February 1, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer and
Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000456/2007006; 05000457/2007006

Dear Mr. Pardee:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Braidwood Station, Units 1 and 2. The enclosed report documents the inspection results, which were discussed on January 10, 2008, with Mr. T. Coutu, and other members of your staff.

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

The report documents two NRC-identified findings of very low safety significance. One finding was reviewed under the NRC traditional enforcement process and was determined to be a Severity Level IV violation of NRC requirements. However, because of the very low safety significance and because both findings were entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations consistent with Section VI.A of the NRC Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Braidwood Station.

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

Sincerely,

/RA/

Richard A. Skokowski, Chief
Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000456/2007006; and 05000457/2007006
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Braidwood Station
Plant Manager - Braidwood Station
Regulatory Assurance Manager - Braidwood Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Braidwood, Byron and LaSalle
Associate General Counsel
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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Letter to C. Pardee from R. Skokowski dated February 1, 2008

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 NRC INTEGRATED INSPECTION REPORT 05000456/2007006; 05000457/2007006

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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:	05000456/2007006 and 05000457/2007006
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	October 1 through December 31, 2007
Inspectors:	S. Ray, Senior Resident Inspector G. Roach, Resident Inspector T. Bilik, Reactor Inspector A. Garmoe, Reactor Engineer J. Jacobson, Senior Reactor Inspector R. Jickling, Senior Emergency Preparedness Analyst R. Jones, Reactor Engineer B. Jose, Reactor Engineer D. Lords, Reactor Engineer M. Mitchell, Health Physicist B. Metrow, Illinois Dept. of Emergency Management M. Perry, Illinois Dept. of Emergency Management
Observers:	K. Streit, Nuclear Safety Professional Development Program A. Shaikh, General Engineer
Approved by:	R. Skokowski, Chief Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2007006, 05000457/2007006; 10/01/2007 – 12/31/2007; Braidwood Station, Units 1 & 2; Inservice Inspection Activities.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. One Green finding and one Severity Level IV finding were identified by the inspectors. These findings were considered Non-Cited Violations of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Non-Cited Violation of 10 CFR 50.55a(a)(3)(i) for failure to apply an approved alternative to the American Society of Mechanical Engineers Code to evaluate susceptibility of bolting corrosion and the potential for failure after identifying leakage at residual heat exchanger flow control valve assembly, valve 2RH606, bolted connection. The primary cause of this failure was related to the cross-cutting component of Human Performance, Work Practices (Item H.4.(b) of IMC 0305) because licensee personnel failed to follow procedures. As part of its corrective actions, the licensee performed a review of 160 bolted-connection boric acid leaks and identified 47 similar examples (including 2RH606). The licensee planned to assign a work group evaluation to determine the appropriate additional corrective actions.

The finding was more than minor because it met the criteria in IMC 0612, Appendix E, "Examples of Minor Issues," Example 4a. Specifically, the licensee routinely failed to perform/document engineering evaluations to evaluate bolted connections with boric acid leaks. The issue was of very low safety significance based on Phase 1 screening in accordance with IMC 609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations." Specifically, no failures of American Society of Mechanical Engineers Code bolted connections had actually occurred due to a failure to perform this evaluation. (Section1R08.3b)

Cornerstone: Miscellaneous

Severity Level IV. The inspectors identified a Severity Level IV Non-Cited Violation of Technical Specification 5.2.2.d for not properly implementing and maintaining procedures for controlling plant staff work hours of personnel performing safety-related activities. Procedure LS-AA-119, "Overtime Controls," Revision 4, was deficient in that it permitted the plant manager to authorize work-hour deviations for routine refueling outage activities. This issue has a cross-cutting aspect in the area of Human Performance, Resources (Item H.2.(c) of IMC 0305), because Procedure LS-AA-119 did not provide adequate instructions to provide reasonable assurance that station management would properly control overtime for plant staff performing safety-related functions to assure nuclear safety as required by Technical Specification 5.2.2.d.

The violation is more than minor because, if left uncorrected, the excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events and would become a more significant safety concern. The finding is not suitable for Significance Determination Process evaluation, but has been reviewed by NRC management in accordance with IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." The resulting increased likelihood of human error, would adversely affect the station's defense-in-depth. However, management determined the violation to be of very low significance, because no significant events or human performance issues were directly linked to personnel fatigue as a result of the hours worked. The licensee added this issue to their corrective action program to address correcting the procedure. In accordance with the NRC Enforcement Policy, Supplement I.D, the issue, being evaluated as having very low safety significance by the Significance Determination Process, is a Severity Level IV Violation. (Section 1R20)

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

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 entered the inspection period shutdown for a refueling outage. The unit remained shutdown until October 25, 2007 when the main generator was synchronized to the grid and a gradual power ascension was commenced. Unit 1 reached full power on November 10, 2007 where it remained until November 16, 2007 when it was ramped down to 86 percent due to an unexpected shift in the 1B main feed pump speed controller. Following repairs to the controller, full power was again achieved on November 21, 2007. On November 25, power was reduced to approximately 87 percent power when a hydraulic leak occurred on the #2 high pressure turbine governor valve. Following repairs to the valve, Unit 1 was ramped back to full power on November 27, 2007 where it remained for the rest of the inspection period.

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

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's 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. The inspectors also reviewed corrective action program items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their corrective action program in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment. The inspectors selected protection of outside tanks, specifically the refueling water storage tanks and protection of the Lake Screen House for the sample. The inspectors ensured that heating systems including power supplies and controllers were operable and operating. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program. This inspection constitutes one winter seasonal readiness preparations sample as defined in Inspection Procedure 71111.01.

b. Findings

No findings of significance were identified.

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:

- 1B residual heat removal (RH) system train in preparation for a 1A RH work window;
- 2B essential service water (SX) system train in preparation for a 2A SX work window;
- 0A control room ventilation (VC) system train in preparation for a 0B VC work window; and
- 1A diesel generator (DG) train in preparation for a 1B DG corrective maintenance outage.

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, Administrative TS, outstanding work orders, 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 corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment. This inspection constitutes four samples as defined in Inspection Procedure 71111.04.

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 completing eight samples as defined by Inspection Procedure 71111.05-05:

- 1A DG and diesel oil storage tank rooms (Zones 9.2-1 and 10.2-1);
- 2A DG and diesel oil storage rooms (Zones 9.2-2 and 10.2-2);
- 1B DG and diesel oil storage tank rooms (Zones 9.1-1 and 10.1-1);
- 2B DG and diesel oil storage tank rooms (Zones 9.1-2 and 10.1-2);
- Division 11, miscellaneous electrical equipment room (MEER) and battery room (Zone 5.6-1);
- Division 21, MEER and battery room (Zone 5.6-2);
- Division 22, engineered safeguards features switchgear room (Zone 5.1-2); and
- 2A safety injection (SI) system pump room (Zone 11.3A-2).

The inspectors reviewed 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 had 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 with later additional insights, their potential to impact equipment that 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 corrective action program.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) (71111.08P)

.1 Piping Systems ISI

a. Inspection Scope

From October 4, 2007 through October 17, 2007, the inspectors conducted a review of the implementation of the licensee's Risk-Informed Inservice Inspection Program (RI-ISI) program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries. The inspectors selected the licensee's RI-ISI program components and American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI required examinations and Code components in order of risk priority as identified in Section 71111.08-03 of the inspection procedure, based upon the ISI activities available for review during the on-site inspection period.

The inspectors observed or performed a record review of the following two types of nondestructive examination (NDE) activities to evaluate compliance with the ASME Code Section XI and Section V requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI requirements or an NRC approved alternative (e.g., approved relief request):

- ultrasonic examination (UT) of SI weld 1SI-02-37 (pipe-to-elbow weld) on line 1RC35AA-6; and
- reviewed records for the dye penetrant (PT) examination of control rod drive mechanism housing weld 1RV-03-54.

The inspectors reviewed an examination completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service to verify that the licensee's acceptance was in accordance with the Section XI of the ASME Code. Specifically, the inspectors reviewed the following records:

- visual examination (VT-2) records of isolation valve 1CS002B (1B containment spray pump suction manual isolation valve). During this examination, the licensee recorded boric acid deposits on an ASME bolted connection. The condition was evaluated in accordance with an approved relief request prior to returning the unit to service.

The inspectors reviewed pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous refueling outage to determine if the welding acceptance and preservice examinations (e.g. visual testing, PT, and weld procedure qualification tensile tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements. Specifically, the inspectors reviewed welds associated with the following work activities;

- repair/replacement (welding) of ASME Class 1, pressurizer heater tube plug and cap (1RY01S, heater number 52); and
- repair/replacement (welding) of ASME Class 2, chemical and volume control system body to bonnet seal weld (1CV8348).

The inspectors observed activities associated with the pressurizer preemptive weld overlays of the alloy 600 penetration welds. This included overlays of three safety valves, pressure operated relief valve, the spray line, and the surge line.

The above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.2 Pressurized Water Reactor Vessel Upper Head Penetration (RVUHP) Inspection Activities

a. Inspection Scope

At the end of Cycle 13, the licensee's effective degradation years for Unit 1 was 2.44, which places the unit in the low susceptibility ranking category. For the NDE activity performed by the licensee with regard to the RVUHPs, the inspectors performed the following through direct observation:

- verified that the activities were performed in accordance with the requirements of NRC Order EA-03-009; and
- verified that indications and defects, if detected, were dispositioned in accordance with the ASME Code or an NRC approved alternative (e.g., approved relief request).

In keeping with the Order, a visual examination (VT-2) was performed. The inspectors conducted direct observation of a minimum of 20 percent of the head penetrations, and confirmed visual examination quality to ensure required examination coverage.

The inspectors reviewed the NDE examination procedures and confirmed that the resolution requirements were met.

There were no examinations completed during the previous outage with relevant/recordable conditions/indications that were accepted for continued service.

There were no welding repairs on the upper head penetrations completed since the beginning of the previous refueling outage.

The above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.3 Boric Acid Corrosion Control (BACC) Inservice Inspection

a. Inspection Scope

From October 1, 2007 through October 17, 2007, the inspectors reviewed the BACC inspection activities conducted pursuant to licensee commitments made in response to NRC Generic Letter (GL) 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary."

The inspectors conducted a direct observation of BACC visual examination activities to evaluate compliance with licensee BACC program requirements and 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" requirements. Specifically;

- on October 1, 2007, following shutdown, the inspectors accompanied licensee personnel for portions of the post-shutdown normal operating pressure normal operating temperature boric acid walkdown in containment. The inspectors verified that visible boric acid leaks were identified by the licensee and entered into their BACC program; and

- the inspectors also reviewed the visual examination procedures and examination records for the BACC examination to determine if degraded or non-conforming conditions were properly identified in the licensee's corrective action system.

The inspectors reviewed the engineering evaluations performed for the following corrective action documents to ensure that ASME Code wall thickness requirements were maintained:

- IR 284476; component 1CS002B, 1 B containment spray pump suction manual isolation valve; and
- IR 548517; component 2RH606, RH heat exchanger 2A flow control valve assembly.

The inspectors also reviewed a number of boric acid leak corrective actions to determine if they were consistent with the requirements of the ASME code and 10 CFR Part 50, Appendix B, Criterion XVI. The documents reviewed during this inspection are listed in the Attachment to this report.

The above counted as one inspection sample.

b. Findings

Failure to Perform Evaluation of a Leaking Bolted Connection

Introduction: The inspectors identified a Green Non-Cited Violation (NCV) of 10 CFR 50.55a(a)(3)(i) for failure to evaluate leakage at an ASME bolted connection for RH flow control valve assembly, valve 2RH606.

Description: In 1998, the licensee submitted a relief request and received NRC Staff approval to forgo the ASME Code requirement to remove all bolting at a bolted connection after leakage is identified and to instead perform a multi-factor evaluation to determine the susceptibility of the bolting corrosion and assess the potential for failure.

On October 11, 2007, the inspectors identified that the licensee had failed to perform an evaluation of a bolted connection with evidence of boric acid leakage. Specifically, during the pre-A2R12 ASME Borated Bolted Connection Inspections on October 23, 2006, while performing an ASME Code VT-2 examination, evidence of leakage was identified on ASME Class 2 valve 2RH606, part of the RH flow control valve assembly. The licensee’s corrective measures consisted of cleaning and re-torqueing the bolting on the valve. The requirement (ER-AP-331-1002, “Boric Acid Corrosion Control Program Identification, Screening, and Evaluation,” Attachments 1 and 3) to complete an evaluation to satisfy the relief request was not met. The reason stated in the screening section of the procedure for not performing the evaluation was that it was “not required” since repairs had been completed. However, there were no provisions in the requirements for an exception on this basis. A failure to perform the required evaluation could result in equipment susceptible to the corrosive affects of boric acid being returned to service in a degraded condition. After identification by the inspectors, the licensee documented the issue in AR 684185 and the licensee planned to assign a work group evaluation to determine appropriate additional corrective actions.

Analysis: The inspectors determined that the failure of the licensee to perform an evaluation of a bolted connection with evidence of leakage as required by relief request 12R-13 was a performance deficiency that warranted a significance evaluation. As part of its corrective actions, the licensee performed a review of 160 bolted-connection boric acid leaks and identified 47 similar examples (including 2RH606). The finding was more than minor because it met the criteria in IMC 0612, Appendix E, “Examples of Minor Issues,” Example 4a. Specifically, the licensee routinely failed to perform/document engineering evaluations to evaluate bolted connections with boric acid leaks.

The inspectors applied the IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situation” to this finding. The inspectors answered “no” to all of the questions in the Mitigation System Cornerstone column of the Phase 1 worksheet, and the finding was determined to be of very low safety significance (Green). Specifically, no failures of ASME code bolted connections had actually occurred due to a failure to perform this evaluation.

The primary cause of this failure was related to the cross-cutting component of Human Performance, Work Practices (Item H.4.(b) of IMC 0305) because licensee personnel failed to follow procedures. Specifically, the licensee repeatedly failed to complete the “Evaluation of Boric Acid Leakage from Bolted Connection” section of the ER-AP-331-1002, Revision 3.

Enforcement: On October 11, 2007, the inspectors identified an NCV of 10 CFR 50.55a(a)(3)(i) in that the licensee failed to complete an engineering evaluation on a bolted connection with evidence of leakage in accordance with an approved alternative to the ASME Boiler and Pressure Vessel Code.

10 CFR 50.50a(g)(4) requires pressurized water reactors to meet the requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda.

IWA-5250(a)(2) of ASME Section XI, 1989 Edition, no Addenda, states that (while performing VT-2 examinations) “if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.”

10 CFR 50.55a(a)(3) permits alternatives to the requirements of paragraph (g) of that section when authorized by the Director of the Office of Nuclear Reactor Regulation (NRR), and provided, in part, in 10 CFR 50.55a(a)(3)(i), that the applicant demonstrates that proposed alternative would provide an acceptable level of quality and safety.

By letter, dated April 17, 1998, the licensee submitted a relief request (12R-13, Revision 0) from the IWA-5250(a)(2) requirement to remove all bolting after leakage is identified and proposed instead that “an evaluation will be performed to determine the susceptibility of the bolting corrosion and assess the potential for failure.”

By letter, dated October 26, 1998, the NRC concluded that relief request, 12R-13, provided an acceptable level of quality and safety, and authorized the proposed alternative for the second interval pursuant to 10 CFR 50.55a(a)(3)(i).

Contrary to the above, on October 23, 2006, an evaluation of the bolted connection (for valve 2RH606) when leakage had been identified, was not performed. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (AR 684185), it is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000456/2007006-01; 05000457/2007006-01, Failure to Perform an Evaluation on a Bolted Connection).

.4 Steam Generator (SG) Tube Inspection Activities

a. Inspection Scope

From October 4, 2007 through October 12, 2007, the inspectors performed an on-site review of SG tube examination activities conducted pursuant to TS and the ASME Code Section XI requirements. The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documents related to the SG ISI program to determine if:

- in-situ SG tube pressure testing screening criteria and the methodologies used to derive these criteria were consistent with the Electric Power Research Institute (EPRI) TR-107620, "Steam Generator In-Situ Pressure Test Guidelines";
- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to identify tube degradation based on site and industry operating experience by confirming that the ET scope completed was consistent with the licensee's procedures, plant TS requirements and EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6;
- the licensee identified new tube degradation mechanisms;
- the SG tube ET examination scope included tube areas which represent ET challenges such as the tubesheet regions, expansion transitions, and support plates;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements;
- the required repair criteria are being adhered to;
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for Eddy Current Examination," of EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6;
- retrieval attempts of foreign objects were made where practicable. For those objects that were unable to be retrieved, evaluations were performed for the potential detrimental affects of the objects and appropriate repairs of the affected tubes were planned/taken; and
- the licensee identified deviations from ET data acquisition or analysis procedures.

The documents reviewed during this inspection are listed in the Attachment.

The reviews as discussed above counted as one inspection sample.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems

The inspectors performed a review of ISI/SG related problems that were identified by the licensee and entered into the corrective action program, conducted interviews with licensee staff and reviewed licensee corrective action records to determine if;

- the licensee had described the scope of the ISI/SG related problems;
- the licensee had established an appropriate threshold for identifying issues;
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity; and
- the licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure 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.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On November 19, 2007, 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. This inspection constitutes one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance 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:

- component cooling water system.

The inspectors reviewed events where ineffective equipment maintenance has resulted in 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 corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment. This inspection constitutes one quarterly maintenance effectiveness sample as defined in Inspection Procedure 71111.12-05.

b. Findings

No findings of significance were identified.

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

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:

- plant risk analysis for entry into Modes 3 and 4 with the 1A reactor containment fan cooler inoperable;
- switchyard bus 15 out-of-service with emergent out-of-service of the 1B DG;
- pressurizer spray valve 1RY455B failed shut with “C” loop pressurizer spray line unavailable;
- emergent repair of Unit 2 condensate drain pipe weld failure unisolable from main condensate header; and
- “B” train essential service water intake line through wall flaw.

These activities were selected based on their potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4) and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed during this inspection are listed in the Attachment. These activities constituted five samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- SX leak monitoring process;
- 1CV8141D repeated failure to close when draining reactor coolant loop;
- 0B VC train operability with process radiation monitor 0PR33J inoperable;
- over voltage condition on 1A DG voltage regulator circuit;
- 1C pressurizer spray line isolated with plant under normal and emergency operating conditions;
- Unit 1 main steam safety valves insulated with plant at normal operating temperature and pressure conditions; and
- “B” train SX intake line through wall flaw ASME code case structural integrity review.

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 also 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. This inspection constitutes seven samples as defined in Inspection Procedure 71111.15.-05

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

.1 Annual Resident Review

a. Inspection Scope

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

- design reviews and installation observations of the Unit 1 digital electro-hydraulic controls upgrade project.

Installation activities consisted mainly of cable termination and main control board panel mounting. Once the unit was brought back online, the inspectors observed system testing under various plant conditions. The engineering design package 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 ongoing and completed work activities to verify that installation was consistent with the design control documents. Documents reviewed are listed in the Attachment. This inspection constitutes one sample as defined in Inspection Procedure 71111.17-05.

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 for review to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- 1A DG after overvoltage event;

- 1SX147A valve stroke;
- 2SX007B valve stroke;
- 1B turbine driven feedwater pump overspeed retest; and
- Unit 1 anticipated transient without SCRAM mitigation system retest.

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 corrective action program and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment. This inspection constitutes five samples as defined in Inspection Procedure 71111.19.

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 and contingency plans for the Unit 1 refueling outage, conducted September 30 through October 25, 2007, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the refueling outage, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. The inspectors also observed licensee performance during the installation of four major plant modifications (pressurizer weld overlay, digital electro hydraulic turbine control, emergency core cooling system sump screen and downstream effects enhancements, and reactor vessel upper internals split pin change out) and their impact on outage safety. The inspectors verified that minor issues identified during the inspection were entered into the licensee’s corrective action program. Documents reviewed during the inspection are listed in the Attachment.

- initial walkdown of containment to look for evidence of reactor coolant system leakage and other discrepancies

- review of licensee configuration management, including maintenance of defense-in-depth commensurate with the outage safety plan for key safety functions and compliance with the applicable TS when taking equipment out-of-service;
- observation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;
- review of the installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication and an accounting for instrument error;'
- review of the licensee's controls over the status and configuration of electrical systems to ensure that TS and outage safety plan requirements were met, and controls over switchyard activities;
- monitoring of decay heat removal processes;
- review of the licensee's controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system;
- monitoring reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- monitoring the licensee's controls over activities that could affect reactivity;
- observations of maintenance on secondary containment as required by TS;
- observation and review of the disassembly and reassembly of the reactor vessel internals and closure head;
- observation and review of refueling activities, including fuel handling;
- observation and review of the licensee's response to leakage identified from the refueling cavity with respect to boric acid corrosion and inventory control;
- observation and review of startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- monitoring and review of licensee identification and resolution of problems related to refueling outage activities.

This inspection constitutes one refueling outage sample as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction: The inspectors identified a Severity Level IV NCV of TS 5.2.2.d for not properly implementing and maintaining procedures for controlling plant staff work hours of personnel performing safety-related activities. Procedure LS-AA-119, "Overtime Controls," Revision 4, was deficient in that it permitted the plant manager to authorize work-hour deviations for routine refueling outage activities. Consequently, the plant manager authorized plant personnel to work greater than 72 hours and up to 84 hours per 7-day period for routine outage support activities during the Braidwood refueling outage (A1R13).

Description: NRC inspectors identified an Exelon memorandum, dated September 12, 2007, that authorized a large number of individuals to work up to 84 hours in a seven-day period during the Unit 1 refueling outage. The licensee planned and scheduled Braidwood A1R13 based on station employees and contractor labor working 12 hours/day, 7 days/week through the duration of the outage. This 84 hour/week work schedule was offered to and accepted by most Exelon personnel and contractors working on A1R13 activities. The plant manager or his designated deputy, approved LS-AA-119, Attachment 1, Overtime Guideline Deviation Authorization forms for Exelon and contract employees to perform routine refueling outage support activities. The affected workers included reactor operators, senior reactor operators, auxiliary operators, health physicists, key maintenance personnel, emergency response organization members, reactor engineers supporting reactivity manipulations and fuel handling, and engineering and professional personnel performing safety-related work.

Technical Specification 5.2.2.d. requires that the amount of overtime worked by unit staff members performing safety-related functions be limited and controlled "in accordance" with the NRC Policy Statement on working hours, Generic Letter (GL) 82-12. Generic Letter 82-12 sets forth the following overtime limits, which are to be followed "during extended periods of shutdown for refueling:"

- an individual should not be permitted to work more than 16 hours straight (excluding shift overtime); and
- an individual should not be permitted to work more than 16 hours in any 24 hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period (all excluding shift turnover time).

Generic Letter 82-12 permits deviation from refueling outage limits only in "very unusual circumstances."

The inspectors determined that LS-AA-119 does not limit authorization of deviations from the work-hour limits to "very unusual circumstances." Sections 2.6.1 and 4.2.1 go beyond the limits specified in the NRC policy statement on work hours (NRC GL 82-12), by permitting deviation from the overtime guidelines to be considered for refueling outage activities. Sections 2.6.1 and 4.2.1 were first added to Procedure LS-AA-119 in Revision 3, which became effective at Braidwood on May 4, 2006. Therefore, during A1R13, overtime for Braidwood workers performing safety-related activities was not limited as required by TS 5.2.2.d.

The inspectors noted that in two instances a total of three contract maintenance personnel were found to be inattentive to their duties due to fatigue. In each instance the workers did not have safety related responsibilities during the time of their inattentiveness. The inspectors observed the licensee's Standards Team. This team was instituted for the duration of the outage in an attempt to ensure outage activities met the expectations of nuclear and industrial safety. The inspectors determined that the additional oversight of field activities during A1R13 was appropriate.

Analysis: The inspectors determined that failure to properly maintain and implement procedures to limit work-hours for plant staff performing safety-related functions in accordance with TS 5.2.2.d was a performance deficiency. Revision 3 to LS-AA-119, which permitted deviation from the work-hour limits for refueling outages, was in effect a change to the TS-required process for controlling plant staff overtime that now conflicted with TS 5.2.2.d requirements. Failure to notify the NRC of this change to the plant staff overtime controls impacted the NRC's ability to perform its regulatory function and is to be addressed through traditional enforcement.

The violation is more than minor because, if left uncorrected, the excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events and would become a more significant safety concern. The finding was not suitable for SDP evaluation, but has been reviewed by NRC management in accordance with IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." The resulting increased likelihood of human error, would adversely affect the station's defense-in-depth. However, management determined the violation to be of very low significance, because no significant events or human performance issues were directly linked to personnel fatigue as a result of the hours worked. In accordance with the NRC Enforcement Policy, Supplement I.D, the issue, being evaluated as having very low safety significance by the SDP process, is a Severity Level IV Violation.

This issue has a cross-cutting aspect in the area of Human Performance Resources (Item H.2.(c) of IMC 0305), because Procedure LS-AA-119 did not provide adequate instructions to provide reasonable assurance that station management would properly control overtime for plant staff performing safety-related functions to assure nuclear safety as required by TS 5.2.2.d. Specifically, Procedure LS-AA-119 permitted deviations to work-hour limits for routine refueling outage activities, in violation of TS 5.2.2.d. The licensee added this issue to their corrective action program, to address correcting the procedure.

Enforcement: Technical Specification 5.2.2.d requires procedures be established, implemented, and maintained covering the control of Plant Staff Overtime, to limit the hours worked by staff performing safety-related functions in accordance with the NRC Policy Statement on working hours (NRC GL 82-12). NRC GL 82-12, Nuclear Power Plant Staff Working Hours, dated June 15, 1982 specifies, in part, that during extended periods of shutdown for refueling, guidelines shall be followed that limit individuals to working no more than 72 hours in any 7-day period. Recognizing that very unusual circumstances may arise, requiring deviation from this guideline, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management.

Contrary to the above, procedures for the control of Plant Staff Overtime were not established, implemented, and maintained to limit work hours in accordance with TS 5.2.2.d. Specifically, Sections 2.6.1 and 4.2.1 of Procedure LS-AA-119, "Overtime Controls," Revision 4, permit the plant manager or designated manager to authorize deviation from the GL 82-12 work hour guidelines during refueling outage activities. Periodic refueling outages do not qualify as "very unusual circumstances" for which work hour deviations may be authorized. Consequently, during various time periods between September 30 and October 25, 2007, while in a refueling outage not qualifying as a "very unusual circumstance," the plant manager or designated manager authorized licensee employees (including reactor operators, senior reactor operators, auxiliary operators, engineers, work planners, health physicists, key maintenance personnel, and the emergency response organization members) and contractors to work 84 hours in a 7-day period to perform routine refueling outage support activities. Many of these workers performed safety-related work and none of these workers were restricted from performing safety-related activities. Because this violation was of very low safety significance, was not repetitive or willful, and it was entered into the licensee's corrective action program (IR 720481), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000456/2007006-02, Deficient Control of Plant Staff Overtime).

1R22 Surveillance Testing (71111.22)

.1 Routine 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:

- 2B auxiliary feedwater pump monthly run; and
- 1A emergency core cooling system sequencer surveillance.

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were 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, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment. This inspection constitutes two routine surveillance testing samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 In-service 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:

- 2A RH system pump ASME test.

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were 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 all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment. This inspection constitutes one inservice inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.3 Containment Isolation Valve Testing

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 containment emergency hatch local leak rate test.

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were 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, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable; where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure; where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished; prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test; equipment was returned to a position or status required to support the performance of its safety functions; and all problems identified during the testing were appropriately documented and dispositioned in the corrective action program. Documents reviewed are listed in the Attachment. This inspection constitutes one containment isolation valve inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification(s):

- installation of a temporary alarm circuit for the 2E main power transformer.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information against the design basis, the UFSAR and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). 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. This inspection constitutes one sample as defined in Inspection Procedure 71111.23-05.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

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

.1 Standardized Emergency Plan Review

a. Inspection Scope

The inspectors performed a screening review of Revisions 17, 18, and 19 of the Exelon Standardized Emergency Plan to determine whether these changes decreased the effectiveness of the licensee's emergency planning for its Illinois nuclear power stations. The inspectors also performed a screening review of the associated Revisions 17, 18, 19, and 20 of the Braidwood Annex to the Standardized Emergency Plan to determine whether changes identified in Revisions 17, 18, 19, and 20 decreased the effectiveness of the licensee's emergency planning for the Braidwood Nuclear Power Station. This review did not constitute an approval of the changes, and as such, the changes are subject to future NRC inspection to ensure that the emergency plan continues to meet NRC regulations. This inspection constitutes one sample as defined in Inspection Procedure 71114.04-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone performance indicators to determine whether or not the conditions surrounding the performance indicators had been evaluated, and identified problems had been entered into the corrective action program for resolution. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews (RWPs)

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas and airborne radioactivity areas in the plant and reviewed work packages which included associated licensee controls and surveys of these areas to determine if radiological controls including surveys, postings and barricades were acceptable for:

- lower internals moves;
- SG manway and diaphragm removal;
- emergency core cooling system sump screen modification work; and
- pressurizer weld overlay project.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors reviewed the RWPs and work packages used to access these four areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to verify that they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors walked down and surveyed (using an NRC survey meter) these four areas to verify that the prescribed RWP, procedure, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors reviewed RWPs for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g. high efficiency particulate air ventilation system operation) and to determine if there was a potential for individual worker internal exposures of >50 millirem committed effective dose equivalent. There were no airborne radioactivity work areas during the inspection period. Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The adequacy of the licensee’s internal dose assessment process for any actual internal exposures > 50 millirem committed effective dose equivalent was assessed. There were no internal exposures > 50 millirem committed effective dose equivalent. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports (LERs), and Special Reports related to the access control program to verify that identified problems were entered into the corrective action program for resolution. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

The inspectors reviewed 15 corrective action reports related to access controls and one high radiation area radiological incident (non-performance indicators identified by the licensee in high radiation areas <1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes;
- identification and implementation of effective corrective actions;
- resolution of NCVs tracked in the corrective action system; and
- implementation/consideration of risk significant operational experience feedback.

This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to determine if licensee controls were adequate. These work areas involved areas where the dose rate gradients were severe that increased the necessity of providing multiple dosimeters and/or enhanced job controls. This inspection constitutes one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends, ongoing and planned activities in order to assess current performance and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure estimates for the following five work activities that were likely to result in the highest personnel collective exposures:

- SG manway removal and bolt cleaning activities;
- emergency core cooling system sump modification activities;
- SG insulation modification activities;
- pressurized weld overlay project activities; and
- pressurized weld overlay shielding and support activities.

This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors reviewed the site specific trends in collective exposures and source-term measurements. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors reviewed procedures associated with maintaining occupational exposures As Low As Is Reasonably Achievable (ALARA) and processes used to estimate and track work activity specific exposures. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning.

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following work activities of highest exposure significance:

- SG manway removal and bolt cleaning activities;

- emergency core cooling system sump modification activities;
- SG insulation modification activities;
- pressurized weld overlay project activities; and
- pressurized weld overlay shielding and support activities.

This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to determine that the licensee had established procedures and engineering and work controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors evaluated the licensee's interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups for interface problems or missing program elements. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors reviewed work activity planning to determine if there was consideration of the benefits of dose rate reduction activities such as shielding provided by water filled components and piping, job scheduling, along with shielding and scaffolding installation and removal activities. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and bases for the current annual collective exposure estimate including procedures, in order to evaluate the licensee's methodology for estimating work activity-specific exposures and the intended dose outcome. Dose rate and man-hour estimates were evaluated for reasonable accuracy. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

The licensee's process for adjusting exposure estimates or re-planning work, when unexpected changes in scope, emergent work or higher than anticipated radiation levels were encountered, was evaluated. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following four jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- SG platform activities;
- pressurized weld overlay project activities;
- emergency core cooling system sump modification activities; and
- split pin inspection activities.

The licensee's use of ALARA controls for these work activities was evaluated by reviewing the use of engineering controls to achieve dose reductions to determine if procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding. This review constitutes one sample as defined by Inspection Procedure 71121.02-5.

The inspectors observed job sites to determine if workers were utilizing the low dose waiting areas and were effective in maintaining their doses ALARA by moving to the low dose waiting area when subjected to temporary work delays. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.5 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and to determine if the licensee was making allowances and had developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and radiation protection technician performance was observed during work activities being performed in radiation areas and high radiation areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas and that work activity controls were followed. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.7 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c). This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

Corrective action reports related to the ALARA program were selectively reviewed by the inspectors, and licensee staff members were interviewed to verify that follow-up activities were being conducted in a timely manner commensurate with their importance to safety and risk using the following criteria:

- initial problem identification, characterization, and tracking;
- disposition of operability/reportability issues;
- evaluation of safety significance/risk and priority for resolution;
- identification of repetitive problems;
- identification of contributing causes; and
- identification and implementation of effective corrective actions.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies in problem identification and resolution had been addressed as applicable. This inspection constitutes one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

2PS1 Radioactive Gaseous and Liquid Effluent Treatment and Monitoring Systems (71122.01)

Review of Blowdown Line Operations and Tritium Remediation Efforts

The inspectors continued to monitor the licensee's activities resulting from previous inadvertent leaks of tritiated liquid from the blowdown line to the Kankakee River. The inspection activities included the following:

- periodic inspections of all vacuum breaker vaults;
- periodic inspections of remediation system pump operations at the Exelon Pond, vacuum breaker 1, and lagoon area;
- efforts to reduce tritium concentrations in secondary plant systems; and
- participation in Community Information Meetings.

In addition, the inspectors continued to accompany licensee employees and contractors during their collection of water samples at 23 monitoring locations of interest. The inspectors verified by direct observation that the water samples were being taken from the locations specified, that proper sampling protocols were followed, and that split samples were properly obtained and labeled. The inspectors took direct custody of the split samples and maintained a chain of custody as the samples were sent to the U.S. Government's contract laboratory in Oak Ridge, Tennessee. The inspectors also reviewed the results of earlier split samples to ensure that the results from the licensee's and NRC's contract laboratories matched within normal statistical variance.

The inspectors verified that minor issues identified during these inspection activities were entered into the licensee's corrective action program. This inspection did not represent a completed inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 3rd Quarter 2007 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Mitigating Systems Performance Index (MSPI) - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Heat Removal System performance indicator MS-08 for both Braidwood Unit 1 and Braidwood Unit 2 for the period from the 4th quarter 2006 through the 3rd quarter 2007. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, event reports, MSPI derivation reports, and NRC Integrated Inspection reports for the period of the 4th quarter 2006 through the 3rd quarter 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 and none were identified. Specific documents reviewed are described in the Appendix to this report. This inspection constitutes two MSPI heat removal system samples as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.3 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Residual Heat Removal System performance indicator MS-09 for both Braidwood Unit 1 and Braidwood Unit 2 for the period from the 4th quarter 2006 through the 3rd quarter 2007. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Revision 5 of the NEI Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection reports for the period of the 4th quarter 2006 through the 3rd quarter 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 and none were identified. Specific documents reviewed are described in the Attachment. This inspection constitutes two MSPI residual heat removal system samples as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.4 Mitigating Systems Performance Index - Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index - Cooling Water Systems performance indicator MS-10 for both Braidwood Unit 1 and Braidwood Unit 2 for the period from the 4th quarter 2006 through the 3rd quarter 2007. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in revision 5 of the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection reports for the period of the 4th quarter 2006 through the 3rd quarter 2007 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. 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 and none were identified. Specific documents reviewed are described in the Attachment. This inspection constitutes two MSPI cooling water system samples as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

4OA2 Problem Identification and Resolution (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to determine whether they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action program as a result of the inspectors' observations are generally denoted in the Attachment. These activities were part of normal inspection activities and were not considered separate samples.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

The inspectors performed a semiannual review to identify trends that might indicate the existence of a more significant safety issue. Specifically, the inspectors selected and reviewed several common cause analysis reports completed during the period March 2007 and October 2007, two Nuclear Oversight (NOS) quarterly reports, Braidwood Station event free clock resets, System Health Report for the Instrument Air (IA) System, Braidwood Station's Maintenance Rule Periodic (a)(3) Assessment for November 2005 - April 2007, and various issue reports.

The inspectors evaluated these documents and reports to determine the licensee's threshold for identifying problems, entering them into the corrective action program, and resolving them. Also, the licensee's efforts in establishing the scope of problems were evaluated by reviewing selected self-assessments results, oversight reports, system health reports, action plans, and results from surveillance tests and preventive maintenance tasks.

Assessment

The inspectors determined that the licensee was effective at identifying problems and entering them into the corrective action program. This was evidenced by the relatively few deficiencies identified by external organizations (including the NRC) that had not been previously identified by the licensee during the review period. The station had implemented an NOS improvement opportunity to modify the oversight program to reduce the number of issues that were identified by NRC inspectors that had not been noted by NOS. Licensee assessments and common cause analyses were generally of sufficient depth and thoroughness, provided evidence that the licensee has been looking for negative trends, and generally appeared successful in identifying robust corrective actions. Also, during this evaluation, there were no instances identified where conditions adverse to quality had been handled outside the corrective action program.

Evaluation of Issues

The inspectors determined that the licensee was effective at problem evaluation. This was demonstrated by the samples of IR reviews and common cause analysis reviews. Evaluations were generally adequate and of appropriate depth. There were no instances noted in which the licensee did not consider operability and reportability requirements. The licensee appropriately considered risk in prioritizing or evaluating issues. The Station Reactivity Management analysis considered five levels of significance of reactivity management issues, which allowed the station to trend and resolve lower level precursors and prevent more significant reactivity management events.

This evaluation represents one (1) semiannual review inspection sample.

.3 Annual Sample: Elevated Steam Generator Tritium Concentrations

a. Inspection Scope

The inspectors reviewed the licensee response to elevated tritium levels in the Unit 1 and Unit 2 steam generators. Elevated tritium levels were initially detected in the Unit 1 steam generators in September 2006 and the Unit 2 steam generators in October 2006. The licensee determined the source to be cross-contamination between the secondary plant and radwaste systems. As a result, the licensee identified numerous sources of potential cross-contamination and installed blind flanges in piping to eliminate the cross-contamination boundary. A significant decrease in steam generator tritium levels was observed following installation of the blind flanges. However, in December 2007, the detected tritium concentration in the steam generators was again elevated and increasing.

b. Observations

The inspectors interviewed plant chemistry personnel, reviewed corrective action documents, and reviewed the licensee's previous actions to address steam generator tritium levels. Based on the information reviewed, the inspectors believe the licensee's actions to date have been appropriate based on the information available. The inspectors also discussed with licensee personnel the development of future actions to mitigate the elevated tritium concentrations.

.4 Annual Sample: Equipment failures due to severe weather (lightning)

a. Inspection Scope

The inspectors reviewed the licensee response to equipment failures and spurious indications and alarms associated with the effects of severe weather, in particular lightning strikes in the vicinity of the site over the previous year. The inspectors reviewed the licensee's corrective action program and compared results to licensee identified trends in the area of severe weather induced failures. In addition, the inspectors reviewed the licensee's augmented surveillance requirements and interviewed the associated system engineers regarding completed actions and proposed system monitoring and modifications to minimize the impact of lightning strikes on plant electrical equipment.

b. Observations

The inspector's review of the licensee's corrective action program corroborated with the licensee's common cause assessment that the systems most often impacted by lightning strikes were loose parts monitoring, seismic monitoring, and the Unit 1 rod control system. The licensee was planning to convert the loose parts monitoring system from analog to a digitally-based system during the next scheduled refueling outage for each respective unit. This is expected to filter out the effects of electronic noise experienced during lightning impacts.

During the fall 2007, Unit 1 refueling outage the licensee in conjunction with the vendor identified a number of loose capacitors on Unit 1 rod control system cards associated with groups that had received rod control urgent failure alarms during recent electrical storms. The licensee re-soldered all loose components and performed an extent of condition of similar circuit cards for loose components. To confirm the effectiveness of the repairs to prevent spurious alarms, the licensee has established a monitoring plan to compare system performance during electrical storm activity to determine if the loose components were prevented from mitigating the impact of electrical transients associated with lightning strikes.

The licensee has created an action through the corrective action program for engineering to evaluate the need to monitor seismic monitoring system power supply voltage through the spring and summer seasons for evaluating the impact of lightning strikes. The system remains capable of recording actual seismic events even after it has been spuriously triggered by a non-seismic transient such as lightning.

The inspectors determined that the licensee's response to these transient events was appropriate. Short term scheduled component replacement and upgrades where new technology exists or where substandard material conditions have been identified were taking place. In addition, more robust system monitoring opportunities were being identified in systems where a lack of hard data prevents determining an immediate corrective action.

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

.1 (Closed) LER 05000457/2007-001-00: Unit 2 Manual Reactor Trip Following Circulating Water Pump Trips.

This event, which occurred on August 23, 2007, was previously discussed in Inspection Report 05000456/2007005; 05000457/2007005, Section 4OA3.3. The inspectors reviewed the LER and the root cause report for the trip. Sudden winds during a particularly violent thunderstorm, generated wave action which in conjunction with a significant amount of debris from a large fish kill that had occurred two days earlier resulted in multiple short duration high differential pressure indications sensed across the circulating water pump intake traveling screens. Two of the three operating circulating water pumps on Unit 2 tripped on high differential pressure and, although operators quickly began reducing power, condenser vacuum entered the "not acceptable" region of the operating procedures that led the operators to initiate a manual reactor trip. The plant responded as designed.

The inspectors determined that the licensee's actions during and after the event were reasonable and did not constitute a performance deficiency. The licensee has temporarily removed the circulating water pump high traveling screen differential pressure trip feature, and is controlling this configuration under procedure via the licensee's temporary configuration control process. The licensee's long term solution to prevent spurious signals from causing an unwanted pump trip, include the installation of a 15 second time delay relay in the trip circuitry. Documents reviewed as part of this inspection are listed in the Attachment. This inspection represented one sample. This LER is closed.

.2 (Closed) LER 05000456/2007-002-00: Unit 1 Power Range N43 Positive Rate Trip Inoperable due to Miscalibration of Time Constant.

On September 22, 2007 with Unit 1 at full power, during the performance of a routine channel calibration the licensee discovered that the time constant setting for the high positive nuclear flux rate reactor trip associated with power range nuclear instrument N43 was outside of the Technical Specification requirement of ≥ 2 seconds. The as-found time constant value was calculated to be 1.68 seconds that would have resulted in a non-conservative trip response input from the N43 power range nuclear instrument.

The licensee's root cause investigation determined that during the previous performance of this surveillance, on March 8, 2006, the technician performing the test had improperly set the time constant function. The technician had used a provided example strip chart from the procedure as a guide for reading the actual strip chart. The example in the procedure was difficult to read, but the technician did not seek supervisory support. In addition the licensee determined that the procedure was not adequate in that it only required a peer check for determining the time constant instead of an independent verification of the calculated value. The procedure did provide accurate written instruction on how to properly determine the time constant, but it appears the technician relied heavily on the example strip chart.

Braidwood TS 3.3.1 Condition E, requires that if one channel of Power Range Neutron Flux –High Positive Rate Trip is inoperable that the channel be placed in trip within six hours or the unit be placed in Mode 3 within twelve hours from discovery of the inoperable channel. The licensee appropriately entered the TS action requirements upon discovery of channel inoperability on September 22, 2007. The channel calibration was completed and the as left value of the time constant was returned within the TS requirements. In addition, the most recent time constant calculation for all remaining channels on both Unit 1 and Unit 2 were reviewed for accuracy with no issues noted. During the time period of March 8, 2006 until September 22, 2007, Braidwood Unit 1 operated with an inoperable Power Range Neutron Flux –High Positive Rate Trip channel contrary to the above TS. The improper setting of the Power Range Neutron Flux –High Positive Rate Trip channel associated with power range nuclear instrument N43 was considered a minor violation not subject to enforcement in accordance with Section IV of the NRC Enforcement Policy. The basis for the minor violation was that the High Positive Rate Trip is not credited as the primary accident response by the reactor protection system for any accident scenario in the licensee's UFSAR. In addition the three remaining channels of Power Range Neutron Flux –High Positive Rate Trip remained operable during the time period of inoperability.

Licensee corrective actions include upgrade of the surveillance procedure with an example strip chart from the vendor and a written requirement for an independent verification of the channel time constant to be performed by a second qualified technician. The inspectors verified that the due date for completion of these corrective actions was prior to the next scheduled performance of this surveillance. This LER is closed.

4OA5 Other Activities

.1 Pressurized Water Reactor Containment Sump Blockage Temporary Instruction (TI) 2515/166

a. Inspection Scope

The purpose of this TI was to support NRC review of licensee's activities in response to NRC GL 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at PWRs." This TI required NRC inspectors to verify actions implemented in response to the NRC GL were complete and where applicable were programmatically controlled.

The inspectors performed a review in accordance with TI 2515/166 of the licensee's response to GL 2004-02 for Unit 1. The inspectors also reviewed changes to the licensee's facility and verified they were evaluated in accordance with 10 CFR Part 50.59. A more detailed, design focused team inspection will review the full scope of the modification during the first quarter of 2008. As a result, the Unit 1 portion of this TI is not yet closed.

The inspectors reviewed the licensee's modification packages and reviewed regulatory submittals as part of their preparation activities before the Unit 1 refueling outage. During the refueling outage the inspectors periodically observed work activities focusing on the critical attributes selected by the inspectors. For example, the inspectors compared trash racks, sump screens, and supports to installation drawings. The inspectors observed steam generator insulation replacement and reviewed the design drawings for the safety injection system valves modified as a part of the downstream effects portion of the modification. In addition, the inspectors closely observed Foreign Material Exclusion programs and practices to ensure material was not left inside of the new sump screens. The inspectors ensured that licensee lessons learned from the recent installation of the sump screen modification into Unit 2 were incorporated into the Unit 1 installation process.

b. Evaluation of Inspection Requirements

The TI requested the inspectors to include answers to the following questions in this inspection report.

1. Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 responses?

The licensee did implement the plant modifications and procedure changes committed to in their GL 2004-02 responses for Unit 1.

2. Has the licensee updated its licensing bases to reflect the corrective actions taken in response to GL 2004-02?

The inspectors reviewed the completed 10 CFR Part 50.59 assessments performed by the licensee and verified that the documents contained updates to the UFSAR to be submitted to the NRC at the next regular update.

The TI for Unit 2 is not complete, as the downstream effects aspect of the emergency core cooling system sumps modification will be installed during the spring 2008 outage. This inspection is not a baseline inspection program sample.

c. Findings

No findings of significance were identified.

.2 (Closed) Unresolved Item (URI 05000456/2007009-04; 05000457/2007009-04): The effects of lack of safe short circuit protection for 5 kV, #2 American Wire Gauge (AWG) cables on other cables in common raceways.

During the 2007 component design basis inspection at Braidwood Station, the inspectors raised a concern regarding the effects of lack of safe short circuit protection for #2 AWG cables supplying 4160 V loads on other cables in common raceways. Specifically, short circuit current capability curves indicated that the safe short circuit withstand time of the No. 2 copper cables for the calculated available fault currents was between 1 and 2 cycles. Review of the applied protective relaying and breaker tripping indicated that it would take between 5 and 6 cycles to isolate a postulated electrical fault at the RH or at the SI motor terminals. The inspectors concluded that the fault clearing time exceeded the cable short circuit current carrying capability. The licensee did not have formal documentation to demonstrate that cable damage resulting from maximum available short circuit currents will not compromise the integrity of other cables in common raceways (RH and SI Pump Power Supply Cables). The issue was considered unresolved pending the inspectors' review of the test data or final licensee calculation.

In response to this URI, the licensee performed a calculation (BRW-07-0135-E/BYR07-091, Revision 0; "Assessment of short circuit withstand capability of a 5 kV power cable") and concluded that a short circuit would not adversely affect adjacent cables. The impact of the fault would be contained within the faulted cable itself. This conclusion was based on analysis of heat transfer models of a faulted cable in a cable tray as well as cable performance during environmental qualification tests.

The inspectors reviewed this calculation and did not identify any deficiencies. Therefore, the inspectors determined no performance deficiencies or violations of regulatory requirements occurred and no additional enforcement action was warranted. The inspectors had no further concerns in this area. This unresolved item is closed.

This inspection activity involved follow-up from a previous inspection; therefore, no inspection samples were completed.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 10, 2008, the inspectors presented the inspection results to Mr. T. Coutu, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. The inspectors reviewed and returned one proprietary document to the licensee.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- Outage ALARA and Access Control Inspection with Mr. J. Moser, Radiation Protection Manager on October 15, 2007.
- Inservice Inspection Inspection with Messrs. J. Petty, M. Sears, B. Casey, and G. Bal on December 4, 2007 via telephone. The inspectors returned proprietary information reviewed during the inspection prior to leaving the site and the licensee confirmed that none of the potential report input discussed was considered proprietary.
- Emergency Preparedness Inspection with Mr. J. Gerrity on December 11, 2007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Coutu, Site Vice President
L. Coyle, Plant Manager
K. Aleshire, Emergency Preparedness Manager
G. Bal, Manager, Programs Engineering
D. Burton, Licensed Operator Requalification Training Group Lead
S. Butler, Operations Training Manager G. Dudek, Site Training Director
B. Casey, Programs Engineering
C. Furlow, Senior Design engineer
R. Gadbois, Maintenance Director
J. Gerrity, Emergency Preparedness Manager, Acting
D. Gullott, Regulatory Assurance Manager
J. Knight, Nuclear Oversight Manager
T. McCool, Operations Director
J. Moser, Radiation Protection Manager
J. Petty, Licensing Engineer
D. Riedinger, Design Engineering Manager
M. Sears, Programs Engineering
M. Smith, Engineering Director
T. Tierney, Chemistry, Environmental, and Radioactive Waste Manager

Nuclear Regulatory Commission

R. Skokowski, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2007006-01; 05000457/2007006-01	NCV	Failure to Perform an Evaluation on a Bolted connection (Section 1R08.3b)
05000456/2007006-02	NCV SLIV	Deficient Control of Plant Staff Overtime (Section 1R20)

Closed

05000456/2007006-01; 05000457/2007006-01	NCV	Failure to Perform an Evaluation on a Bolted connection (Section 1R08.3b)
05000456/2007006-02	NCV SLIV	Deficient Control of Plant Staff Overtime (Section 1R20)
05000457/2007-001-00	LER	Unit 2 Manual Reactor Trip Following Circulating Water Pump Trips (Section 40A3.1)
05000456/2007-002-00	LER	Unit 1 Power Range N43 Positive Rate Trip Inoperable due to Miscalibration of Time Constant (Section 40A3.2)

05000456/2007009-04; 05000457/2007009-04	URI	The effects of lack of safe short circuit protection for 5 kV, #2 AWG cables on other cables in common raceways. (Section 4OA5.1)

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 inspectors 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

- 0BwOS XFT-A1; Unit Common Freezing Temperature Equipment Protection Surveillance; Revision 14
- 0BwOS XFT-A3; Unit Common Cold Weather Surveillance; Revision 7
- 0BwOS XFT-A4; Unit Common Freezing Temperature Equipment Protection Inside Surveillance; Revision 2
- WC-AA-107; Seasonal Readiness; Revision 4

1R04 Equipment Alignment

- BwOP RH-E1; Electrical Lineup – Unit 1 Operating; Revision 2
- BwOP RH-M2; Operating Mechanical Lineup Unit 1 1B Train; Revision 8
- BwOP SX-E2; Electrical Lineup - Unit 2 Essential Service Water System; Revision 8
- BwOP SX-M2; Operating Mechanical Lineup Unit 2; Revision 25
- BwOP VC-E3; Electrical Lineup - Unit 0 “A” Chiller; Revision 4E1
- BwOP VC-M1; Operating Mechanical Lineup Unit 0 Control Room HVAC; Revision 5
- BwOP VC-M2; Operating Mechanical Lineup Unit 0; Revision 2
- BwOP VC-M3; Operating Mechanical Lineup Unit 0 “A” Chilled Water; Revision 3
- BwOP DG-E1; Electrical Lineup – Unit 1 1A Diesel Generator; Revision 6
- BwOP DG-M1; Operating Mechanical Lineup Unit 1 1A Diesel Generator; Revision 14

1R05 Fire Protection

- Byron/Braidwood Nuclear Stations Fire Protection Report; Amendment 22; December 2006
- Fire Protection Report; Figure 2.3-12, sheet 1; Grade Floor Elevation 401; Amendment 15
- Fire Protection Report; Figure 2.3-12, sheet 3; Grade Floor Elevation 401; Amendment 15
- Fire Protection Report; Figure 2.3-14, sheet 3; Basement Floor Plan Elevation 364; Amendment 18
- Fire Protection Report; Figure 2.3-8, sheet 1; Main Floor Plan Elevation 451; Amendment 22
- Braidwood Station 2007 Threat Based Scenario Full Scale Exercise Manual, July 11, 2007
- IR 577586; Battery Voltage Low Alarm for 2PA49J; January 10, 2007
- IR 636922; 2A Diesel Generator Oil Storage Tank Foam Fire Protection System Leakage; June 4, 2007
- IR 652480; 1A/C Diesel Oil Storage Tank Fire Protection Foam Nozzle Plugged; July 21, 2007

1R08 Inservice Inspection (ISI) Activities

NDE Procedures

- EXE-PDI-UT-2; Ultrasonic Examination of Austenitic Piping Welds in Accordance wit PDI-UT-2; Revision 4
- EXE-ISI-11; Liquid Penetrant Examination; Revision 1

- ER-AP-331-1002; Boric Acid Corrosion Control Program Identification, Screening, and Evaluation; Revision 3
- ER-AP-331-1001; Boric Acid Corrosion Control (BACC) Inspection Locations, Implementation and Inspection Guidelines; Revision 3
- ER-AA-330-001; Section XI Pressure Testing; Revision 4
- ER-AA-335-014; VT-1 Visual Examination; Revision 3
- ER-AA-335-015; VT-2 Visual Examination; Revision 4
- ER-AA-335-015; VT-2 Visual Examination; Revision 6
- EXE-PDI-UT-108; Ultrasonic Examination of Weld Overlaid Similar Dissimilar Welds in Accordance with PDI-UT-8; Revision 1
- GQP-9.7; Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials and Cladding; Revision 11

Head Exam

- EC 367637; Braidwood Unit 1 Effective Degradation Years (EDY) Evaluation End of Cycle 13 As-Found Condition and Projection Through End of Cycle 15 In Accordance with NRC Order EA-03-009; October 3, 2007
- ER-AP-335-1012; Bare Metal Visual Examination of PWR Vessel Penetrations and Nozzle Safe-Ends; Revision 3

NDE Exam Documents

- A1R13-UT-2; Calibration Data Sheet; October 4, 2007
- Relief Request 12R-13; Alternative Rules for Corrective Measures if Leakage Occurs at Bolted Connections; Revision 0
- A1R13-PT-009; Surface Examination Data for CRDM Housing Weld RV-03-54;
- October 6, 2007
- Code Case N-504-2; Alternative Rules for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Section XI, Division 1; March 12 1997
- Code Case N-513-1; Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1; March 28, 2001
- Code Case N-566-2; Corrective Action for Leakage Identified at Bolted Connections Section X, Division 1; March 28, 2001
- WDI-PJF-1303706-EPP-001; Examination Program Plan for the Pre-Service Inspection of Pressurizer Nozzle Structural Weld Overlays at Braidwood Unit 1; Revision 0, Amendment 1

SG Exam Documents

- ER-AP-420-0051; Conduct of Steam Generator Management Program Activities; Revision 10
- ED-BRW-06-0009; Braidwood Unit 1 A1R12 Condition Monitoring and Operational Assessment Report, July 2006; July 12, 2006
- ED-BRW-07-0006; Braidwood Unit 1 Steam Generator Inspection Degradation Assessment and Condition Monitoring Input Checklist for A1R13; September 7, 2007
- MRS 2.4.2 GEN-45; Standard In-Situ Pressure Test Using the Computerized Data Acquisition System; Revision 5
- MRS-TRC-1840; Use of Appendix H Qualified Techniques at Braidwood A1R13 Outage; September 11, 2007

Corrective Action Documents

- AR 482283; FAC Component 1RD147 Below TMin Wall Thickness; April 23, 2006
- AR 482843; UT Performed Observed wall Thinning on Line 1SX05BA-3"; April 24, 2006
- AR 482854; Pipe Rubbing on Bottom of Support Member Causing Wall Loss; April 24, 2006
- AR 563315; Inspection of the Pipe 0VF38A-4" not Performed; November 29, 2006
- AR 578313; Line 1SX25AA Doesn't Meet Wall Thickness Screening Criteria; January 11, 2007
- AR 680720; Follow Up Actions for IR#628310, SX System Through Wall Leak; October 5, 2007
- AR 541215; Boron Leakage from 2A RH Pump Seal; October 5, 2006
- AR 477836; 1RH03AB-8" Flange Boric Acid Leakage (Repair Pre AIR12); April 12, 2006
- AR 548517; 2RH606 Boric Acid Lkg. @ Blk. Off Plate (Clean/Tighten); October 23, 2006
- AR 482199; NRC Identified Issues with 1CV01AA1A & AB-1B ASME Evaluations

Corrective Action Documents Generated as a Result of ISI Inspection

- AR 00684185; BACC Program (NRC Identified Compliance Issue; October 12, 2007)

Work Orders

- WO 456491-01; Visual Exam (VT-2) of CS SYS Class 2&3 Components At NOP SECXI Period & Int; August 9, 2004
- WO 697822-01; MM-Replace Body to Bonnet Gasket on valve 1CS002B; February 14, 2005

Welding Documents

- W.O. 00746571-02; Replace Body to Bonnet Seal Weld (1CV8348); dated April 11, 2006
- W.O. 00915754-03; Plug and Cap Pressurizer 1RY015 Heater Tube #52; dated April 26, 2006

Welding Procedures and Qualification Records

- WPS 8-8-GTSM; Groove Welds and Fillet Welds, P8-P8, GTAW/SMAW, Without PWHT; Revision 1
- WPS 55-WP8/8/F6AW (Areva); Revision 1

1R11 Licensed Operator Requalification Program

- Scenario #0761; Uncontrolled Rod Motion and Steam Line Rupture Outside of Containment; Revision 0

1R12 Maintenance Effectiveness

- Operations Narrative Logs for August 6, 2006
- Component Cooling Performance Monitoring Summary for 4th Quarter 2007
- IR 518383; 2CC9412A Failed PMT Valve Stroke; August 8, 2006
- IR 680968; 1B CC Pump Trip; October 6, 2007
- IR 517198; 1CC9473B Would Not Stroke; August 6, 2006
- Engineering Change 367204; Component Cooling System LOCA Performance Evaluation During ECCS System Alignment to Cold Leg Recirculation

1R13 Maintenance Risk Assessments and Emergent Work Control

- MA-AA-716-004; Troubleshooting Log for 1RY455B; October 30, 2007
- 20E-1-4031RY27; Loop Schematic Diagram Pressurizer Pressure Control Cabinet 5 (1PA05J); Revision D
- OP-AA-106-101-1106; Plant Issue Resolution Documentation, 1RY025, 1C Pressurizer Spray Valve Manual Isolation Valve; October 25, 2007
- OU-AA-103; Shutdown Safety Approval, Enter Mode 4 with 1A RCFC Inoperable; October 23, 2007
- BB PRA-017.71A; Braidwood Unit 1 Technical Specification (TS) 3.0.4b Risk Assessment for 1A RCFC During Startup from Unit 1 Refueling Outage; Revision 1
- IR 689146; 1A RCFC High Amps; October 25, 2007
- IR 307532; Install Valve to Support Fill From Online Unit to Support Chemistry; March 2, 2005
- EC 354366; Install 2" Valve Tap on Unit 2 Similar to Unit 1 Tap Valve 1CD238; May 1, 2005
- M-124, sheet 3; Diagram of Condensate Piping (CD) Unit 2; Revision AH
- IR 694376; Cracked Weld on Connection to Condensate Header (2CDN1A); November 4, 2007

1R15 Operability Evaluations

- OP-AA-106-101-1106; Plant Issue Resolution Documentation, 1RY025, 1C Pressurizer Spray Valve Manual Isolation Valve; October 25, 2007
- OP-AA-108-111; Adverse Condition Monitoring and Contingency Plan – Monitoring Plan for Through Wall Leakage on the 2SX27DA10 SX Supply to 2A Diesel Generator; May 11, 2007
- IR 680720; Follow Up Actions for IR 628310, SX System Through Wall Leak; October 5, 2007
- IR 682624; 2SX27DA Leak Rate Has Changed; October 10, 2007
- IR 683765; 0PR33J Local Disconnect Breaker Not Turning On; October 12, 2007
- IR 683786; Surveillance Requirement (SR) 3.7.10.3 Question; October 12, 2007
- EC 367973; Install Mechanical Device on Valve 1RY025 to Prevent Rotation of Valve Stem; Revision 0
- IR 688891; 1RY455C Failed to Reduce Pressure While Stroking Open; October 24, 2007
- M – 60, sheet 5; Diagram of Reactor Coolant; Revision AO
- IR 691515; Potential Trend with Air Leaks on 1CV8141D; October 30, 2007
- IR 372831; 1CV8141D Air Line Has a Leak; September 13, 2005
- IR 479918; 1CV8141D Failed to Close When Depressurizing Unit 1 Reactor Coolant System; April 18, 2006
- IR 678515; 1CV8141D Air Line Broken; October 1, 2007
- WO 612772; 1CV8141D-Replace Actuator Diaphragm, Air Regulator, Air Supply Hose; April 28, 2006
- 1BwOA RCP-1; Reactor Coolant Pump Seal Failure Unit 1; Revision 101
- Quick Human Performance Investigation Report - New 1A Diesel Generator Voltage Regulator Not Working; October 17, 2007
- IR 686106; New 1A Diesel Generator Voltage Regulator Not Working; October 17, 2007
- IR 686160; 1A Diesel Generator Evaluate the Generator/Exciter for Overvoltage; October 18, 2007
- IR 510293-18; Perform Walkdown of Main Steam System in Main Steam Isolation Valve (MSIV)/ Feedwater Isolation Valve (FWIV) Rooms; January 12, 2007
- IR 579644; 1MS013D Install Insulation Missing From the Valve and Piping; January 16, 2007
- IR 643267; MSIV Rooms – Clarification of Insulation Requirements; June 22, 2007

- IR 689141; New Insulation Installed on the Unit 1 Main Steam Safety Valves; October 25, 2007
- IR 709590; Work Order Required for Augmented Inspection of Line 0SX01CE-30”; December 10, 2007
- IR 709700; Timeliness of Technical Requirements Manual TLCO 3.4.F Required Actions; December 10, 2007
- ER-AA-335-009; Instrument Performance Report 2007-223 Ultrasonic Instrument; December 4, 2007
- Braidwood Extent of Condition Review of SX Piping Minimum Wall Issues as Requested by IR 68723
- Operability Evaluation; Through-Wall Leakage of Line 0SX01CF-30”; Revision 0
- Calculation BRDW-06Q-301; Flaw Evaluation of Braidwood SX Piping; Revision 1
- IR 536541; Metal Plate Falls into Lake Screen House SX Pit During Work on 0SX115F; September 26, 2006
- IR 706376; 0SX115F SX Suction Valve Pit Flooded and Overflowing; December 3, 2007
- Essential Service Water Headers Byron Station – Units 1 and 2 Braidwood Station Units 1 and 2; Architect and Engineer Data Table
- M – 42, sheet 1A; Diagram of Essential Service Water Units 1 & 2; Revision BH
- M – 946, sheet 8; Chemical Feed System Quill Design & Installation Details Essential Service Water (SX) Units 1 & 2; Revision B
- American Society of Mechanical Engineers (ASME) Code Case N-513-1; Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1

<u>1R17 Permanent Plant Modifications</u>

- 1BwGP 100-3T10; Digital Electro hydraulic (DEH)Testing; Revision 1
- EC 349301-4; Turbine Overspeed Trip test; Revision 0
- OP-AA-106-101-1006; Nuclear Instrumentation System (NIS) Limiter; Revision 3
- Project Review – Non Outage Work Review
- EC 349301; Incorporation of Grid Frequency Response Function and Setpoint Change for System Frequency Low Relay 0SSL-SY077 for DEH Upgrade; Revision 1

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

- IR 686106; New 1A Diesel Generator Voltage Regulator Not Working; October 17, 2007
- IR 686160; 1A Diesel Generator Evaluate the Generator/Exciter for Overvoltage; October 18, 2007
- WO 965852-05; Verify Control Circuit Functions Properly for 1DG01KA
- WO 965852-06; 1A Diesel Generator Voltage Regulator Adjustments
- 1BwOS FW-M3; Unit 1 Turbine Driven Feedwater Pumps Mechanical Overspeed Trip Surveillance; Revision 14
- IR 698051; 1PI-FW456 – Low Oil Trip Header Pressure For 1B TDFWP; November 12, 2007
- IR 700931; 1C Main Feedwater Pump Auto Stop Oil Pressure Low; November 17, 2007
- IR 702188; 1FW01PB Reset Light Failed to Go Off During 1BwOS FW-M3; November 21, 2007
- IR 1BwOS ATWS-SA1 Surveillance Problem Noted; November 28, 2007
- 1BwOS ATWS-SA1; Unit One ATWS Mitigation System Surveillance; Revision 8
- WO 978455; 2SX007 Motor Operated Valve Diagnostic Test; Revision 3
- 2BwOSR 5.5.8.SX-4; Unit 2 Component Cooling Heat Exchanger Essential Service Water Outlet Valve Stroke Surveillance; Revision 0

- 2BwOSR 5.5.8.SX-5; Unit 2 Component Cooling Heat Exchanger Essential Service Water Outlet Valve Indication Surveillance; Revision 0
- WO 1053959; ASME Requirements for Essential Service Water Valves; Revision 1
- 1BwOSR 3.3.2.6-610A; Unit One ESFAS Instrumentation Slave Relay Surveillance (Train A Automatic Safety Injection – K610); Revision 4
- 1BwOSR 5.5.8.SX-1A; Stroke Test of 1SX147A, Containment Chiller 1A Essential Service Water Bypass Valve; Revision 6

1R20 Outage Activities

- EC360475; Braidwood Unit 1 Pressurizer Weld Over Lay; Revision 0
- 10 CFR 50.55a Relief Request 12R-48; Request for Relief for Alternate Requirements of Structural Weld Overlays (SWOLs) of the Pressurizer Surge, Spray, Safety and Relief Nozzles, Dissimilar Welds including the SWOLs of the adjacent Safe-End to Pipe, Reducer and Elbow Welds on Pressurizer Surge, Spray, Safety and Relief Nozzles In Accordance with 10 CFR 50.55a (a)(3)(i); Revision 0
- Confirmatory Action Letter NRR-07-008; Braidwood Station, Unit 1 and 2; March 22, 2007
- HU-AA-1211; Heightened Level of Awareness Brief -Reactor Lower Core Barrel Removal; October 7, 2007
- IR 652868; 1D Auxiliary Feedwater is 336 F. Higher Than Expected; July 23, 2007
- IR 677961; 1BwOA SEC-1 Entry Due to 1C Feedwater Pump Trip; October 1, 2007
- IR 678880; Boric Acid Leak/Stuck Open 1RC8042C; October 2, 2007
- IR 679564; Liquid Penetrant Test Indication Discovered on Base Metal Exam on “C” Nozzle; October 3, 2007
- IR 679792; Liquid Penetrant Test Indication Discovered on the “A” Safety Nozzle;
- October 4, 2007
- IR 680890; Transfer System Panel Knife Switch Found In Off Position; October 6, 2007
- IR 683625; NRC Observations During Unit 1 Containment Walkdown/Inspection
- October 11, 2007
- IR 683701; 1A Diesel Generator Air Compressor Catastrophically Failed, Belt Created Smoke; October 12, 2007
- IR 683956; NRC Found Contractor Inattentive in Containment; October 12, 2007
  IR 684488; Elongation Measurement 1D Steam Generator Secondary Manway; October 14, 2007
- IR 684348; 1D Secondary Manway Not Installed In Accordance with Procedure; October 13, 2007
- IR 685598; 1AF014D Check Valve Failed As Found Checks; October 16, 2007
- IR 685600; 1AF01H Leak Check Failed on Check Valve Removed From System; October 16, 2007
- IR 687278; Wear On Unit 1 Thimble Tube Number 44 at Location R-8; October 20, 2007
- IR 689206; White Substance Identified on Control Rod Drive Mechanism During Mode 3 Walkdown; October 25, 2007
- IR 689216; Debris Found in Unit 1 In-Core Sump During Mode 3 Walk Down, October 25, 2007
- IR 720481; NRC Severity Level IV NCV on TS 5.2.2.D (Overtime rules); January 10, 2008
- NF-CB-07-139; Braidwood Unit 1 Cycle 14 Safety Assessment; August 10, 2007
- OU-AA-103; A1R13 Shutdown Safety Management Plan, September 21, 2007
- 1BwGP 100-1; Plant Heatup; Revision 23
- 1BwGP 100-1T2; Mode 5 to 4 Checklist; Revision 21
- 1BwGP 100-2; Plant Startup; Revision 27
- BwVS 500-6; Low Power Physics Test Program; Revision 20

- 1BwGP 100-5; Plant Shutdown and Cooldown; Revision 31
- 1BwGP 100-5T1; 1BwGP 100-5 Flow Chart; Revision 16
- 1BwGP 100-4T1; 1BwGP 100-4 Flow Chart; Revision 13
- 1BwGP 100-4; Power Descension; Revision 26
- M – 37; Diagram of Auxiliary Feedwater Unit 1; Revision BF
- BwMP 3100-008; Reactor Vessel Closure Head Installation; Revision 22
- BwMP 3100-009; Reactor Vessel Closure Head Removal; Revision 15
- IR 679971; Reactor Vessel Head Drop Analysis; October 4, 2007
- Enforcement Guidance Memorandum 07-006; Enforcement Discretion for Heavy Load Handling Activities; September 28, 2007

1R22 Surveillance Testing

- BwOP AF-7T1; Diesel Driven Auxiliary Feedwater Pump Operating Log; Revision 4
- BwOP AF-7; Auxiliary Feedwater Pump 2B Startup on Recirculation; Revision 28
- BwOP AF-8; Auxiliary Feedwater Pump 2B Shutdown; Revision 24
- IR 550500; 2AF01PB-K Diesel Tripped During Post Operations Testing; October 29, 2006
- IR 632473; 2B AF SX Booster Pump High Vibe Point; May 22, 2007
- 1BwOS PC-1A; Containment Air Lock Door Seals; Revision 1
- 1BwOSR 3.6.2.1-4; Unit One Primary Containment Type B Local Leakage Rate Tests of the Emergency Hatch Airlock Door Gasket Interspaces; Revision 6
- WO 923335-01; Emergency Hatch Airlock Strongback Type B LLRT; September 12, 2007
- 1BwOSR 3.8.1.19-1; 1A Diesel Generator ECCS Sequencer Surveillance; Revision 6
- 1BwOSR 3.8.1.11-1; 1A Diesel Generator Loss of ESF Bus Voltage with No Safety Injection Signal; Revision 5
- 1BwOSR 3.8.1.10-1; 1A Diesel Generator Full Load Rejection and Simulated Safety Injection in Conjunction with Undervoltage During Load Testing; Revision 6
- 1BwOSR 3.3.2.8-611A; Unit One ESFAS Instrumentation Slave Relay Surveillance (Train A Automatic Safety Injection – K611); Revision 5

1R23 Temporary Plant Modifications

- WO 1034286; Install TCCP EC 368526 on 2E Main Power Transformer; Revision 3
- EC 368259; Install Temporary Wiring to Eliminate Ground Causing Main Power Transformer Trouble Alarm; Revision 0
- EC 368526, Sketch 1; December 11, 2007
- 20E-2-4428B; Wiring Diagram Main Transformer 2E Cooling System; Revision H

1EP4 Emergency Action Level and Emergency Plan Changes

- Braidwood Station Annex of the Exelon Standardized Emergency Plan; Revisions 16, 17, 18, 19, and 20
- Exelon Standardized Emergency Plan; Revisions 17, 18, and 19

2OS1 Access Control to Radiologically Significant Areas

- IR 661057, 1AR11J Radiation Levels Near Alert Setpoint; August 15, 2007
- IR 678484, High radiation Area Briefing Given With Outdated Survey Maps; October 1, 2007
- IR 675501, Unnecessary Neutron Dose from Work on Incore drives; September 25, 2007
- IR 677962, Radiation Work Permit Settings are Incorrect When Logging In; October 10, 2007
- IR 680884, Dose Rates From Water in Refuel Cavity Higher Than Expected; October 6, 2007

- IR 681763, High Radiation Alarm When Moving Reactor Vessel Core Barrel; October 10, 2007
- IR 683956, NRC Found Contractor Inattentive in Containment; October 12, 2007
- IR 684107, NRC Found Second Contractor Inattentive in Containment; October 12, 2007
- IR 686559, Core Barrel Lift Affects AR11 and 12 Detectors; October 11, 2007
- IR 661209, Nuclear Oversight Identified Three Radiation Protection Areas; August 16, 2007

2OS2 As Low As Is Reasonably Achievable Planning And Controls

- IR 65375, Pre-outage Focused Area Self-Assessment Deficiency - Dose Goal At Risk; July 27, 2007
- IR 654812, Pre-outage Assessment Recommendations; July 28, 2007
- IR 665228, Radiation Protection Department Briefing Multiple Jobs in One ALARA Brief; August 30, 2007
- IR 673446, ALARA Estimate for Unit 1 Equipment Hatch LLRT Underestimated; September 20, 2007
- IR 676346, ALARA Plans for Pressurizer Weld Overlay Project Not Approved; September 27, 2007
- IR 679709, Inaccurate Work Estimates/Scheduling; October 4, 2007
- ASSA 561153-04, Boiling Water Reactor and Pressurized Water Reactor Shutdown Survey Process and Data Evaluation Focused Area Self-Assessment; May 25, 2007
- ASSA 5607760, Braidwood Station unit 1 Cycle 13 Refuel Outage ALARA Preparations and Controls; August 23, 2007
- RP-AA-401; Operational ALARA Planning and Controls; Revision 7
- RWP 10008061, A1R13 Lower Internals Moves; Revision 7
- RWP 10008134; A1R13: Manway and Diaphragm Removal, Installation and Bolt Cleaning; Revision 1
- RWP 10008124; A1R13: ECCS Sump Screen Modification Work; Revision 1
- RWP 10008180; A1R13 ECCS Steam Generator Insulation Modification; Revision 0
- RWP 10008181; A1R13: Pressurizer Weld Overlay Project; Revision 0
- RWP 10008182; A1R13: Pressurizer Weld Overlay Insulation, Shielding and Support Activities (U-1 CNMT.); Revision 0

4OA1 Performance Indicator Verification

- BB PRA-017.27A; Reactor Oversight Process Mitigating Systems Performance Index (MSPI) Bases Document Braidwood Nuclear Generating Station; Revision 5; February 2007
- IR 639557; 2RH01PB Lower Motor Bearing Oil Discolored; June 12, 2007
- IR 654702; 1RH01PA Flow Trending Downward; July 27, 2007
- IR 681841; Valve 1RH014B Leaks By; October 9, 2007
- IR 688024; 1A RH Pump Discharge, 1CV8724A Difficult to Operate; October 23, 2007
- IR 680968; 1B Component Cooling (CC) Water Pump Trip; October 6, 2007
- IR 601437; MSPI System SX Defined to Have Low Margin to White Indicator; March 9, 2007
- IR 543636; MSPI Cooling Water System Function Entry Into Exelon Action Level; October 13, 2006

4OA2 Problem Identification and Resolution

- IR 710950; Secondary System Tritium Levels Above 20,000 PCI/L; December 13, 2007
- M-48, Sheet 15; Diagram of Waste Disposal Radwaste Monitor Tanks, Revision AI
- M-48, Sheet 4A; Diagram of Waste Disposal Blowdown Monitor Tanks, Revision AX
- M-48, Sheet 2; Diagram of Waste Disposal Steam Generator Blowdown, Revision BF

- M-48, Sheet 22; Diagram of Waste Disposal Turbine Bldg Equip. Drains, Revision U
- M-48, Sheet 3A; Diagram of Waste Disposal Blowdown Mixed Bed Demin, Revision AJ
- M-48, Sheet 1; Diagram of Waste Disposal 30,000 Gallon Release Tank, Revision AV
- M-48, Sheet 18; Diagram of Waste Disposal Resin Removal, Revision AQ
- List of Common Cause Assessments completed March 1, 2007 to October 15, 2007
- Nuclear Oversight Quarterly Report, Braidwood Station, NOSPA-BW-07-1Q, January through March 2007 Nuclear Oversight Quarterly Report, Braidwood, NOSPA-BW-07-2Q, April through June 2007
- IR 515009-02; Common Cause Analysis, Potential Trend – Equipment Failures Due to Severe Weather (Lightning); October 6, 2006
- System Health Overview Report, September 2007, Instrument Air Supply System
- IR 595136-02; Common Cause Analysis, Complete CCA for 2006 QRT Results; May 18, 2007
- IR 597695; Lightning Strike to Site Causes Alarms; March 1, 2007
- IR 597926; Lightning Strike to Site Causes Multiple Alarms; March 1, 2007
- IR 601656-02; Common Cause Analysis, Complete CCA for NOS Improvement Opportunities; April 13, 2007
- IR 621541; Security Active Vehicle Barrier Unexpected Activation; April 25, 2007
- IR 623101-03; Common Cause Analysis, Repeat Leakage following Maintenance; August 1, 2007
- IR 592284-04; Common Cause Analysis, conduct a CCA for Clearance and Tagging issues; April 20, 2007
- IR 644837; Reactor Containment Fan Cooler Essential Service Water Process Radiation Monitor 2PR03J Failure; June 27, 2007
- IR 649066; 6.9KV Bus Voltages to Be Checked a Minimum of Once per Shift; July 11, 2007
- IR 652712; Common Cause Analysis, Station Reactivity Management Performance; July 23, 2007
- IR 654138; Common Cause Analysis, Potential Adverse Trend due to failures on the Unit 2 Plant Process Computer; July 26, 2007
- IR 604553-06; Common Cause Analysis, Trend identified in Chemistry Department use of written instructions; April 30, 2007
- Braidwood Station Event Free Clock Resets
- IR 615956; Sea van inspections not meeting procedure requirement; April 11, 2007
- IR 620669; Could not inspect all sides to sea vans; April 23, 2007
- IR 620657; Cannot inspect bottoms of sea vans; April 23, 2007
- IR 628446; LORT Simulator OOB failure – crew 6; May 11, 2007
- IR 583435; Security door violation potential trend; January 25, 2007
- IR 583919; MMD resource requirement changes - trend; January 26, 2007
- IR 601656; Perform CCA for NOS improvement opportunities; March 9, 2007
- IR 623101; Perform CCA – additional improvement needed for leaks; April 29, 2007
- IR 610809; Adverse security human performance trend; March 30, 3007
- IR 584200; Adverse performance with work package quality; January 27, 2007
- IR 592284; Potential trend – clearance and tagging admin (maintenance); February 16, 2007
- IR 631214; E-4 to E-1 stability CCA – May 18, 2007 PARS meeting action; May 18, 2007
- IR 587986; Adverse trend with security keycard control issues; February 6, 2007
- IR 604553; Chemistry negative trend in use of written instructions; March 15, 2007
- IR 642087; Trend identified large number of RP IRS coded 5SL; June 19, 2007
- IR 657453-04; Review of Results of Westinghouse Enhanced Card Maintenance Following A1R13, Document Findings and Initiate Additional Actions as Required; November 9, 2007
- IR 663569; Historic Classification of Rod Drive Events in Maintenance Rule; August 23, 2007
- Braidwood Station's Maintenance Rule periodic (a)(3) Assessment, November 2005 – April 2007

- 1BwOSR 0.1-1,2,3; Unit One – Modes 1,2, and 3 Shiftly and Daily Operating Surveillance Data Sheet; Revision 46

4OA3 Followup of Events and Notices of Enforcement Discretion

- BwOP CW-26; Defeating Circulating Water Pump Trip on Traveling Screen Differential Pressure; Revision 0
- Licensee Event Report 05000457/2007-001-00; Unit 2 Manual Reactor Trip Following Circulating Water Pump Trips
- IR 663914-03; Root Cause Report for Reactor Trip Due to Loss of 2 Circulating Water Pumps; September 21, 2007
- EC 367591; Install Time Delay Relay in Lake Screen House Traveling Screen Control Panel to Prevent Spurious Circulating Water Pump Trips; Revision 0
- Licensee Event Report 05000456/2007-002-00; Unit 1 Power Range N43 Positive Rate Trip Inoperable Due to Miscalibration of Time Constant
- IR 674383; Root Cause Report for Instrument Maintenance Technicians Calculated Incorrect Data in 2006 From Recorder Trace that Resulted in N43 Power Range Instrument Being Inoperable for 18-Months; November 14, 2007
- IR 674383; B2 Trend Code 1NY-NR8043A Delay Time Constant Did Not Meet Technical Specifications; September 22, 2007
- IR 463827; B4 Trend Code: 1NY-NR8043A1/ NM311 Delay Out Of Tolerance High; March 9, 2006
- BwISR 3.3.1.11-207; Channel Verification/Calibration of Nuclear Instrumentation System Power Range N43; Revision 15

4OA5 Other Activities

- EC 358828; Replacement of the Unit 1 Containment Recirculation Sump Screens; Revision 1
- LS-AA-104-1003; 50.59 Screening Form – Replacement of the Containment Recirculation Sump Screens; Revision 1
- LS-AA-104-1004; 50.59 Evaluation Form – BRW-E-2007-58; Revision 0
- Work Plan Instructions EC 358828; Revision 1
- Design Consideration Summary Demolition Activities Related to GSI-191, Unit 1
- M – 61, Sheet 4; Diagram of Safety Injection Unit 1; Revision AY
- S – 1065A, Sheet 1; Containment Building Recirculation Building Recirculating Sump Screens Plans, Sections, and Details
- IR 682766; Gasket "Cupping" Issue Identified on Safety Injection System Flanges; October 10, 2007
- Apparent Cause Report; Safety Injection Flow Filler Plate 1SI106MD Not Installed; October 17, 2007

NRC and IEMA Identified Issues

- IR 679846; IEMA identified catch container not working; October 3, 2007
- IR 680169; Rust on supports in LCSR; October, 4, 2007
- IR 680177; Water in LCSR; October 4, 2007
- IR 680720; Clamp blocking leak on SX pipe; October 5, 2007
- IR 682765; Screw missing on elevator; October 10, 2007
- IR 683625; NRC observations during Unit 1 containment inspection; October 11, 2007
- IR 683956; NRC found contractor inattentive in containment; October 12, 2007
- IR 684017; IEMA identified precipitation computer point (49019) erratic; October 12, 2007

- IR 689018; IEMA question on actions from tarp blowing into switchyard; October 24, 2007
- IR 689556; A1R13 containment walkdown; October 25, 2007
- IR 691515; Potential trend with air leaks on 1CV814LD; October 29, 2007
- IR 692509; IEMA chain fall on support bracket for 1B SG PORV, October 31. 2007
- IR 695398; NRC identified IR 687180 not screened for past operability; November 6, 2007
- IR 695840; IEMA identified tarp flopping around in wind; November 7, 2007
- IR 695907; NRC and IEMA identified question with past operability; November 7, 2007
- IR 696295; NRC and IEMA identified inconsistencies with operability documentation; November 8, 2007
- IR 697128; Enhancement to AF MSPI basis document found; November 9, 2007
- IR 696786; IEMA identified inappropriate material storage near hydrant OFPNS; November 16, 2007
- IR 701120; OW0347 valve has incorrect information on label plate; November 18, 2007
- IR 702109; 2SI161 boric acid; November 20, 2007
- IR 702568; IEMA identified emergency lighting battery capacity at 25 to 50% level; November 21, 2007
- IR 702571; IEMA identified emergency lighting battery capacity at 25 to 50% level; November 21, 2007
- IR 703092; SEALITE separated from junction box (1TS-SI059); November 24, 2007
- IR 704566; IEMA identified emergency lights resolved unexpectedly; November 28, 2007
- IR 709175; Unit 2 accident monitoring instruments missing identities; December 8, 2007
- IR 709180; Unit 1 accident monitoring instruments missing identities; December 8, 2007
- IR 715334; IEMA identified mobile items stored in accordance with BWAP1100-23

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management system
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
AWG	American Wire Gauge
BACC	Boric Acid Corrosion Control
CFR	Code of Federal Regulations
DG	Diesel Generator
EPRI	Electric Power Research Institute
ET	Eddy Current Test
GL	Generic Letter
IMC	Inspection Manual Chapter
IR	Issue Report
ISI	Inservice Inspection
kV	Kilovolt
LER	Licensee Event Report
MEER	Miscellaneous Electrical Equipment Room
MSPI	Mitigating System Performance Index
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NOS	Nuclear Oversight
NRC	U.S. Nuclear Regulatory Commission
PT	Dye Penetrant Examination
RH	Residual Heat Removal
RI-ISI	Risk-Informed Inservice Inspection Program
RVUHP	Reactor Vessel Upper Head Penetration
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Steam Generator
SI	Safety Injection
SX	Essential Service Water
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VC	Control room Ventilation