange. This indication was characterized as a small surface indication with a maximum depth of approximately 0.003 inches. The licensee engaged the vendor and determined that this indication did not exceed the vendor's acceptance criteria that would require an immediate repair. The licensee made a conservative decision to repair the flaw. Additionally, the licensee's repair method consisted of performing the repair underwater by honing out the indication. This choice eliminated the necessity to perform an additional Reactor Coolant System drain down and reduced inventory operation. Additionally, this choice minimized the dose received by the workers.

.6 Annual Sample: Unit 1 Reactor Head Vent Corrective Actions

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized corrective actions item documenting a longstanding degraded issue with the Unit 1 reactor head vents. The station's reactor head vents have had issues regarding operation and indication since 1994. The nature of a number of the issues previously identified involved valve position indications issues caused by externally mounted reed switches. Due to the design and desire to rely on proper valve position indication for these types of valves, the valve position indications issues have led to the unavailability of a reactor head vent train(s) for extending periods of operations.

The inspectors focused on recent licensee performance associated with addressing the underlying valve position indications issues.

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

b. Findings

No findings of significance were identified. The station has implemented a modification designed to improve the reliability of this system's valve position indication. This modification was implemented during the Unit 1 Fall refueling outage and both Unit 1 trains have been available to date. A similar repair was planned for the Unit 2 "A" Reactor Head Vent System train during the 2011 Spring refueling outage.

.7 Annual Sample: Beacon Core Monitoring System Non-Conservative Model Corrective Actions

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting three errors identified in the Bryon and Braidwood Beacon Core Monitoring System. These errors had been communicated by a vendor to the licensee and were associated with instances in which a method or assumptions utilized in calculating core physics parameters were either not conservative or were different than the methodology approved in the current licensing basis. A summary of the errors identified included:

- The station utilizing a methodology different than the methodology approved in the station's current licensee basis to calculate reactor coolant temperature thermocouple measurement.
- The method for which the station measures the reactor core thermocouple's uncertainty was not consistent with the station's current licensee's basis uncertainty analysis utilized in the departure from nucleate boiling analysis.
- A value of Reference Peak Power that was edited in the TS summary section of the flux map output might not be the correct value from which to determine the appropriate response to the surveillance requirements.

The inspectors reviewed the licensee's corrective actions for the issues identified to verify whether: (1) the problems were accurately identified; (2) the causes were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) previous occurrences were considered; and (5) corrective actions proposed/implemented were appropriately focused to address the problems and were commensurate with the safety significance of the issues. Additionally, the inspector reviewed the licensee's evaluation that determined that Braidwood had not operated outside of its accident analysis or violated the TSs with respect to these issues.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings of significance were identified.

.8 Annual Sample: Multiple Reactor Coolant Temperature Detector Failures

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized two occurrences in which reactor coolant temperature thermocouples had failed during the operating cycle.

- Unit 2, "C" channel reactor coolant system resistance temperature detector failed low, Issue Report 1132131; and
- Unit 1, "A" channel reactor coolant system resistance temperature detector failed low, Issue Report 1145482.

The inspectors reviewed the licensee's corrective actions for the issues identified to verify whether: (1) the problems were accurately identified; (2) the causes were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) previous occurrences were considered; and (5) corrective actions proposed/implemented were appropriately focused to address the problems and were commensurate with the safety significance of the issues.

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

b. Findings

No findings of significance were identified.

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

.1 (Closed) Unresolved Item 05000456/2010003-01; 05000457/2010003-01: Degraded Condition of Reactor Head Vent Valves

a. Inspection Scope

This Unresolved Item (URI) was identified in IR 05000456/457/2010003 based on an inspector review of the condition of the reactor vessel head vents. The inspectors reviewed IR 915503, which documented a long-standing degraded condition of the reactor head vent valves. The inspectors noted that these systems had significant periods of inoperability since 1998.

The reactor head vent system consists of two redundant flow paths, each with two solenoid valves in series. The reactor head vents are safety-related components that are utilized in plant emergency operating procedures and severe accident management guidance documents to mitigate the consequences of an accident. In the event the valves fail to meet the surveillance acceptance criteria, the Technical Requirements Manual allows one flow path to be isolated and de-energized if the remaining flow path is operable and available. The inspectors reviewed the performance history of the reactor head vents within the licensee's Maintenance Rule database and had several questions regarding the availability and reliability of the reactor head vents. Specifically, the inspectors had questions regarding the scoping and monitoring of the reactor head vent valves by the licensee's Maintenance Rule program. The inspectors questioned if the licensee should be monitoring both reliability and availability/unavailability consistent with a recommendation in NUMARC 93-01. Additionally the inspectors questioned the technical justification for the established reliability criteria and the effectiveness of monitoring the reactor vessel head vents at the component level.

b. Findings

No findings of significance were identified.

After discussions with NRR staff and Region III subject matter experts and additional review of NUMARC 93-01, the inspectors concluded that there were no violations of NRC requirements. This URI is closed.

.2 (Closed) Licensee Event Report 05000456/2010-001-00: Reactor Trip Due to Water Intrusion in Breakers Causing Circulating Water Pump Trips and Resulting in Loss of Condenser Vacuum

On August 16, 2010, as a result of a Unit 2 reactor trip, condensate water overflowed from the Unit 2 auxiliary feedwater standpipe onto the turbine deck. The water spread through openings in the floor to the elevation below, and entered a Unit 1 nonsafety-related substation cabinet. This resulted in the trip of electrical breakers that caused the Unit 1 "A" and "C" circulating water pumps to trip. Additionally, power to the respective circulating water pump discharge valves was lost. These conditions caused the Unit 1 condenser vacuum to degrade. At 2:19 a.m., the Unit 1 main turbine automatically tripped based on a low vacuum condition in the main condenser. The main turbine trip caused an automatic reactor trip.

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations. In addition, the inspectors reviewed the response of personnel following the reactor trip and the root cause evaluation. Documents reviewed are listed in the Attachment to this report.

A licensee identified violation of TS 3.3.9, was documented in Section 4OA7 of NRC Special Inspection Team Report 05000456/2010010; 05000457/2010010. This report documented one Green finding related to the cause of the reactor trip and three Green findings associated with equipment issues following the trip.

This LER is closed. This event follow-up review constituted one sample as defined in IP 71153-05.

.3 <u>(Closed) Licensee Event Report 05000456/2010-002-00: Limiting Condition of Operation Action Not Completed Within the Required Time</u>

On August 16, 2010, at 2:19 a.m., Unit 1 experienced an automatic reactor trip due to an automatic turbine trip on low condenser vacuum. The low condenser vacuum was the result of the Unit 1 "A" and "C" circulating water pumps tripping due to water intrusion into a nonsafety-related cabinet.

Following the Unit 1 trip, at 2:41 a.m., a volume control tank (VCT) high level alarm was received, which caused a loss of the boron dilution protection system (BDPS) function. This condition met the entry criteria into TS Limiting Condition of Operation (LCO) 3.3.9 "Boron Dilution Protection System," Condition A, "One Boron Dilution Alert Channel Inoperable," and Condition C, "Two Boron Dilution Alert Channel Inoperable." Since the entry conditions to TS LCO 3.3.9 were not recognized, the 1-hour TS required actions for Condition C to close non-borated water source isolation valves and to verify shutdown margin was within its limit were not completed within the required completion time, which in turn met the entry condition to TS LCO 3.0.3.

At 5:40 a.m., during a review of the main control board indications, the operators identified that entry into TS LCO 3.3.9 was required. The unit operator was instructed to lower VCT level to clear the high level alarm and restore the BDPS to an operable status. This was accomplished at 5:45 a.m. The licensee reported this condition under 10 CFR 50.72(a)(2)(i)(B), "Condition Prohibited by Plant's Technical Specification."

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations. In addition, the inspectors reviewed the response of licensee personnel following the discovery of the missed TS LCO entry and the root cause evaluation. Corrective actions to prevent recurrence included revising the reactor trip response procedure to check the VCT level greater than 70 percent and initiate entry into TS LCO 3.3.9. Documents reviewed are listed in the Attachment to this report. This LER is closed.

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

A licensee identified violation of TS 3.3.9, was documented in Section 4OA7 of NRC Special Inspection Team Report 05000456/2010010; 05000457/2010010 for this issue.

.4 <u>(Closed) Licensee Event Report 05000457/2010-002-00: Containment Spray Pump Suction Valve Failed to Close Resulting in Unanalyzed Plant Condition Due to Procedure Error</u>

On June 17, 2010, Unit 2 containment spray (CS) pump suction valve 2CS009B failed to close during a quarterly surveillance test. Upon evaluation of the potential consequences of this failure, the licensee identified and reported an 8-hour non-emergency unanalyzed condition that significantly degraded plant safety per 10 CFR 50.72(b)(3)(ii)(B). The licensee manually restored the valve to the normally closed position, eliminating the condition.

During a loss-of-coolant accident, both the emergency core cooling and CS systems utilize the RWST as a water source. Upon the receipt of a RWST lo-2 level, both automatic and manual actions are performed to align the emergency core cooling system (ECCS) water source from the RWST to the containment sump. Upon a RWST lo-3 level, the plant licensing basis assumes that the CS system is manually aligned from the RWST to the containment sump by opening CS009A and CS009B valves and isolating the RWST by shutting the CS001B valve.

The license identified that during the performance of the surveillance, if the CS009A/B valves failed to shut (as on June 17, 2010, or from an assumed license basis active single failure) and if a safety injection had occurred, that upon the receipt of a RWST lo-2 level signal, a portion of the RWST would drain back into the containment sump via the open CS009A/B valve. Until the ECCS swap over was completed, an unanalyzed amount of RWST borated water would have drained into the containment sump and would not have been available for injection and assumed accident boration. This condition also applied to Unit 1.

The licensee reported this unanalyzed plant condition under 10 CFR 50.73(a)(2)(ii)(B). The licensee determined that the root cause of this condition stemmed from inadequate 10 CFR 50.59 reviews since the plant was originally licensed and inadequate procedures created and implemented in 1988. Upon licensing, additional revisions to the surveillance procedure did not identify the unanalyzed condition.

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations. The licensee's review determined that the station did not have an adequate procedure for performing this surveillance activity and implemented changes to prevent recurrence. The inspectors identified that this issue was a performance deficiency that had been identified by the licensee and entered into the station's CAP. The enforcement aspects of this finding are discussed in Section 4OA7 of this report. This LER is closed.

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

.5 (Closed) Licensee Event Report 05000457/2010-003-00, Reactor Trip Caused by Phase to Ground Fault of a Failed Crossover Damper/Deionizer Assembly due to an Inadequate Inspection Acceptance Criteria and Preventive Maintenance Inspection Frequency

a. Inspection Scope

On August 16, 2010, at 2:06 a.m., the Unit 2 main generator received a generator lockout relay trip, which caused an automatic Unit 2 reactor trip on a turbine trip above 30 percent power. Following the reactor trip, the auxiliary feedwater pumps started and the steams dumps opened to control pressure, as expected for a reactor trip from full power. All systems operated as expected with the exception of the flow control valve to the 2D steam generator which failed open. The operators responded to this by taking manual control of an isolation valve to control flow to the steam generator.

The root cause of this event was determined to be a deionizer fin section of the damper assembly that had detached and caused the Unit 2 isophase bus duct ground. The inspection of the assemblies identified excessive wear and degraded flow damper blades.

The inspectors reviewed this LER and determined that it was completed in accordance with NRC regulations. In addition, the inspectors reviewed the response of personnel following the reactor trip and the root cause evaluation. Documents reviewed as part of this inspection are listed in the Attachment to the report. This LER is closed.

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

b. Findings

Inadequate Evaluation of Operating Experience Contributes to a Unit 2 Reactor Trip

Introduction: A finding of very low safety significance (Green) was identified by the inspectors when licensee personnel failed to adequately utilize operating experience that ultimately contributed to an August 16, 2010, Unit 2 reactor trip. Specifically, the licensee did not properly evaluate operating experience as documented in IR 259836, "OPEX Review: Isophase Bus Ground Faults." A portion of this document emphasized the need to consider re-evaluating the associated preventative maintenance frequency for deionizer grids, louvers, and dampers if the isophase air flow through these devices had been raised since the last inspection. The station had occasionally raised air flow since 2002, but no actions were taken to address this change.

Description: Braidwood Station uses an isophase bus to carry large currents between the generator and its step-up transformer. The current from each of the three-phases ("A", "B" and "C") is carried on a separate conductor, which is enclosed in a separate grounded metal housing. If a fault occurs in one of these phases, it is detected by a protective relay, which signals the main generator to trip in order to interrupt the fault.

On August 16, 2010, the Unit 2 main generator received a generator lockout relay trip due to a ground fault. This condition led to a Unit 2 turbine trip and reactor trip per the design of the plant. The root cause of this event was a deionizer fin section of the damper assembly that had detached and caused a Unit 2 isophase bus duct ground.

The licensee determined that an increase in the degradation of the failed damper was due to increased turbulent flow that the dampers had been subjected to. Since 2002, the licensee had changed the manner in which the bus duct cooling fans were operated. Due to higher lake temperatures and the increased generator output following a power uprate, the licensee had occasionally raised the flow though the bus ducts by implementing two-fan operation. The licensee's root cause analysis stated that although it appeared that the fans were not run in this configuration for long periods of time, they recognized that not all instances of two-fan operation were recorded in the operations logs. The licensee identified that the inspection frequency and criteria did not consider the fact that the fans were sometimes operated in parallel with the increased flow.

As documented in the root cause analysis, there were numerous missed opportunities in which the inadequate inspection frequency could have been discovered. The inspector determined that the most recent reasonable opportunity to recognize and correct the inadequate preventative maintenance frequency was when the station created IR 259836 to evaluate operating experience of isophase bus ground faults in October 2004. The operating experience was related to four nuclear plants that experienced reactor trips due to ground faults in their isophase bus ducts. The IR documented that increases in bus duct flow could affect components such as crossover dampers and specifically stated to consider re-evaluating the inspection criteria if flow had been increased.

The operating experience evaluation was completed in February 2005 using procedure LS-AA-115, "Operating Experience Program". The evaluation recommended that no actions be taken since the inspection criteria and frequency was in accordance with the fleet boilerplate. In the evaluation, the licensee stated that failure mechanisms for damper and fans were evaluated and that the current preventive maintenance activity was adequate. The inspectors determined that this response did not address the basis for the inspection frequencies, as specified in Procedure LS-AA-115, Attachment 1, "OPEX Reviewer's Guidelines," for evaluating operating experience. The licensee also did not address that the flows had been raised since the previous damper inspection as specified in the evaluation guideline. The incomplete responses resulted in an inadequate operating experience evaluation and consequently the licensee missed an opportunity to increase the inspection frequency as recommended by the operating experience.

<u>Analysis:</u> The inspectors determined that the failure to adequately evaluate readily available industry operating experience in accordance with Station Procedure LS-AA-115, "Operating Experience Program," was a performance deficiency. Specifically, the station concluded a portion of operating experience was not applicable to Braidwood station even though air flow through the dampers had been raised occasionally since 2002 and no actions to re-evaluate the preventive maintenance frequency were taken.

The finding was determined to be more than minor because it was associated with the Procedure Adequacy 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 performance deficiency contributed to the cause of the August 16, 2010, Unit 2 reactor trip. The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events Cornerstone. The finding screened as having very low safety significance (Green) because it was determined not to contribute to both a plant trip and the likelihood that mitigating system equipment or functions would not be available. The inspectors did not identify a cross-cutting aspect associated with this finding since it was not considered to reflect current performance.

Enforcement: No violations of NRC requirements were identified because the affected components were not safety-related. Because this finding did not involve a violation and has very low safety significance, it is identified as a Finding.

The licensee entered this issue into their CAP as IR 1101855. Corrective actions for this event included revising the preventative maintenance schedule to ensure crossover dampers were inspected and/or replaced prior to failure, with the periodicity to be based upon a thorough engineering analysis. The maintenance procedure for the isophase bus duct was also revised to include inspection criteria for the crossover dampers. **(FIN 05000457/2010005-03, Inadequate Evaluation of Operating Experience Contributes to a Unit 2 Reactor Trip).**

.6 (Closed) Licensee Event Report 05000456/2010-003-00, Through-Weld Leak of the Line from the 1B Seal Injection Filter to the Vent Valve

On September 15, 2010, at 7:58 a.m., Operations was notified that a potential for leakage existed on a welded ¾-inch diameter stainless steel line which extended from the Unit 1 reactor coolant pump seal water injection filter to the "B" seal water injection filter vent valve. The station entered TS LCO 3.5.5, "Seal Injection Flow" based on a postulated failure of this line due to pressure boundary leakage. The LCO was exited at 9:39 a.m. when the "B" seal injection filter, including the potentially leaking vent pipe, was isolated from the Chemical Volume and Control System within the allowed completion time of the LCO. A through-weld leak was later confirmed on one weld through visual inspection.

The cause of the leakage was determined to be transgranular stress corrosion cracking that was accelerated by the existence of an original fabrication flaw.

The inspectors reviewed this LER and determined that it was conservatively completed in accordance with NRC regulations. The inspectors did not identify a performance deficiency associated with this issue. This issue was entered into the station's CAP. Corrective actions include a repair to the through-weld leak. Documents reviewed are listed in the Attachment to this report.

This LER is closed. This event follow-up review constituted one sample as defined in IP 71153-05.

4OA5 Other Activities

.1 Reactor Coolant System Dissimilar Metal Butt Welds (Temporary Instruction 2515/172, Revision 1)

a. Inspection Scope

The inspectors conducted a review of the licensee's activities regarding licensee dissimilar metal butt weld (DMBW) mitigation and inspection implemented in accordance with the industry self-imposed mandatory requirements of Materials Reliability Program MRP-139, "Primary System Piping Butt Weld Inspection and Evaluation Guidelines."

Temporary Instruction (TI) 2515/172, "Reactor Coolant System Dissimilar Metal Butt Welds," was issued to support NRC review and evaluation of the licensees' implementation of MRP-139. The review for Unit 1 DMBWs under Revision 0 and the Draft Revision 1 to TI 2515/172 had been previously completed (reference Braidwood Inspection Reports 05000456/2008003; 05000457/2008003; and 5000456/2009003; 05000457/2009003). From October 6 through October 15, 2010, the inspectors performed a review for the Unit 1 DMBWs in accordance with TI 2515/172, Revision 1, as described below.

b. Observations

Braidwood Unit 1 is a Westinghouse four loop designed plant. The licensee identified a population of DMBWs susceptible to primary water stress corrosion cracking in accordance with MRP-139 guidelines. The licensee had previously completed mitigation by weld overlay repair to the pressurizer DMBWs. The licensee was considering mitigation of the Unit 1 DMBWs located on the reactor coolant loop hot legs using a mechanical stress improvement process in Refueling Outage No. 16 (A1R16).

Based on the schedule of DMBW examinations under MRP-139, no examinations were required for the current Unit 1 refueling outage (A1R15) and hence none were performed. Likewise, mitigation was also not performed during A1R15. Additionally, the licensee had not made any changes to the MRP-139 inspection program since the NRC had previously reviewed this program. Therefore, no additional inspection was necessary for Revision 1 of TI 2515/172. Revision 1 is considered complete.

No findings of significance were identified.

.2 (Closed) Temporary Instruction 2515/179, "Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System Pursuant to Title 10, Code of Federal Regulations, Part 20.2207 (10 CFR 20.2207)"

a. Inspection Scope

The inspectors confirmed that the licensee has reported the initial inventories of sealed sources pursuant to 10 CFR 20.2207 and verified that the National Source Tracking System database correctly reflects the Category 1 and 2 sealed sources in custody of the licensee. The inspectors interviewed personnel and performed the following:

- Reviewed the licensee's source inventory;
- Verified the presence of any Category 1 or 2 sources;
- Reviewed procedures for and evaluated the effectiveness of storage and handling of sources;
- Reviewed documents involving transactions of sources; and
- Reviewed adequacy of licensee maintenance, posting, and labeling of nationally tracked sources.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 5, 2011, the inspectors presented the inspection results to Mr. A. Shahkarami, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the Inservice inspection with Mr. L. Coyle on October 15, 2010.
- The results of the Radiological Hazard Assessment and Exposure Controls inspection and the Occupational ALARA Planning and Controls inspection, and Verification of Licensee Responses to NRC Requirement for Inventories of Materials Tracked in the National Source Tracking System inspection with Mr. L. Coyle on October 22, 2010.
- The results of the Licensed Operator Requalification Training Annual Operating Test inspection with Mr. R. Cameron via telephone on December 7, 2010.
- The results of the Emergency Action Level and Emergency Plan Changes inspection with Mr. R. Gaston via telephone on December 13, 2010.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

4OA7 Licensee-Identified Violations

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

- 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings", requires, in part, that activities affective 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, procedure, or drawings. Contrary to the above, since the original licensing of the plant on May 20, 1988, licensee personnel failed to prescribe an adequate procedure for performing a surveillance stroke test for the Unit 2 CS valve 2CS009B since this condition was not analyzed assuming an active single failure for this valve to close. This finding was determined to be of very low safety significance based on an estimated core damage probability of 9.6 E-8. This issue was entered into the station's CAP that prescribed a cause evaluation, extent of condition review, and actuator rebuild.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Shahkarami, Site Vice President
L. Coyle, Plant Manager
P. Boyle, Maintenance Director
S. Butler, Emergency Preparedness Manager
R. Cameron, Licensed Operator Requalification Training Lead
P. Daly, Radiation Protection Manager
B. Finlay, Security Operations Manager
R. Gadbois, Maintenance Manager
G. Galloway, Work Control Manager
R. Gaston, Regulatory Assurance Manager
M. Marchionda, Operations Manager
R. Radulovich, Nuclear Oversight Manager
T. Schuster, Chemistry/Environmental Manager
M. Smith, Engineering Manager
R. Zuffa, Resident Inspector, Illinois Emergency Management Agency

Nuclear Regulatory Commission

E. Duncan, Chief, Branch 3, Division of Reactor Projects
B. Dickson, Chief, Plant Support Team, Division of Reactor Safety

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000457/2010005-01	NCV	Degraded Fire Seal Between Two Fire Zones (Section 1R05)
05000456/2010005-02; 05000457/2010005-02	NCV	Failure to Submit a Licensee Event Report per 10 CFR 50.73(a)(2)(v) (Section 4OA2.2)
05000457/2010005-03	FIN	Inadequate Evaluation of Operating Experience Contributes to a Unit 2 Reactor Trip (Section 4OA3.5)

Closed

05000457/2010005-01	NCV	Degraded Fire Seal Between Two Fire Zones (Section 1R05)
05000456/2010005-02; 05000457/2010005-02	NCV	Failure to Submit a Licensee Event Report per 10 CFR 50.73(a)(2)(v) (Section 4OA2.2)
05000457/2010005-03	FIN	Inadequate Evaluation of Operating Experience Contributes to a Unit 2 Reactor Trip (Section 4OA3.5)
05000456/2010003-01; 05000457/2010003-01	URI	Degraded Condition of Reactor Head Vent Valves (Section 4OA3.1)

05000456/2010-001-00	LER	Reactor Trip Due to Water Intrusion in Breakers Causing Circulating Water Pump Trips and Resulting in Loss of Condenser Vacuum (Section 4OA3.2)
05000456/2010-002-00	LER	Limiting Condition for Operation Action not Completed Within the Required Time (Section 4OA3.3)
05000457/2010-002-00	LER	Containment Spray Pump Suction Valve Failed to Close Resulting in Unanalyzed Plant Condition due to Procedure Error (Section 4OA3.4)
05000456/2010-003-00	LER	Through-weld Leak of the Line from the 1B Seal Injection Filter to the Vent Line (Section 4OA3.6)
05000457/2010-003-00	LER	Reactor Trip Caused by Phase to Ground Fault of a Failed Crossover Damper/Deionizer Assembly due to an Inadequate Inspection Acceptance Criteria and Preventive Maintenance Inspection Frequency (Section 4OA3.5)

Discussed

None

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

- IR 1046665; NRC ID – Change Required to Eliminate UFSAR Discrepancy; March 23, 2010
- OP-AA-108-111-111; Adverse Condition Monitoring and Contingency Planning; Revision 6
- OP-AA-108-111-1001; Severe Weather and Natural Disaster Guidelines; Revision 5
- UFSAR 2.4.3.6 Probable Maximum Flood
- 0BwOA PRI-8; Auxiliary Building Flooding; Revision 5
- Braidwood Internal Site Certification Letter for Winter Readiness; November 15, 2010

1R04 Equipment Alignment

- IR 0361111; 1RH01PA – Rust/Brown Residue Between Motor and Top of Pump; August 8, 2005
- IR 0369340; Potential FME Issue 1B RHR Pump (Coating on Room Ceiling); September 1, 2005
- IR 0454535; 1AB011 Boric Acid Leakage at PKG (Clean and Adjust Packing); February 14, 2006
- IR 0454541; 1AB012 Boric Acid Leakage at PKG. (Clean/Adjust Minor Maintenance); February 14, 2006
- IR 0477836; 1RH03AB-8” Flange Boric Acid Leakage (Repair Pre A1R12); April 12, 2006
- IR 0506071; 2007 CM Boric Acid Pump Seal Leak Repairs; July 3, 2006
- IR 0597713; Wet Boric Acid Leakage (Pipe Cap) Valve 1RH8733B; February 28, 2007
- IR 0597757; Insulation Laying Loose for 1RH8724A (Superseded by IR 599519; February 28, 2007
- IR 0597815; 1RH 018C Boric Acid Leakage. (Corrective Actions Completed); February 28, 2007
- IR 0610707; 1A RH Pump Numerous Rust Spots; March 30, 2007
- IR 0626348; 1B RHR, Drain Appears Plugged, Boric Acid Accumulation on Targets; May 7, 2007
- IR 0626367; Minor Boric Acid Accumulation, Quick Disconnect Valve 1RH026A; May 7, 2007
- IR 0634797; Boric Acid Leak from 1WE0BB – ¾ Piping; May 29, 2007
- IR 06822404; Minor boric Acid Leak from 1RH01PB Seal Cooling Line; October 3, 2007
- IR 0718896; Boric Acid Leakage 1AB011 (Corrective Actions Complete FNM); January 4, 2008
- IR 0718907; Boric Acid Leakage 1AB012 (Corrective Actions Complete FNM); January 4, 2008
- IR 0754197; 1Ft-0619 Boric Acid at Packing (Corrective Actions Completed); (March 5, 2008
- IR 0786918; 1FIS-0611 Boric Acid at Threaded Connection (CA’s Completed); June 6, 2008
- IR 0789266; 1Ft-0619 Boric Acid at Packing (Corrective Actions Completed); June 6, 2008
- IR 0819862; LCO 3.5.3 Bases Improvement Recommendation; September 19, 2008
- IR 0820805; 1A RH HX Room Sump – WF Sump Need Cleaned; August 31, 2008
- IR 0870408; 1RH011B (Boric Acid Leakage at Pipe Cap) CA’s Completed; December 6, 2008

- IR 0879909; 1RH8701A Boric Acid Leakage Repair Deferral (Req. Actions); February 12, 2009
- IR 0901563; 1RH02061R (1RC8042C Target, CAs Required); April 2, 2009
- IR 1065346; 1RH01PB – Boric Acid Deposits at Pump Seal from Cooling Line; April 29, 2010
- IR 1065359; Dried Boric Acid Deposits at 1RH011B Quick Disconnect; April 29, 2010
- IR 1095833; Standing Water on Top of 1A RH Pump Casing – 1RH01PA; July 29, 2010
- IR 1098067; 1AB011 (Boric Acid at Packing, Clean/Adjust FNM); August 3, 2010
- Drawing M-62; Unit 1 Diagram of Residual Heat Removal; May 5, 1976
- BwOP FC-M1; Operating Mechanical Lineup Unit 1; Revision 8
- Drawing M-63; Diagram of Fuel Pool Cooling and Clean-up Units 1 & 2; July 23, 1975

<u>1R05 Fire Protection</u>

- Braidwood Generating Station Pre-Fire Plan #11; MCR 451' Control Room; Fire Zone 2.1-0
- Braidwood Generating Station Pre-Fire Plan #50; SWGA 451' Unit 2, Aux. Electrical Equipment Room; Fire Zone 5.5-2
- Braidwood Generating Station Pre-Fire Plan #103; AB 346' HR Pump 1A Room; Fire Zone 11.2A-1
- Braidwood Generating Station Pre-Fire Plan #104; AB 346' RHR Pump 2A Room; Fire Zone 11.2A-2
- Braidwood Generating Station Pre-Fire Plan #109; AB 346' RHR Pump 1B Room; Fire Zone 11.2D-1
- Braidwood Generating Station Pre-Fire Plan #110; AB 346' RHR Pump 2B Room; Fire Zone 11.2D-2
- Braidwood Fire Protection Report; Control Room (Fire Zone 2.1-0); Amendment 23
- Braidwood Fire Protection Report; Section 2.4.72 RHR Pump 1A Room (Fire Zone 11.2A-1) Amendment 22
- Braidwood Fire Protection Report; Section 2.4.2-78 Fire Zone Boundary BTP CMEB 9.5-1 Deviations and Generic Letter 86-10 Evaluations; Amendment 22
- Byron/Braidwood UFSAR; Section 2.3.2.1 Control Room (Fire Zone 2.1-0); Amendment 23
- Braidwood Fire Protection Report; Unit 1 Auxiliary Electrical Equipment Room (Fire Zone 5.5-1); Amendment 23
- Byron/Braidwood UFSAR; Section 2.3.5.9 Unit 1 Auxiliary Electrical Equipment Room (Fire Zone 5.5-1); Amendment 22
- BwOP FP-100T35; Unit 1 and Unit 2 Auxiliary Electrical Equipment Rooms; Fire Zones 5.5-1 and 5.5-2; Revision 4
- Figure 2.3-8 Byron/Braidwood Fire Protection Report; Main Floor at El. 451'-0"; Sheet 1 of 4; December 2006
- Figure 2.3-15; Floor Plan at 346'-0"; December 1998
- Calculation BRW-97-0477-M; HVAC Justification for Removal of Block Walls for Access to Filter Pipe Tunnels # 2 & 3; May 6, 1997
- PBI 12912; Remove the Removable Outer Block Wall Section; June 10, 2010
- PBI 13411; Need to Prop Open Door D-288; October 21, 2010
- PBI 13412; Need to Prop Door Open D573; October 21, 2010
- PBI 13413; Prop Door Open D-748; October 21, 2010
- PBI 13414; Prop Door Open D-749; October 21, 2010
- CC-AA-201; Plant Barrier Control Program; Revision 6
- Drawing A-229; Auxiliary Building Upper Basement Floor Plan, El. 383'-0, Area 2; Revision CV
- Drawing A-323; Masonry Details for Interior Masonry Walls, Units 1 & 2; Revision J
- IR 1126534, Seal Between Floor and CNMT Wall Pulled Away from Wall, October 14, 2010

1R06 Flood Protection

IR 01097315; Possible Structural Damage to Cable Vaults 1E and 2E; August 8, 2010

1R07 Heat Sink Performance

- ER-AA-340-1002; HX Inspection Report; Revision 4
- 1A Safety Injection PP Cubicle Cooler Inspection Results; October 12, 2010

1R08 Inservice Inspection Activities

- IR 901376; MT Indications Identified During ISI Exam; April 1, 2009
- IR 904471; 1RC 8042C Loss of Base Metal; April 8, 2009
- IR 904935; 1RC 8042C Structural Integrity Review Question by NRC; April 8, 2009
- ER-AA-330-009; ASME Section XI Repair Replacement Plan; Revision 5
- WPS 1-1-GTSM-PWHT; ASME Welding Procedure Specification Record (QW-482); Revision 1
- M-02 Ex. A; Pipe/Tube Fabrication and Installation Checklist; Revision 0
- BwMP 3305-025; Adjustment and Repacking of Rising Stem Valves with Garlock or Equivalent Graphite Packing; Revision 7
- MA-AA-736-610; Application of Freeze Seal to All Piping; Revision 3
- ER-AP-331; Boric Acid Corrosion Control Program; Revision 5
- ER-AP-331-1001; Boric Acid Corrosion Control (BACC) Inspection
- Locations, Implementation and Inspection Guidelines; Revision 5
- ER-AP-331-1002; Boric Acid Corrosion Control Program Identification, Screening and Evaluation; Revision 6
- ER-AP-335-01; Bare Metal Visual Examination for Alloy 600/82/182 Materials; Revision 1
- EXE-UT-350; Procedure for Acquiring Material Thickness and Weld Contours; Revision 2
- EXE-PDI-UT-1; Ultrasonic Examination of Ferritic Pipe Welds in Accordance with PDI-UT-1; Revision 6

1R11 Licensed Operator Requalification Program

- Results; Licensed Operator Annual Operating Test

1R12 Maintenance Effectiveness

- IR 1022011; Equipment Status Tag (EST) Review for OWA; January 7, 2010
- IR 1064922; 2RC014C Shows Dual at 2PM11J; May 3, 2010
- IR 1117910; Constant Buzzing Noise in the Plant Page System; September 24, 2010
- IR 1118469; PA System & Containment Evacuation Alarm Degraded; September 27, 2010
- IR 1118512, Implementation of New SSPS Cards; September 27, 2010
- IR 1121159; Emergent Dose – PA (CQ); October 2, 2010
- IR 1121371; A1R15 LL – Potential Plant Page Impact on Outage Activities; October 3, 2010
- IR 1121580; Unit 1 Cont. Evacuation. Strobes and Plant PA Speakers (Dup); October 4, 2010
- IR 1121998; Testing of Containment PA/Evacuation Sys; October 4, 2010
- IR 1122132; VLC Emergency Page Feature not Installed (PA System) October 5, 2010
- IR 1122862; Plant PA System Issues in Service Buildings; October 5, 2010
- IR 1123297; PA System Needs Expert Panel (A)(1) Determination; October 7, 2010
- IR 1127343, SSPS Response not as Expected during Return to Service; October 16, 2010
- IR 1127388; SSPS Safeguard Driver Board Indications (Train A) – 1PA09J; October 17, 2010
- IR 1128208; Adverse Trend in Instrument Issues; October 2010

- IR 1128827; Replace SSPS Train “A” Auto Tester Board A1R15; October 20, 2010
- IR 0901419; 1RC014A Closed Light Went Out when 1RC014B Stroked (DUP); April 1, 2009
- IR 0920417; 1RC014A Stroked Too Slow – Limit Switch; April 14, 2009
- IR 0915503; NOS IDS Long Standing Degraded RX Head Vent Valve Issues; May 4, 2009
- IR 0915503; 03; Rx Head Vent Valves; August 11, 2009
- IR 0331722; 2RC014 Head Vent Valve Package Active RCS LKG. Borated; October 23, 2004
- IR 0266402; External Leakage (1RC014 Sol. VA. Assemblies); October 23, 2004
- IR 0695907; NRC ID’D Question W/ Past Operability in IRS 678641/ 687349; November 7, 2007
- IR 0774791; Insulation Between RX Head Vent Valves 2RC14A/C & 2RC14B/D Missing; May 12, 2008
- IR 0265533; 1RC014A Indicates Dual When Stroked; November 19, 2004
- IR 0265561; 1RC014D Open Light Indication Does not Work; November 20, 2004
- IR 0985692; 2RC014A Has Dual Indication in the Closed Position; October 28, 2009
- IR 0953224; IR 915503 Says Replace 1RC014A in A1R15; August 13, 2009
- IR 0901386; Light Indication Lost While Attempting to Close 1RC014A; April 1, 2009
- IR 0480439; 1FSV-RC014A RX Head Vent Does not Indicate Closed; April 18, 2006
- IR 0678641; 1RC014A Slow to Indicate Full Open at PM11J; October 2, /2007
- IR 0920417; 1RC014A Stroked Too Slow- Limit Switch; April 14, 2009
- IR 0687349; 1FSV-RC014A Slow to Operate; Maybe Unreliable; October 2, 2007
- IR 0255770; Need ENG Review of New Qual. Report for 1RC014A-D Valves; September 22, 2004
- Braidwood System Health Monitoring Report; 3rd Quarter 2009; November 18, 2009
- Braidwood System Health Monitoring Report; 4th Quarter 2009; February 7, 2010
- Braidwood System Health Monitoring Report; 1st Quarter 2010; May 13, 2010
- Braidwood System Health Monitoring Report; 2nd Quarter 2010; August 1, 2010
- EP-AA-112-100-F Step; 1.3.H; Plant PA Speaker Compensatory Actions; March 2010
- EP-AA-112-100-F-01; Shift Emergency Director Checklist; Revision K
- EP-AA-120; Emergency Plan Administration; Revision 12
- EP-AA-121; Emergency Response Facilities and Equipment Readiness; Revision 9
- ER-AA-2001; System Health Indicator Program; 2010 Second Quarter
  ER-AA-2011; System Health Monitoring; Revision 11
  ER-AA-310; Implementation of the Maintenance Rule; Revision 8
  ER-AA-310-1002; Maintenance Rule Functions – Safety Significance Classification; Revision 3
  ER-AA-310-1003; Maintenance Rule – Performance Criteria Selection; Revision 3
  ER-AA-310-1004; Maintenance Rule – Performance Monitoring; Revision 8
  ER-AA-310-1005; Maintenance Rule – Dispositioning Between (a)(1) and (a)(2); Revision 5
  ER-AA-310-1007; Maintenance Rule – Periodic (a)(3) Assessment; Revision 4
  LS-AA-120; Issue Identification and Screening Process; Revision 12
  OP-AA-102-104; Security/Operations Equipment Issues Communications; Revision 1
  OP-AA-102-104; Standing Order to Address PA System Failures that Impact Site Emergency Response; Revision 1
- OP-AA-102-104; Plant PA Speaker Compensatory Actions; Revision 11
- OP-AA-108-115; Operability Determinations (CM-1); Revision 0
- SY-AA-500-127; Safety/Security Interface; Revision 0
- Condition Report (CAP001); October 7, 2010
- Maintenance Rule Periodic (a)(3) Assessment #9; November 2008 – April 2010
- EQ-GEN040; Braidwood Station Justification and Analysis (Valcor Solenoid Valves); Revision 101 3B
- BwAP 1340-5T3; Vendor Manual (2928) Solenoid Valves, Revision 0

- NRC Part 9900 Technical Guidance; Maintenance – Preconditioning of Structures , Systems, and Components Before Determining Operability; September 28, 1998
- Technical Requirements Manual; Discussion of Changes- TRM Sec. 3.4; Revision B Markup
- Technical Specification Bases; Reactor Coolant System; Amendment 89
- Technical Specification; Reactor Coolant System; Amendment 89
- Technical Requirements Manual; TRM Sec. 3.4; Revision 37
- 1BwFR-H.1; Response to Loss of Secondary Heat Sink Unit 1; Revision 202
- 1BwFR-C.1; Response to Inadequate Core Cooling Unit1; Revision 200
- BwAR 1-14-E4; RX Head Vent Temperature High; Revision 5E2
- Expert Panel Meeting Minutes; Reactor Coolant; July 24, 2000
- Expert Panel Meeting Minutes; Reactor Coolant; April 15, 1999
- Expert Panel Meeting Minutes; Reactor Coolant; May 6, 1999
- OWA Aggregate Review Report; 4th Quarter 2009
- C/O 733307; 1RC014A Reactor Vessel Head Vent Valve (A1R14) Adjust Limit Switches for Proper Identification; April 16, 2009
- 1BwOSR 5.5.8.RC-1; Reactor Coolant System Valve Stroke and Indication Test Surveillance; Revision 2
- 1BwOS TRM 3.4.e.2; Reactor Head Vent Path Valve Cycle Surveillance; Revision 1
- WO 0528957-01; Fab and Install Valve Assembly- Contingency A1R15
- WO 7944914; RX Head Vent Valve Cycle 18 Month Surveillance; April 30. 2006
- WO 0920322; RX Head Vent Valve Cycle 18 Month Surveillance; October 22, 2007
- WO 0748261 01; OP RC Valve Stroke Test (TS for Head Vent FLW Verification); April 30, 2006
- WO 1076439; RX Head Vent VLV Cycle 18 Month Surveillance; April 16, 2009
- WO 1099722 01; OP RC Valve Stroke Test (TS for Head Vent FLW Verification); April 15, 2009
- WO 0917744 01; OP RC Valve Stroke Test (TS for Head Vent FLW Verification); October 22, 2007
- 1BwOP RC-15; Reactor Head Vent; Revision 9
- ER-AA-310-1103; Maintenance Rule- Performance Criteria Selection; Revision 3
- ER-AA-310-1104; Maintenance Rule- Performance Monitoring; Revision 8
- NUMARC 93-01; Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants; Revision 2
- NUREG 0737; Clarification of TMI Action Plan Requirements; November 1980
- NUREG 1526; Lessons 'Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants; June 1995
- -NUREG 1648; Lessons Learned from Maintenance Rule Baseline Inspections; September 1999

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

- 1BwGP 100-6; Refueling Outage; Revision 23
- 1BwGP 100-6T3; Mode 5 to 6 Checklist; Revision 14
- ER-AA-600-1043; Shutdown Risk Management; Revision 4
- OU-AA-103; Shutdown Safety Management Program; Revision 11
- IR 1137603; Discrepancies found During MOV Maintenance – 0SX165A; November 9, 2010
- IR 1128331; Boric Acid Residue: Insul. (1RH10AB-3”) Invest. LKG. Source; October 19, 2010
- Configuration Risk Management Assessment BW-CRM-028, 0SX165A Unable to Close; Revision 1
- BwOP SX-13; Essential Service Water System Leak Isolation; Revision 7
- OU-AA-103; Shutdown Safety Management Program; Revision 11

- OU-AP-104; Shutdown Safety Management Program Byron/Braidwood Annex; Revision 14
- OP-AA-208-117; Protected Equipment Program; Revision 0
- Shutdown Safety Equipment Status Checklist; October 19, 2010
- Protected Equipment List; Bus 141 and 131X Protection; October 19, 2010
- Protected Equipment List; A1R15 Unit 1 SX to Train A Protection; October 19, 2010
- Protected Equipment List; A1R15 Unit 1 Spare Penetration Protection for Containment Integrity; October 19, 2010
- Protected Equipment List; A1R15 Unit 1 SAT Protection for Power Feed; October 20, 2010
- Protected Equipment List; 1A RH Pump Designated Protected for Shutdown Cooling; October 20, 2010
- Protected Equipment List; 1B CV Pump and Train Designated Protected for Inventory Control; October 20, 2010
- Protected Equipment List; A1R15 Unit 1 RWST Protection for Makeup Source; October 20, 2010
- Protected Equipment List; A1R15 Scheduled Activities; October 20, 2010

1R15 Operability Evaluations

- IR 1128891; 1B DG KW Load Delta Between MCB and 1PL08J; October 21, 2010
- IR 0897366; Preliminary NRC info on Single Failure for SGTR MTO; March 23, 2010
- IR 1155373; Additional Information on Single Failure for SGTR MTO; December 22, 2010
- IR 1132131; 2C NR Hot Leg RTD Spare Failing Low; October 28, 2010
- IR 1100061; Applicability of Bryon IRS on CC System Design Concerns; August 8, 2010
- IR 1139618; Potential Non-Conservative Tech Spec for CC and RH; November 12, 2010
- 1BwOSR 3.8.1.10-2; 1B Diesel Generator Full Load Rejection and Simulated SI in Conjunction with UV During Load Testing; Revision 10
- 1BwOSR 3.8.1.2-2; 1B Diesel Generator Operability Surveillance
- OP-AA-108-115; Operability Determinations (CM-1); Revision 9
- 0BwOA PRI-8; Auxiliary Building Flooding; Revision 5
- Operability Evaluation 10-011; U-0 Pump Potential Non-Conservative Tech Spec; November 22, 2010
- Standing Order 10-018; Component Cooling Water Pump and Residual Heat Removal Administrative Controls; November 12, 2010
- Standing Order 10-018, Revision 1; Component Cooling Water Pump and Residual Heat Removal Administrative Controls; November 18, 2010
- UFSAR 9.2.2.1; Component Cooling Water
- IR 1137603; Discrepancies found during MOV Maintenance-0SX165A; November 9, 2010
- M42; Essential Service Water, Sheet 6; Revision T
- M42; Essential Service Water, Sheet 2A; Revision AU
- BwOP SX-13; Essential Service Water System LEAK Isolation; Revision 7
- 50.59 Screen, 0BWOA PRI-8; Revision 6 and BwOP SX-13; Revision 8 changes
- Exelon Nuclear, Braidwood PRA Application Notebook BW-CRM-028; Configuration Risk Management Assessment – 0SX165A Unable to Close; Revision 1

1R18 Plant Modifications

- 10 CFR 50.59 BRW-E-2010-163; 2C DT/TA Loop – Two Thot RTD Operation for Cycle BR2C15; Revision 0,2
- 10 CFR 50.59 BRW-E-2010-163; 1A DT/TA Loop – Two Thot RTD Operation for Cycle BR1C16; Revision 0,2
- IR 1147168; 50.59 Evals for 1A and 2C RCS RTDs Need Revision; December 1, 2010

- EC 382328; 1A DT/TA Loop- Two Thot RTD Operation Installation; Revision 0
- EC 381875; 2C DT/TA Loop- Two Thot RTD OperatioN for failed RTD 2TE-0431AZ/0430A1; Revision 0
- Letter from T.W. Simpkin, Nuclear Licensing Administrator to U.S.NRC; April 15, 1992
- WCAP-12583; Westinghouse Setpoint Methodology for Protection Systems; May 1990
- WCAP-12523; Bases for Westinghouse Setpoint Methodology for Protection Systems; October 1990
- NRC Regulatory Guide 1.187; Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments
- EC 380017; Eliminating Vacuum Breakers 2-11 on CW Blowdown Line; Revision 0
- EC 361017; Reference 21, Full Vacuum Condition Design for Pre-stressed Concrete Steel Cylinder Pipe, April 2006
- Braidwood analysis BRW-06-0073-M; Hydraulic Transient Analysis of the Circulating Water Blowdown Pipeline; Revision 4
- 10 CFR 50.59 Screen BRW-S-2010-75; EC 280017; Revision 000
- EC 0000380017; Eliminating Vacuum Breakers 2-11 on CW Blowdown Line; Revision 000

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

- WO 1128337; 1D RCP 10 Year Inspection and Related Activities
- WO 1260575; SI 8819 High Point Vent Modification
- WO 1362930; Unit 1 480V 133 Substation Transformer Replacement
- WO 1259438; SSPS Universal Card 1PA10J Logic Card Replacement
- WO 1294196; 1B AFW Rocker Cover Replacement
- WO 1217999; CS-0008B Check Valve Disassembly and Inspect
- WO 1227457; 1VP01CB Inspect Turning Vane; October 17, 2010
- WO 1243734; SX174 Removal/Disassembly, Inspection, and Reassembly
- WO 1309340; Perform RCP 10-Yr Motor Inspection in Containment; October 14, 2010
- WO 900870; 1VP01CB-M Replace RCFC Fan/Motor Shaft & Bearing; October 17, 2010
- WR 00347481; 1A S/G Upper Lateral Support Shim Package Bolting Repair
- IR 1127504; 1B RCFC (1VP01CB) Rotating in Reverse; October 18, 2010
- IR 1128337; 1RC01PD Axial End Play found Out of Spec; October 19, 2010
- IR 1129692; NRC Questions on PBI Requirements and Implementation; October 22, 2010
- 1B RCFC Supply Fan Velocity Trend; March 20, 2008 to October 20, 2010
- MA-BE-724-431, AC Motor Maintenance, Revision 4
- Corrective Action Documents Resulted from NRC Inspection

<u>1R20 Refueling and Other Outage Activities</u>

- 1BwGP 100-4T1; Flowchart; Revision 17
- 1BwGP 100-5T1; Flowchart; Revision 16
- 1BwGP 100-4; Power Descension; Revision 31
- 1BwGP 100-5 Plant Shutdown and Cooldown; Revision 40
- 1BwGP 100-6; Refueling Outage; Revision 23
- 2BwGP 100-2; Plant Startup; Revision 25
- Braidwood Station A1R15 Critpath Schedule
- A1R15 GL-88-05 Boric Acid Leakage Scope; October 2, 2010
- A1R15 GL-88-05 Boric Acid Leakage Scope; October 7, 2010
- WO 1381682, Unit 1 Containment Buttress Loose Siding
- IR 1094037; Extent of Condition from Byron EACE (B2R15 SG Support); July 21, 2010

- IR 1121593; SG 1RC01BA Upper Lateral Support Shim Pack Bolting Damaged; October 4, 2010
- IR 1121928; 2RC01BA Upper Lateral Support Shim Pack Additional Damage; October 4, 2010
- IR 1121934; SG 1RC01BC Upper Lateral Support Shim Pack Loose Nut; October 4, 2010
- IR 1122302; 134Y Feed to 134Y1 and 134Y2 Tripped Open; October 5, 2010
- IR 1122347; Containment Liner Metal Reduction Exceeding 10% Metal Loss; October 5, 2010
- IR 1123022; Critical Path Delays due to Tensioner Issues; October 6, 2010
- IR 1123611; 1RY8047 Failed as Found LLRT; October 7, 2010
- IR 1123723; Chunk of Boric Acid – 1B ECCS Recirculation Sump Piping; October 8, 2010
- IR 1123728; Boric Acid Accumulation in 1A ECCS Recirc Sump Piping; October 8, 2010
- IR 1129261; Significant Corrosion on CW Pipes for Outlet Waterboxes; October 21, 2010
- IR 1124021; Individual Entered High Radiation Area Without HRA Brief; October 8, 2010
- IR 1124173; A1R15 LL – Heavy Concentration of Boric Acid on RPV O-Rings; October 9, 2010
- IR 1124323; Structural Degradation of the North Wall of Cable Vault 1E; October 9, 2010
- IR 1124721; Clearance Order Walkdown Identified Redundant Breaker On; October 11, 2010
- IR 1124877; A1R15 LL – FME in Spent Fuel Pool; October 11, 2010
- IR 1125119; Problem Placing Fuel Insert in Spent Fuel Assembly; October 12, 2010
- IR 1125556; CO2 Tank Relief Lifting While Venting Tank; October 12, 2010
- IR 1125590; Fuel Insert 66D Was Replaced in the Fuel Shuffle Sequence; October 13, 2010
- IR 1126093; Source Range N-32 has a Rising Trend; October 14, 2010
- IR 1126044; 1SI8853A Failed it’s as found Set Pressure Test; October 14, 2010
- IR 1126650; Incorrect RF Sump Spool Piece Removed; October 14, 2010
- Prompt Investigation – IR 1126650; Incorrect RF Sump Spool Piece Removed
- IR 1126660; 1SI8880 Will not Stay Open; October 15, 2010
- IR 1127416; Fatigue Assessment; October 17, 2010
- IR 1127107; (1MP04YA) Bypass Damper Found Installed Backwards; October 15, 2010
- IR 1127139; Foreign Material Found in 1SI8889 Check Valves; October 16, 2010
- IR 1127250; Fire Extinguisher Missing and No Checkout Location Given; October 16, 2010
- IR 1127488; IST Pump Evaluation Required for 1A SI Pump; October 18, 2010
- IR 1127425; 1A SI Pump Failed DP and Flow Testing Criteria; October 17, 2010
- IR 1127461; Discrepancy Between 1CB118A/B/C Spray Valves; October 17, 2010
- IR 1127491; 1VP01CA (1A RCFC) Fan Gasket not Installed Causing Leakage; October 18, 2010
- IR 1127510; IST Pump Evaluation Required for 1B SI Pump; October 18, 2010
- IR 1127531; CST Overflow During Demin Flushing Pump Restoration; October 17, 2010
- IR 1127707; Work not Complete Before 1C RCFC Closed; October 18, 2010
- IR 1127893; Flash and Smoke Reported at Transformer 133Z; October 18, 2010
- IR 1127898; Results of 15B Drain cooler Pressure Test Questionable; October 18, 2010
- IR 1127932; Debris found on Top of Reactor Head Flange; October 19, 2010
- IR 1128331; Boric Acid Residue, 1RH10AB-3”; October 19, 2010
- IR 1128397; 1RH10CB-3” to 1RH11AB-0.75 Indication of Pressure Boundary Leakage; October 20, 2010
- IR 1128686; 1TE-0613: 1B RH HX Thermocouple Loose in Well; October 20, 2010
- IR 1129826; Laboratory Temp Power Secured w/o Notifying Operations; October 22, 2010
- IR 1130063; A1R15 LL Waterbox WOs Need Revision to Reflect New Procedure; October 24, 2010
- IR 1135562; 1C RCP Seal Injection Flow Indication Reading High; November 4, 2010
- IR 1136437; Tripped U-1 Main Turbine due to High Bearing Vibes; November 5, 2010
- IR 1136784; Security Officer Self Declared Fatigue; November 7, 2010
- IR 1136945; SI Pump Discharge Header Depressurized Using BWOP SI-2; November 7, 2010

- CAP102 Report BRW MRC CR Review; Repetitive Issues with DG Temperature Switches
- ER-AP-331-1002; Evaluation of Boric Acid Leakage; Revision 5
- OP-AA-108; Startup Checklist for A1R15
- OP-AP-300-1003; Reactivity Maneuver (ReMA) Form; Revision 3
- HU-AA-1211; Low Power Physics Testing, Revision 5
- LS-AA-119; Fatigue Management and Work Hour Limits; Revision 8
- LS-AA-119-1003; Calculating Work Hours; Revision 0
- LS-AA-119-1004; Reviews and Reporting; Revision 0
- Issue Terminator Team Turnover; October 4, 2010
- Figure 4.4-4; RSG Structural Model Steam Generator Supports & Restraints
- Drawing S-1114-1; N.S.S.S. Support Framing Steam Generator Support Sections and Details Unit 1
- Violation Report; October 1, 2010 – October 22, 2010

<u>1R22 Surveillance Testing</u>

- IR 1121641; A1R15KK – Impact of RH Suction Valve Position Testing; October 4, 2010
- Units 1 & 2 Inservice Testing Program Third Ten Year Interval; Commercial Service Dates: Unit 1-7/29/88; Unit 2-10/17/88; Revision May 7, 2010
- IST Program Plan Units 1 & 2, Third Interval; Cold Shutdown Justification: CS-5; July 21, 2009
- IST Program Plan Units 1 & 2, Third Interval; Cold Shutdown Justification: CS-17; July 21, 2009
- Braidwood IST Bases Document; Valve 1RH8702A/B, MOV B RH PP Suction. From C HL DWST Isolation Valve
- Braidwood IST Bases Document; Valve 1CV8355A; CV 1/2CV83355A-D; RCP Seal Injection Isolation Valve
- WO 1225496 01; IST-STT-1RH8702A/B Valve Stroke Surveillance; October 11, 2010
- WO 1227409 01; IST-STT-1CV8355A/B/C/D-U1 RCP Seal Injection Isolation Valves; October 5, 2010
- WO 1229691 01; IST-PIT-8355A/B/C/D – Seal Injection Isolation Valves; October 5, 2010
- ER-AA-321; Administrative Requirements for Inservice Testing; Revision 10
- ER-AA-321-1007; IST Program Corporate Technical Positions; Revision 0
- Generic Letter 89-04; Guidance on Developing Acceptable IST Programs; April 3, 1989
- NUREG-1482; Guidelines for Inservice Testing at Nuclear Power Plants; April 1995
- NUREG-1482; Guidelines for Inservice Testing at Nuclear Power Plants – Final Report; Revision 1;
- NRC Inspection Manual Part 9900; Maintenance – Preconditioning of Structures, Systems, and Components Before Determining Operability; September 28, 1998
- Appendix J to Part 50 – Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
- UFSAR Table 6.2-58; 1RH8702A
- UFSAR Table 6.2-58; 1CV8355A/B; Containment Isolation Provisions
- UFSAR Bases B 3.4.14-1; RCS Pressure Isolation Valve Leakage
- UFSAR Bases B 3.6.3; Containment Isolation Valves
- UFSAR Bases 3.5.2 ECCS – Operating; Amendment 98
- UFSAR Bases 3.5.3 ECCS – Shutdown; Amendment 134
- 1BwGP 100-5; Plant Shutdown and Cooldown; Revision 40
- 1BwGP 100-6; Refueling Outage; Revision 23
- 1BwGP 100-6T-1; Flowchart; Revision 15
- BwOP SI-100; Energizing and De-Energizing SVAG Valve MCCS and SI Accumulator Outlet Valves in Modes One through Four; Revision 3

- 1BwOSR 3.4.14.1; RCS Pressure Isolation Valve Leakage Surveillance
- ASME OMb Code-2003 Addenda to ASME OM Code-2001 Code for Operation and Maintenance of Nuclear Power Plants
- Braidwood Operations Log; From 10/4/2010 to 10/4/2010
- Braidwood Operations Log; From 10/5/2010 to 10/5/2010

2RS1 Radiological Hazard Assessment and Exposure Controls

- RP-AA-203-1001; Personnel Exposure Investigation; EDE Results for Primary TLD; March 02, 2010
- RP-AA-210; Dosimetry Issue, Usage, and Control; Revision No. 18
- RP-AA-460; Controls for High and Locked High Radiation Areas; Revision No. 20
- RP-AA-460-001; Controls for Very High Radiation Areas; Revision No. 02
- RP-AA-460-002; Additional High Radiation Exposure Control; Revision No. 0
- RP-AA-350; Personnel Contamination Monitoring, Decontamination, and Reporting; Revision No. 09
- RP-AA-460-002; Approval for Working in Area > 1500mrem/hr Radiation Field and/or Electronic Dosimeter Accumulated Dose Alarm Greater than 500 mrem; Diving Activities; October 14, 2010
- RP-AA-203-1001; Personnel Exposure Investigation; Revision No. 06
- RP-AA-203-1001; Personnel Exposure Investigation; Worker Working on RE Line where RCS Drain Down Received Dose Rates Alarm; October 06, 2010
- A1R15 Outage Exposure Totals; October 21, 2010
- IR 1124021; Individual Entered High Radiation Area without HRA Brief; October 08, 2010
- IR 1120500; Vacuum Equipments Brought to the Plant without Proper Hepa Filters; October 01, 2010
- IR 1122259; Reactor Head Cono-Seal Shielding; October 05, 2010
- IR 1122291; Operator Received ED Dose Alarm While at RHR Room; October 05, 2010
- IR 1123945; A Laborer Received Dose Rate Alarm while working at 401' Outside the Missile Barrier; October 06, 2010
- IR 01124196; A1R15; Personnel Contamination Event due to Poor Human Performance; October 08, 2010
- IR 1124499; Prior to Cavity Flood-up, Trinuc Filtration Units were not Plugged to the proper Receptacle; October 09, 2010
- IR 1124735; Personnel Contamination Event on Orex Pants; October 10, 2010
- IR 1125207; Dose Received during Inspecting Wrong Area on the Reactor Head Resulting in 19 mrem Dose; October 10, 2010
- IR 1125665; Temporary Light Cord was Running through a Door into Chemistry Office from RCA to non-RCA without Boundary; October 11, 2010

2RS2 Occupational ALARA Planning and Controls

- RWP-10011314; A1R15; All Incore Sump Entries/Inspections; Revision 0
- RWP-10011303; A1R15; Fuel Moves and Trinuc Work; Revision 0
- ALARA Work-In-Progress Review of RWP-10011303; 80 Percent Completed and Significant Increased of RCA Time; October 14, 2010
- RWP-10011309; Scaffold Construction (Install/Remove); October 08, 2010
- RWP-10011310; Lead Shielding Install and Maintenance and Removal; October 07, 2010
- RWP-10011315; Snubber Removal and Installation; October 10, 2010
- RWP-10001321; Valve Team Outage Activities at Unit-1 Containment; October 15, 2010

- RWP-10011333; Reactor Head Component Disassembly and Reassembly Including Reactor Lift Prep; Revision 0
- RWP-10011356; A1R15; Diving Operations; Revision 0
- ALARA Plan No. 10011356; Diving Activities in Contaminated Water; Revision 11
- ALARA Briefing Checklist; Diving Operations in Reactor Cavity; October 13, 2010
- RP-AA-400; ALARA Program; Revision No.7
- RP-AA-401; Operational ALARA Planning and Control; Revision No.12
- RP-AA-403; Administration of the Radiation Work Permit Program; Revision No. 01
- Radiation Protection Audit Report: Braidwood Station; August 03 – 13, 2009
- NRC-Form 748; National Source Tracking Transaction Report; January 21, 2009

4OA1 Performance Indicator Verification

- LS-AA-2100; Monthly Data Elements for NRC RCS Leakage – 3rd Quarter 2009; Revision 5
- LS-AA-2100; Monthly Data Elements for NRC RCS Leakage – 4th Quarter 2009; Revision 5
- LS-AA-2100; Monthly Data Elements for NRC RCS Leakage – 1st Quarter 2010; Revision 5
- LS-AA-2100; Monthly Data Elements for NRC RCS Leakage – 2nd Quarter 2010; Revision 5
- LS-AA-2100; Monthly Data Elements for NRC RCS Leakage – 3rd Quarter 2010; Revision 5

4OA2 Identification and Resolution of Problems

- IR 0129797; Non Rated Barrier Assemblies; January 30, 2002
- IR 0237657; 2CV121 Limit Switch Pickup Needs Adjustment; July 21, 2004
- IR 0237705; 2CV121 (CV Pump Flow Control Valve) Was Full Open; July 21, 2004
- IR 0318044; 2CV121 Worked Erratically in Auto after Unit 2 Reactor Trip; March 28, 2005
- IR 0318594; VCT Level High Causing LCO 3.3.9 Entry Post U2 RX Trips; March 29, 2005
- IR 0345458; Issue 122993, CV – CV121 Transient Response, Mystery, August 15, 2005
- IR 0648427; 1CV121 Control Problems; July 9, 2007
- IR 0798384; Higher Demand than Expected on 2CV121; July 20, 2008
- IR 0911478; Fluctuating Output on 2CV121 after Rx Trip; April 24, 2009
- IR 0906323; U2 Seal Injection flow Shift on all 4 Pumps; April 12, 2009
- IR 0909802; Problems Maintaining Pressurizer Level on Unit 2; April 21, 2009
- IR 1019219; 2CV121 Operation; January 21, 2010
- IR 1028567, PDMS Declared Conservatively Inoperable due to Vendor Notice; February 10, 2010
- IR 1028610, Bryon and Braidwood Beacon Models – Incorrect Grid Locations; February 10, 2010
- IR 1029805, Beacon to Potentially Calculate DNB Non-Conservatively; February 12, 2010
- IR 1073616, RH System Issue Associate with Westinghouse NSAL 09-08; May 26, 2010
- IR 1102032; Valve, 2CV121, not Controlling Properly in Auto During Trip; August 16, 2010
- IR 1106760; Review 2CV121 Performance from Unit 2 Trip; August 27, 2010
- IR 1103659; 1CV121 Oscillation at Low Flow; August 19, 2010
- IR 1124149; Water Intrusion, October 9, 2010
- IR 1124168; NRC Questioned Use of Flagging on VCT Vent Valve C/S; October 8, 2010
- IR 1124182; NOS IDS Unsafe Work Practices; October 9, 2010
- IR 1124322; Inspect Valve 1RE9160B for Water Instrusion; October 9, 2010
- IR 1124351; 2C Forebay Diver Inspection Bryozoa Found; October 9, 2010
- IR 1125073; Review Bryon IR 1124843 for Applicability to Braidwood; October 11, 2010
- IR 1143538; Operations Work Hours Rule Issue; November 22, 2010
- IR 1127314; Overflow of the 1RC087A Vent Line; October 17, 2010
- IR 1123535; Debris found in Unit 1 “A” Isophase; October 7, 2010

- IR 1129932; Bryzozoa Inspection 1B CW Forebay Postponed; October 23, 2010
- IR 1121857; Water Found on Replacement Transformer 133V; October 4, 2010
- IR 1121869; NOS ID Deficiencies during a Walkdown of Unit 1 TB 451'; October 4, 2010
- IR 1123894; 1 B RH Pump Differential Pressure Outside IST Allowable; October 8, 2010
- IR 1127531; CST Overflow during Demin Flushing Pump Restoration; October 17, 2010
- IR 1128860; Running Current Exceeded Nameplate in High Speed; October 21, 2010
- IR 1147601; NRC ID – Fire Brigade Truck Keys Unlocked at Storage Location; December 2, 2010
- IR 1155372; RH System Issue Resulting in LER – Tracking; December 22, 2010
- IR 1126534; Seal between Floor and CNMT Wall Pulled away from Wall; October 14, 2010
- IR 1126594; Unit 1 need Sesimic Cap Repaired – Electrical Pent Area; October 14, 2010
- IR 1151841; Use of Ice Melt Salt on Fresh Concrete for ISFSI Project; December 13, 2010
- IR 1145953; NRC Questions to Shift Manager; November 29, 2010
- IR 1145713; Water Inside 2AR-20J RM-80 and Junction Box; November 29, 2010
- IR 1154339; LCO 3.7.8 Question; December 20, 2010
- IR 1137938; RCR 860458 CAPR1 Deemed Ineffective; November 9, 2010
- IR 1123611; 1RY8047 Failed as found LLRT; October 7, 2010
- IR 1130350; SAT 242-1 Gas in Oil Level Exceeds NEIL limits; October 25, 2010
- IR 1141271; 1B SX Pump Dead Bryzoa Found at Tubesheet; November 16, 2010
- IR 1140502, Unsearched Individual in the Protected Area; November 15, 2010
- IR 1140458; Containment Isolation Valve 2RE9157 Failed Stoke Time Surveillance; November 15, 2010
- IR 1154282; Inadequate Extreme Weather Protection; December 19, 2010
- IR 1154522; Screen 2CV 121 for Operator Work Around/Challenge; December 20, 2010
- IR 1115504; B Train FWI Signal; September 20, 2010
- 1BwOA PRI-15; Loss of Normal Charging Unit 1; Revision 0
- OP-AA-102-103; Operator Work-Around Program; Revision 3
- OP-AA-102-103-F-01; Operator Burden Aggregate Assessment Form; Revision B
- OP-AA-102-103-1001; Operator Burden and Plant Significant Decisions Impact Assessment Program; Revision 3
- OP-AA-102-103-1001; Operator Burden/Degraded Equipment Aggregate Assessment Form; Revision 3
- OP-AA-103-102; Watch-Standing Practices; Revision 8
- OP-AA-103-103; Operation of Plant Equipment; Revision 0
- OP-AA-108-105; Equipment Deficiency Identification and Documentation; Revision 7
- OP-AA-108-117; Protected Equipment Program; Revision 0
- OP-AA-115-101; Operator Aid Postings; Revision 2
- CV1,CVCS; November 18, 1009; Revision 13
- CV-2, CVCS Notes; November 28, 2007; Revision 8
- CV-3, RMCS & PRI-2 Flowpath; September 23, 2009; Revision 14
- ECCS-1, ECCS System; October 12; Revision 9

<u>4OA3 Event Followup</u>

- IR 1081818; Reportability Regarding 2CS009B Failing to Close; June 17, 2010
- IR 1081379; 2CS009B Did not Stroke; June 17, 2010
- IR 1081380; 2CS19B Limit and Torque Switches Found Out of Adjustment; June 17, 2010
- WO 1297971; IST 2CS001B/19B/09B3
- IR 1081818; Reportability Regarding 2CS009B Failing to Close; June 17, 2010
- IR 1131679; NRC Question Regarding Potential EAL Entry; October 27, 2010
- IR 1136544; Failure of 1MS009D to Open; November 6, 2010

- IR 893049; SSPS Universal Logic Cards not Replaced PER PCM Template; March 16, 2009
- IR 1115492; U-1 RX Trip; September 20, 2010
- IR 1073616; RH System Issue Associated with Westinghouse NSAL 09-08; May 26, 2010
- IR 110873; Missed LCO 3.3.9 entry; August 16, 2010
- LER 05000456/2010-002-00; Limiting Condition of Operation Action Not Completed Within the Required Time
- Root Cause Report; Operator Response during Unit 1 Reactor Trip Recovery Resulting in One Hour LCO Action not Completed

1EP4 Emergency Action Level and Emergency Plan Changes

- Exelon Nuclear Radiological Emergency Plan Annex for Braidwood Station; Revisions 22, 23, 24, and 25

LIST OF ACRONYMS USED

AB	Auxiliary Building
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BDPS	Boric Dilution Protection System
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CS	Containment Spray
ºF	Degree Fahrenheit
DMBW	Dissimilar Metal Butt Weld
ECCS	Emergency Core Cooling System
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Issue Report
LCO	Limiting Condition of Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OSP	Outage Safety Plan
PARS	Publicly Available Records System
RFO	Refueling Outage
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SDP	Significance Determination Process
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VCT	Volume Control Tank
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
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Enclosure: Inspection Report 05000456/2010005; 05000457/2010005
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Letter to M. Pacilio from E. Duncan dated February 8, 2011.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2, NRC INTEGRATED INSPECTION REPORT 05000456/2010005; 05000457/2010005

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