January 26, 2004

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000456/2003008;
05000457/2003008

Dear Mr. Skolds:

On December 31, 2003, 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 findings which were discussed on January 7, 2004, with Mr. M. Pacilio and other members of your staff.

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

Based on the results of this inspection, one NRC-identified finding of very low safety significance which involved a violation of NRC requirements was identified. However, because the violation was of very low safety significance and because the issue was entered into the licensee's corrective action program, the NRC is treating the finding as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, two licensee-identified violations are listed in Section 4OA7 of this report.

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, 801 Warrenville Road, Lisle, IL 60532-4351; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Braidwood facility.

In accordance with 10 CFR 2.790 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). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Branch 3
Division of Reactor Projects

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

Enclosure: Inspection Report 05000456/2003008 05000457/2003008
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Braidwood
Braidwood Station Plant Manager
Regulatory Assurance Manager - Braidwood
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
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REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report No:	05000456/2003008; 05000457/2003008
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	35100 S. Route 53 Suite 84 Braceville, IL 60407-9617
Dates:	October 1 through December 31, 2003
Inspectors:	S. Ray, Senior Resident Inspector N. Shah, Resident Inspector T. Bilik, Reactor Engineer R. Daley, Reactor Engineer M. Holmberg, Reactor Engineer R. Jickling, Emergency Preparedness Analyst D. Jones, Reactor Engineer D. McNeil, Reactor Engineer D. Nelson, Radiation Specialist C. Phillips, Senior Operations Examiner P. Smith, Illinois Emergency Management Agency D. Tharp, Reactor Engineer T. Tongue, Project Engineer
Observers:	L. Haeg, Reactor Inspector
Approved by:	Ann Marie Stone, Chief Branch 3 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000456/2003008, 05000457/2003008; 10/01/03 - 12/31/03; Braidwood Station, Units 1 & 2; Operability Evaluations.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on inservice inspection activities, licensed operator requalification program, emergency action level and emergency plan changes, and outage radiological access control and as low as reasonably achievable exposure programs. In addition, inspections were conducted using Temporary Instructions (TI) 2515/150, Revision 2; TI 2515/152, Revision 1; TI 2515/153; and TI 2515/154. The inspection was conducted by Region III inspectors and the resident inspectors. One Green finding associated with a non-cited violation was identified. 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 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Green. The inspectors identified a Non-Cited Violation of Criteria XVI of 10 CFR 50, Appendix B, having very low safety significance (Green) for failing to identify and correct a condition adverse to quality. Specifically, the licensee failed to recognize that the containment atmosphere radiation gaseous monitors were inoperable when it was determined that the monitors were not capable of detecting reactor coolant leakage in a reasonable period of time. The finding also affected the cross-cutting area of Problem Identification and Resolution because the issue was discovered by the licensee’s staff; however, it was not adequately resolved until questioned by the NRC inspectors.

The finding was greater than minor because the finding was associated with the barrier integrity cornerstone and, if left uncorrected, could result in an undetected reactor coolant system leak. The finding was determined to be of very low safety significance by management review because alternate methods of detecting small reactor coolant system leaks were available. The licensee’s corrective actions included declaring the monitor inoperable and submitting a technical specification change request. (Section 1R15)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

Unit 1 was operated at or near full power until December 31, 2003, when power was reduced to 29 percent to complete repairs and post-maintenance testing on a feedwater isolation valve hydraulic operator problem.

Unit 2 was operated at or near full power until it began a gradual coastdown on October 15, 2003. On November 3, Unit 2 reached 84 percent and was then shut down for a refueling outage. Unit 2 was brought critical on November 18, the generator was placed on line on November 19, and the unit reached full power on November 21, 2003. Unit 2 tripped from full power on December 3, 2003, due to a dual feed pump trip. The reactor was returned to criticality on December 4, was synchronized to the grid on December 5, and reached full power on December 7, 2003. Unit 2 was operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

Preparation for Seasonal Susceptibilities

a. Inspection Scope

The inspectors verified that the licensee had completed its seasonal preparations for cold weather in a timely manner before the cold weather actually presented a challenge. The inspectors reviewed the licensee's completed freezing temperature annual surveillance, as well as other documents listed in the Attachment, and verified that it adequately covered risk-significant equipment and ensured that the equipment was in a condition to meet the requirements of Technical Specifications (TS), the Technical Requirements Manual, and the Updated Final Safety Analysis Report (UFSAR) with respect to protection from low temperatures. This review constituted one sample of this inspection requirement. As part of this inspection, the inspectors performed walkdowns of the following risk-significant systems:

- Units 1 and 2 refueling water storage tanks and the associated heating systems and power supplies;
- Units 1 and 2 condensate storage tanks and the associated heating systems and power supplies; and
- heat trace control panels and associated power supplies.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

Partial Walkdowns

a. Inspection Scope

The inspectors performed a partial walkdown of the accessible portions of the Unit 1 feedwater train. The inspectors utilized the valve and electric breaker checklists, as well as other documents listed in the Attachment, to verify that the components were properly positioned and that support systems were lined up as needed. The inspectors also examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors reviewed outstanding work orders (WOs) and condition reports (CRs) associated with the train to verify that those documents did not reveal issues that could affect train function. The inspectors used the information in the appropriate sections of the TS and UFSAR to determine the functional requirements of the system. The inspectors also reviewed the licensee's identification of and the controls over the redundant risk related equipment required to remain in service. This walkdown constituted one sample of this inspection requirement.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

Quarterly Area Walkdowns

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of fire fighting equipment; the control of transient combustibles and ignition sources; and on the condition and operating status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate a plant transient, or their impact on the plant's ability to respond to a security event. The inspectors used the documents listed in the Attachment to verify 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 that fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

The inspectors completed four samples of this inspection requirements during the following walkdowns:

- auxiliary building fire zone 11.3-0;
- fuel handling building fire zone 12.1-0 (for this zone, the inspectors also verified that all fire hoses had solid stream vice fog nozzles attached);
- Unit 2 containment (multiple fire zones); and
- Unit 2 auxiliary building (multiple fire zones).

The Unit 2 containment and auxiliary building walkdowns were performed during the refueling outage, during periods of ongoing hotwork (i.e., welding, grinding, etc).

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

Semi-Annual Inspection of Internal Flood Protection Features

a. Inspection Scope

The inspectors reviewed the licensee documents listed in the Attachment and conducted walkdowns of risk significant areas of the plant to verify that measures to detect and/or prevent internal flooding were being adequately maintained. The inspectors also reviewed the last performance of preventive maintenance or testing activities on the measures. In addition, the inspectors reviewed the licensee’s corrective action data base to verify that no adverse trends or significant deficiencies existed with the measures. The inspectors verified that minor issues identified during the inspection were entered into the licensee’s corrective action system. The inspectors completed two samples of this requirement by inspecting the following features:

- leak detection sumps in the auxiliary building, and
- watertight doors and floor drains.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

a. Inspection Scope

The inspectors conducted a review of the implementation of the licensee’s inservice inspection program for monitoring degradation of the reactor coolant system (RCS) boundary and the risk significant piping system boundaries.

Specifically, the inspectors conducted an onsite and record review of the following nine nondestructive examination activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME Code: This review counted as two samples.

- visual examination of the Unit 2 reactor pressure vessel lower head penetration (LHP) (bottom mounted instrumentation) nozzles: 3, 12, 37, and 47;
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 2 (2RC-32-02);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 3 (2RC-32-03);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 4 (2RC-32-04);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 5 (2RC-32-05);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 11 (2RC-32-11);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 13 (2RC-32-13);
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 16 (2RC-32-16); and
- ultrasonic examination of Unit 2 RCS (pressurizer) weld 17 (2RC-32-17).

The inspectors also reviewed the following examination from the previous outage with recordable indications that have been accepted by the licensee for continued service to verify that the licensee's acceptance for continued service was in accordance with the ASME Code: (This review counted as one sample.)

- ultrasonic examination of a safety injection (SI) system 1SI-35-10 cap-pipe weld (indications found to be acceptable).

The inspectors determined that there were no pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous refueling outage, to verify that the welding acceptance (e.g., radiography) and preservice examinations were performed in accordance with ASME Code requirements. Therefore, no inspection samples were completed.

The inspectors reviewed one ASME Section XI Code replacement to verify that the replacement met ASME Code requirements. This review counted as one sample.

- removal and replacement of a Code Class 2, Unit 1 residual heat removal system, letdown booster pump.

The inspectors reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program to assess conformance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the Inservice Inspection group.

The inspectors also confirmed that the steam generator tube eddy current testing (ECT) scope and expansion criteria met TS requirements, Electric Power Research Institute guidelines, and commitments made to the NRC; confirmed that all areas of potential degradation (based on site-specific experience and industry experience) were inspected, especially areas which are known to represent potential ECT challenges (e.g., top-of-tubesheet, tube support plates, U-bends); confirmed that the ECT probes and equipment were qualified for the expected types of tube degradation; assessed the site specific qualification of one or more techniques (e.g., equipment, data quality/noise issues, degradation mode); assessed corrective actions for loose parts or foreign material discovered on the secondary side of the steam generator; and reviewed the following eddy current data because questions arose regarding eddy current data analyses: This review counted as one sample.

- steam generator 21, row 1, column 3; and
- steam generator 21, row 1, column 7.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Written Examination and Operating Test Results

a. Inspection Scope

The inspectors reviewed the pass/fail results of individual written tests, operating tests, and simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee during calender year 2003. This activity completes the sample discussed in Inspection Report 05000456/2003006; 05000457/2003006.

b. Findings

No findings of significance were identified.

.2 Quarterly Review of Testing/Training Activity

a. Inspection Scope

The inspectors observed the operating crew response during the simulator portion of the emergency preparedness drill. The inspectors evaluated crew performance in the following areas:

- clarity and formality of communications;
- ability to take timely actions in the safe direction;
- prioritization, interpretation, and verification of alarms;
- procedure use;
- control board manipulations;
- oversight and direction from supervisors; and
- group dynamics.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in the Exelon procedures listed in the Attachment. The inspectors verified that the crew completed the critical tasks listed in the simulator guide. The inspectors also compared simulator configurations with actual control board configurations. For any weaknesses identified, the inspectors observed the licensee evaluators to verify that they also noted the issues and discussed them in the critique at the end of the session. This inspection constituted one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

Routine Inspection

a. Inspection Scope

The inspectors reviewed the licensee's overall maintenance effectiveness for risk-significant mitigating systems. This evaluation consisted of the following specific activities:

- observing the conduct of planned and emergent maintenance activities where possible;
- reviewing selected CRs, open WOs, and control room log entries in order to identify system deficiencies;
- reviewing licensee system monitoring and trend reports;
- a partial walkdown of the selected system; and
- interviews with the appropriate system engineer.

The inspectors also reviewed whether the licensee properly implemented the Maintenance Rule, 10 CFR 50.65, for the system. Specifically, the inspectors determined whether:

- the system was scoped in accordance with 10 CFR 50.65;
- performance problems constituted maintenance rule functional failures;
- the system had been assigned the proper safety significance classification;
- the system was properly classified as (a)(1) or (a)(2); and
- the goals and corrective actions for the system were appropriate.

The above aspects were evaluated using the maintenance rule program and other documents listed in the Attachment. The inspectors also verified that the licensee was appropriately tracking reliability and/or unavailability for the systems.

The inspectors completed two samples in this inspection requirement by reviewing the following systems:

- Units 1 and 2 main feedwater systems, and
- Units 1 and 2 engineered safety features actuation system.

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 management of plant risk during emergent maintenance activities or during activities where more than one significant system or train was unavailable. The activities were chosen based on their potential impact on increasing the probability of an initiating event or impacting the operation of safety-significant equipment. The inspections were conducted to verify that evaluation, planning, control, and performance of the work were done in a manner to reduce the risk and minimize the duration where practical, and that contingency plans were in place where appropriate.

The licensee's daily configuration risk assessments records, observations of operator turnover and plan-of-the-day meetings, observations of work in progress, and the documents listed in the Attachment were used by the inspectors to verify that the equipment configurations were properly listed, that protected equipment was identified and were controlled where appropriate, that work was being conducted properly, and that significant aspects of plant risk were being communicated to the necessary personnel. The inspectors verified that the licensee controlled emergent work in accordance with the expectations in the procedures listed in the Attachment.

In addition, the inspectors reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance.

The inspectors completed three samples by reviewing the following activities:

- an unplanned out-of-service of direct current bus 111 due to unexpected low battery charger voltage concurrent with an unplanned out-of-service of the 2A chemical and volume control pump due to the unavailability of its cubicle cooler fans;
- an unplanned out-of-service of the engineered safety feature sequencing logic for the 1B emergency diesel generator due to an unexpected failure of an Eagle relay timer during surveillance testing; and
- an unplanned trip of the 2B and 2C turbine-driven feedwater pumps which resulted in a Unit 2 reactor trip and the subsequent recovery efforts.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors completed three samples by reviewing the control room operator response to the following events:

- an unplanned transfer of water from the Unit 2 refueling water storage tank to the spent fuel pool;
- a trip of the 2C turbine-driven feedwater pump followed by a trip of the 2B turbine-driven feedwater pump from full reactor power that resulted in a Unit 2 reactor trip on 2D steam generator low-low water level, (this event is also discussed in Section 4OA3.5 of this report); and
- an unscheduled power reduction from 100 percent to 29 percent on Unit 1 for repair and testing of a feedwater isolation valve.

For each of these events, the inspectors, as applicable, observed control room activities, interviewed plant operators and/or other personnel, and reviewed plant records including control room logs, operator turnovers and condition reports. The inspectors verified that personnel errors contributing to the events were identified and entered into the licensee corrective action program, that the operator response to the events was in accordance with the applicable plant procedures, and that plans, briefings, and contingency plans were adequate for preplanned non-routine evolutions. Documents reviewed as part of this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors evaluated plant conditions and selected CRs for risk-significant components and systems in which operability issues were questioned. These conditions were evaluated to determine whether the operability of components was justified. The inspectors compared the operability and design criteria in the appropriate section of the UFSAR to the licensee's evaluations presented in the CRs and documents listed in the Attachment to verify that the components or systems were operable. The inspectors also conducted interviews with the appropriate licensee system engineers to obtain further information regarding operability questions.

The inspectors completed three samples by reviewing the following operability evaluations and conditions:

- CR 179714, "Westinghouse Nuclear Safety Advisory Letter 03-09–U2 Steam Generator Level Uncertainties";

- CR 183954, “Failed Fuel Canister Located In Spent Fuel Pool Region II Rack”; and
- CR 188384, “Repeat Maintenance–2AF014A Check Valve Leakage.”

In addition, the inspectors reviewed additional information with respect to Unresolved Item (URI) 05000456/02-02-03; 05000457/02-02-03, “Ability of Radiation Monitors to Detect RCS Leakage.” This review was not considered an inspection sample.

b. Findings

Introduction: The inspectors identified a Non-Cited Violation (NCV) of Criteria XVI of 10 CFR 50 Appendix B having very low safety significance (Green) for failing to identify and correct a condition adverse to quality. Specifically, the licensee failed to recognize that the containment atmosphere radiation gaseous monitors were inoperable.

Description: As previously discussed in Inspection Report 05000456/02-02; 05000456/02-02, Section 1R15, in January 2002, a member of the Byron station staff identified a non-conservative error for the containment atmosphere radiation monitor (1/2PR11J) setpoint at both Braidwood and Byron. As documented in CR 89364, the RCS activities used to calculate the 1 gallon per minute (gpm) leak rate were substantially more than the existing RCS activities. That error affected the monitor’s ability to detect a 1 gpm leak from the RCS within 1 hour. For example, the assumed Xe-135 concentration was 1.26 curies per gram (Ci/gm) and the actual concentration was 1.30 E-3 Ci/gm which was roughly a factor of 1000 lower. The licensee immediately evaluated the condition and in CR 89583, stated that there was no operability concern because the monitors met the TS surveillance requirements; however, additional review was necessary.

Regulatory Guide 1.45 "Reactor Coolant Pressure Boundary Leakage Detection Systems," stated that, "In analyzing the sensitivity of leak detection systems...a realistic primary coolant radioactivity concentration assumption should be used. The expected values used in the plant environmental report would be acceptable." As stated in the UFSAR, Appendix A, the licensee was committed to Regulatory Guide 1.45, with the caveat that leak detector sensitivity was as low as practicable. The licensee confirmed that the radiation monitor setpoints were not based on actual RCS activity values but were realistic RCS activities as allowed by Regulatory Guide 1.45. The licensee also stated that the monitors were original equipment and that no modifications had been done to change the characteristics of the detectors. The licensee concluded that the particulate and gaseous monitors were operable because the monitors met their design basis and could detect a 1 gpm leak within 1 hour at the reactor coolant activities specified in the plant environmental report. The licensee planned to clarify the TS bases and UFSAR to reflect the actual capabilities of the monitors and define other available means to detect leakage.

The inspectors noted that TS 3.4.15 required either the gaseous or particulate containment atmosphere radioactivity monitor be operable. The bases for this TS stated, in part, that “radioactivity detection systems shall be operable to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the unit in a safe condition, when RCS leakage indicated a possible reactor coolant pressure boundary degradation." The bases for another technical specification, TS 3.4.13, further stated that 1 gpm of unidentified leakage was allowed as a reasonable minimum detectable amount that the containment air monitoring system could detect within a reasonable time period. The inspectors questioned whether the 1/2PR11J containment atmosphere radiation monitors were technically operable because at current activity levels, a 1 gpm RCS leak would be detected by the gaseous containment atmosphere radiation monitors in 223 to 839 hours. The inspectors opened the unresolved item pending further review by the NRC Office of Nuclear Reactor Regulation.

On February 20, 2003, the NRC concluded that the gaseous monitor sensitivity did not meet the bases for TS 3.4.13 and therefore, was not sufficient to support the leak before break monitoring assumptions. The licensee declared the gaseous monitors inoperable.

In a letter dated August 15, 2003, the licensee requested an amendment to TS 3.4.15, to remove reference to the gaseous monitor and provided justification for the operability of the particulate monitor. At the issuance of this inspection report, this amendment was in the review process.

Analysis: The inspectors determined that failing to identify that the containment atmosphere radiation gaseous monitors were inoperable was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with IMC 0612, "Power Reactor Inspections Reports," Appendix B, "Issue Disposition Screening." The finding was associated with the reactor coolant system barrier and if left uncorrected, could result in an undetected reactor coolant system leak. The finding also affected the cross-cutting area of Problem Identification and Resolution because the issue was discovered by the licensee's staff; however, it was not adequately resolved until questioned by the NRC inspectors.

The inspectors determined that the finding could not be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," because the SDP for the RCS barrier only applied to a degraded barrier, not the ability to detect a degraded barrier. Therefore, this finding was reviewed by the regional branch chief in accordance with IMC 0612, Section 05.04c, and determined to be of very low safety significance (Green) because alternate methods of detecting small RCS leaks were available and no actual RCS leak had occurred. The finding was assigned to the barrier integrity cornerstone for both units.

Enforcement: Criterion XVI of 10 CFR 50, Appendix B, stated, in part, that measures shall be established to assure that conditions adverse to quality, such as non-conformances are promptly identified and corrected. Contrary to the above, in January 2002, the licensee failed to identify that the gaseous containment atmosphere radiation monitors were inoperable when the licensee determined that the monitors were not capable of detecting a 1 gpm leak within 1 hour or a reasonable period of time as specified in TS Bases 3.4.15 and 3.4.13. This violation was characterized as having very low risk significance (i.e., Green) and is being treated as a Non-Cited Violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000456/2003008-01; 05000457/2003008-01). This violation is in the licensee's corrective action program as CR 85983. The associated URI (05000456/02-02-03; 05000457/02-02-03) is closed.

1R16 Operator Workarounds (71111.16)

.1 Review of Selected Potential Operator Workaround

a. Inspection Scope

While reviewing a previous configuration control event, the inspectors became aware that the design of the component cooling (CC) subsystem supplying the containment penetration coolers was such that it might be considered an operator workaround. Specifically, whenever a second CC water pump was started on a unit, such as would happen during a SI, the CC supply to the containment penetration coolers would isolate on high flow, causing a main control board annunciator and requiring manual operator action in the field to reset the isolation valve.

The inspectors reviewed the UFSAR, TS, and the documents listed in the Attachment, as well as interviews with operators, and determined that the issue was not an operator workaround. This was because the system performed as designed and, since resetting of the isolation valve was not required to be completed rapidly to maintain operability of any mitigating or barrier integrity equipment, it would not significantly complicate the operators response to an event. This inspection was considered to be one sample of the operator workarounds inspection requirement.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Review of the Cumulative Effect of Operator Workarounds

a. Inspection Scope

On October 23, 2003, the inspectors completed a semi-annual review of the cumulative effects of operator workarounds. The inspectors verified that the workarounds did not have a significant effect on the reliability, availability, or the ability to correctly operate mitigating systems and that they would not significantly increase operator response time to transients and accidents. The inspectors also verified that the licensee had plans and schedules established to correct the conditions in a reasonable time. In addition to operator workarounds, the inspectors reviewed operability evaluations, operator challenges, troubleshooting plans, equipment status tags, and temporary modifications for cumulative effects. The inspectors reviewed the documents listed in the Attachment as part of this inspection. This inspection constituted one sample of the semi-annual requirement.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

Annual Review

a. Inspection Scope

The inspectors evaluated the permanent plant modification installed under Engineering Change 344306 and 344307 to relocate the 1B and 2B auxiliary feedwater (AF) pump diesel governor oil reservoirs. This modification was installed to correct a problem which licensee engineering personnel believed caused some failures of the 1B AF diesel engine to start in the last several years. This modification was chosen because it involved a highly risk-significant mitigating system which had experienced performance problems leading to the performance indicator for Unit 1 heat removal system unavailability being in the White band. The modification was performed on the 1B AF diesel with the unit on line and on the 2B AF diesel during the refueling outage.

As part of this inspection the inspectors reviewed the design change packages and associated work orders for installation, observed portions of the installations, and observed the post-modification testing. The inspectors verified that the modification did not appear to introduce any new system vulnerabilities, did not create any new system interface problems, and that the testing verified that the system performed as designed with the most conservative initial conditions expected. The inspectors also reviewed the licensee's plan for returning the 1B AF train from an accelerated testing frequency to the normal frequency. The inspectors verified that minor issues identified by the licensee and the NRC during the modification were entered into the licensee's corrective action program. Documents reviewed as part of this inspection are listed in the Attachment. This activity constituted one inspection sample of the annual requirement.

b. Findings

No findings of significance were identified.

1RST Post-Maintenance and Surveillance Testing - Pilot (71111.ST)

a. Inspection Scope

The inspectors reviewed post-maintenance and surveillance testing activities associated with important mitigating, barrier integrity, and support systems to ensure that the testing adequately verified system operability and functional capability. For post-maintenance testing, the inspectors used the appropriate sections of the TS and UFSAR, as well as the work orders for the work performed, to evaluate the scope of the maintenance and to verify that the post-maintenance testing was performed adequately, demonstrated that the maintenance was successful, and that operability was restored. For surveillance testing, the inspectors verified that the testing met the TS, the UFSAR, and licensee procedural requirements, and demonstrated that the equipment was capable of performing its intended safety functions. The inspectors verified that the testing met the frequency requirements; that the tests were conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the tests were properly reviewed and recorded. The activities were selected based on their importance in verifying mitigating systems capability and barrier integrity. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action system. Documents reviewed as part of this inspection are listed in the Attachment.

Note that this inspection is a pilot for a proposed consolidated procedure combining the previous Post-Maintenance Testing (71111.19) and Surveillance Testing (71111.22) procedures.

Four samples were completed by observing post-maintenance testing after the following activities:

- replacement of the Unit 2 centrifugal charging pump discharge check valves;
- replacement of the Unit 2 CC heat exchanger tubes;
- replacement of the 2B AF essential service water cubicle cooler valves; and
- replacement of the 2B AF diesel blower.

Six samples were completed by observing and evaluating the following surveillance tests:

- 2B containment spray additive flow rate verification test;
- 1B diesel generator slave relay start and semi-annual fast start surveillance;
- 2PT-455A cold overpressure surveillance;
- 2B diesel generator sequencer test;
- 1B AF pump monthly test; and
- 1B containment spray pump ASME test.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed the licensee's performance during the Unit 2 refueling outage conducted between November 4 and November 22, 2003. This inspection constituted one sample of the inspection requirement.

This inspection consisted of a review of the licensee's outage schedule, safe shutdown plan and administrative procedures governing the outage, periodic observations of equipment alignment, and plant and control room outage activities. Specifically, the inspectors determined whether the licensee effectively managed elements of shutdown risk pertaining to reactivity control, decay heat removal, inventory control, electrical power control, and containment integrity.

The inspectors performed the following activities daily, during the outage:

- attended control room operator and outage management turnover meetings to verify that the current shutdown risk status was well understood and communicated;
- performed walkdowns of the main control room to observe the alignment of systems important to shutdown risk;
- observed the operability of RCS instrumentation and compared channels and trains against one another;
- performed walkdowns of the auxiliary and containment buildings to observe ongoing work activities; and
- reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance, and that operability issues were resolved prior to startup.

Additionally, the inspectors performed the following specific activities:

- reviewed the detailed outage schedule and risk control plans;
- attended a licensee lessons learned training session regarding industry fuel handling events;
- observed the control room staff perform the Unit 2 shutdown and initial cooldown;
- observed the licensee's align the residual heat removal system for shutdown cooling;
- observed the control room staff drain the reactor vessel to the flange;
- performed a walkdown of the Unit 1 and 2 spent fuel cooling system in preparation for fuel unloading;
- observed the Unit 2 fuel unloading, fuel shuffles and reloading;
- observed the material condition of the Unit 2 containment emergency core cooling system sumps;
- performed a closeout inspection of the Unit 2 containment (as part of this inspection, the inspectors verified that all discrepancies observed were properly recorded and corrected); and
- observed portions of the low power physics testing, the approach to criticality, and portions of the power ascension.

During the routine walkdowns, the inspectors selectively verified that equipment configuration was appropriately maintained and that redundant equipment was available when maintenance was occurring on plant systems. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action system. Documents reviewed during these inspection activities are listed in the Attachment.

b. <u>Findings</u>

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors observed the installation of a freeze seal on the Unit 1 condensate storage tank return line (line 1CP04A). The freeze seal was being installed in order to facilitate repairs to this line. This activity was considered risk significant, due to the potential for both Unit 1 AF pumps to become inoperable if the freeze seal failed.

The inspectors observed the installation of the freeze seal and reviewed, as applicable, the associated 10 CFR 50.59 screening or safety evaluations, the UFSAR, the TS, and other documents listed in the Attachment. The inspectors also discussed the contingency actions, should the freeze seal fail, with the on-shift operating staff. The inspectors verified that the freeze seal was installed in accordance with the above documents. This inspection constituted one sample.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector reviewed Revisions 11, 12, and 13 of the Braidwood Station Annex to Exelon's Standardized Emergency Plan to determine if changes identified in these annex revisions reduced the Plan's effectiveness, pending on-site inspection of the implementation of these changes. This inspection constituted one sample of the inspection requirement.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed the annual emergency preparedness drill from the simulator control room and the technical support center. The inspectors verified that the drill plans identified the timing and location of expected classification, notification, and protective action recommendation opportunities and observed the conduct of the drills to verify that those opportunities had been met or that drill evaluators identified where they were not met. The inspectors also observed internal communications, NRC notifications, command and control transfers, assembly and accountability activities, and other aspects of drill performance to identify weaknesses and ensured that the licensee evaluators had also noted the same weaknesses. The inspectors verified that deficiencies noted during the drill, by either the inspectors or licensee evaluators, were entered into the licensee's corrective action program. The inspectors also attended the facility critique for the simulator control room. Documents reviewed as part of this inspection are listed in the Attachment. This activity constituted one inspection sample.

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 (PIs) for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's records to determine if any occupational exposure control cornerstone PIs had been identified during the previous five calender quarters. If PIs had been identified, the inspectors determined whether or not the conditions surrounding the PIs had been evaluated and identified problems had been entered into the corrective action program for resolution. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed records to determine if airborne radioactivity areas with the potential for individual worker internal exposures of >50 millirem committed effective dose equivalent had been identified within the facility. Work areas having a history of, or the potential for, airborne transuranics were also evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection. This review represented one sample.

The inspectors reviewed the adequacy of the licensee's internal dose assessment process for internal exposures > 50 millirem committed effective dose equivalent. This review represented one sample.

The inspectors reviewed records to determine if highly activated and/or contaminated materials (non-fuel) were stored within the spent fuel or other storage pools. The inspectors evaluated the licensee's physical and programmatic controls for those materials. This review represented one sample.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates >25 rem per hour at 30 centimeters or >500 rem per hour at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures >100 millirem total effective dose equivalent (or >5 rem shallow dose equivalent or >1.5 rem lens dose equivalent), were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure. This review represented one sample.

b. Findings

No findings of significance were identified.

.4 High Risk Significant, High Dose Rate High Radiation Area, and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate/high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to verify that any procedure modifications did not substantially reduce the effectiveness and level of worker protection. This review represented one sample.

The inspectors discussed with radiation protection supervisors the controls that were in place for special areas that had the potential to become very high radiation areas during certain plant operations, to determine if these plant operations required communication beforehand with the radiation protection group, so as to allow corresponding timely actions to properly post and control the radiation hazards. This review represented one sample.

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 review represented one sample.

b. Findings

No findings of significance were identified.

.2 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 determined that 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 review represented one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events and Mitigating Systems

.1 Reactor Safety Strategic Area

a. Inspection Scope

The inspectors reviewed documents listed in the Attachment to verify that the licensee had corrected reported PI data, in accordance with the criteria in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2. The data reported by the licensee was compared to a sampling of control room logs, CRs, and other sources of data generated since the last verification. The inspectors completed four samples by verifying the following PIs:

Unit 1

- unplanned power changes per 7000 critical hours for the period from July 1, 2002, to September 30, 2003; and
- safety system unavailability, pressurized water reactor (PWR) high pressure SI system from November 1, 2002, to October 3, 2003.

Unit 2

- unplanned power changes per 7000 critical hours for the period from July 1, 2002, to September 30, 2003; and
- safety system unavailability, PWR high pressure SI system from November 1, 2002, to October 3, 2003.

b. Findings

No findings of significance were identified.

Cornerstone: Occupational Radiation Safety

.2 Radiation Safety Strategic Area

a. Inspection Scope

The inspectors sampled the licensee's submittals for the PI and period listed below:

- Occupational Exposure Control Effectiveness

The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety, to determine if indicator related data was adequately assessed and reported during the previous five quarters. The inspectors compared the licensee's PI data with the condition report database, radiological restricted area exit electronic dosimetry transaction records, conducted walkdowns of accessible locked high radiation area entrances to verify the adequacy of controls in place for these areas, and discussed PI data collection and analyses methods with licensee representatives to verity that there were no unaccounted for occurrences in the occupational radiation safety PI as defined in Revision 2 of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline." This inspection constituted one sample.

b. Findings

No findings of significance were identified.

4OA2 <u>Identification and Resolution of Problems</u> (71152)

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 verify that they were being entered into the licensee's corrective action system 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 system as a result of 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 finding of significance were identified.

4OA3 Event Followup (71153)

The inspectors completed five inspection samples in this area.

.1 (Closed) Licensee Event Report (LER) 05000456/2003-003-00 and Unresolved Items (URI) 05000456/2003006-03 and 05000457/2003006-03: Licensed Maximum Power Level Exceeded due to an Error in a Westinghouse Supplied Calorimetric Calculation Constant.

The inspectors reviewed the LER, related CRs, and other associated documents as listed in the Attachment at the end of this report. The inspectors also discussed the event with appropriate members of the licensee's engineering and operating staff.

This issue was previously described in Section 4OA3.5 of Inspection Report 05000456/2003006; 05000457/2003006. As stated in that report, the licensee identified that an error in the Westinghouse calculations used to determine net heat value for Braidwood Units 1 and 2 had resulted in both units potentially exceeding their licensed thermal power limits by 0.4 megawatts thermal (MWt). The net heat value was the amount of heat, seen at the steam generators, which was not supplied by the reactor. This value consisted of the heat supplied by the reactor coolant pumps, pressurizer heaters, and other minor heat removals and additions. The heat removed by reactor coolant pump seal leak off was non-conservatively omitted from the net heat calculation.

The licensee's corrective actions, as described in the LER, included reducing both units reactor power to 99.98 percent and notifying the NRC of the potential violation of License Condition 2.C(1), "Maximum Power Level." Additionally, the licensee verified that the additional 0.4 MWt was bounded by the existing design-bases calculations for reactor thermal power.

The inspectors determined that this issue was not a licensee performance deficiency and was therefore not considered a finding. However, in order to characterize the significance level of the license violation, the inspectors used the Significance Determination Process (SDP) and concluded that this issue was of very low safety significance (Green). Specifically, using the SDP Phase 1 Screening Worksheet of IMC 0609, Appendix A, Attachment 1, the inspectors determined that the 0.4 MWt increase in Units 1 and 2 reactor power did not significantly challenge either the reactor coolant or fuel integrity barriers. The licensee entered this item into its corrective action system as CR 173182. The enforcement aspects of this issue are discussed in Section 4OA7.

.2 (Closed) LER 05000457/2003-002-00 and URI 05000457/2003006-02: Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination.

The inspectors reviewed the LER, related CRs, and other associated documents as listed in the Attachment at the end of this report. The inspectors also discussed the event with appropriate members of the licensee's engineering and operating staff.

This issue was previously described in Section 4OA3.4 of Inspection Report 05000456/2003006; 05000457/2003006. As stated in that report, the licensee used an ultrasonic flow measurement system to measure the time needed for feedwater disturbances to travel a known distance in the feedwater piping. The licensee identified that this system was incorrectly measuring feedwater flow owing to signal noise generated by feedwater flow pressure pulses. These pressure pulses were caused by the feedwater regulating valves, which were located upstream of the system. The pressure pulses and the resultant noise were not identified during the initial installation and subsequent testing of the ultrasonic system.

The licensee's corrective actions, as described in the LER, included reducing Unit 2 reactor power to below the licensed thermal power limit, removing the ultrasonic system from service, and notifying the NRC of the potential violation of License Condition 2.C(1), "Maximum Power Level." Additionally, the licensee was performing a technical review to determine how to reconfigure the ultrasonic system to reduce or remove the signal noise. This included a planned revision to station procedures to periodically check the ultrasonic system for signal noise and to remove the system from service if significant noise were found. The licensee also verified that a similar ultrasonic system installed on Unit 1 was operating correctly and that no overpower condition existed.

The inspectors determined that this issue was not a licensee performance deficiency and was, therefore, not a finding. However, in order to characterize the significance level of the license violation, the inspectors used the SDP and concluded that it was of very low safety significance (Green). Specifically, using the SDP Phase 1 Screening Worksheet of IMC 0609, Appendix A, Attachment 1, the inspectors determined that the 1.2 percent increase (as determined using the venturis to measure feedwater flow) in Unit 2 power did not significantly challenge either the reactor coolant or fuel integrity barriers. The inspectors' analysis also included the additional 0.4 MWt from the error in the Westinghouse calculation as described in Section 4AO3.1. The licensee entered this item into its corrective action system as CRs 173548 and 173819. The enforcement aspects of this issue are discussed in Section 4OA7.

.3 (Closed) LER 05000456/2002-003-01 and 05000457/2002-003-01: Isolated Loop RCS Boron Sample Outside of Technical Specification Frequency Requirement Due to Misapplication of the Implementing Procedure.

On November 26, 2003, the licensee issued a revision to an LER first submitted in November 2002. In the revision, the licensee simply reported that it had determined that a license amendment discussed in the original LER would not be required. No new issues were identified.

.4 Inadvertent Engineered Safeguards Actuation of AF During Clearance Order Hanging

a. Inspection Scope

On November 5, 2003, the licensee reported to the NRC, via the Emergency Notification System, that it had experienced an inadvertent actuation of the AF system on Unit 2 when two of four 6.9 kilovolt buses were de-energized, as planned, before all AF equipment was in a condition to prevent actuation. The inspectors reviewed the cause and immediate corrective actions for the event and reviewed the effect of the event on AF equipment to ensure that no adverse consequences had resulted.

b. Findings

No findings of significance were identified. The inspectors determined that the event was a minor issue. The event was caused by improper sequencing of outage work and inadequate review of plant conditions before authorizing work. The actuation resulted in the starting of auxiliary oil pumps on the 2A and 2B AF pumps and the opening of the pump discharge valves. However, the AF pumps were properly out-of-service and did not start. No water was pumped to the steam generators. Although the event was caused by performance deficiencies involving human performance in planning and sequencing of outage work, it had no adverse effect on equipment and the inspectors answered “no” to all the minor issues questions in Appendix B of IMC 0612, “Power Reactor Inspection Reports,” dated June 20, 2003. The licensee entered the event into its corrective action system as CR 184790. Documents reviewed as part of this inspection are listed in the Attachment. The licensee intended to submit an LER for this event in January 2004.

.5 Unit 2 Reactor Trip due to Loss of Feedwater

a. Inspection Scope

On December 3, 2003, the licensee reported to the NRC, via the Emergency Notification System, that Unit 2 had experienced a reactor trip from low-low level in the 2D steam generator due to loss of both the turbine-driven feedwater pumps. The inspectors reviewed the cause and immediate corrective actions for this event. Inspection of the operator response to this event was discussed in Section 1R14 of this report.

b. Findings

No findings of significance were identified. Although an operator performance deficiency in inadvertently bumping a feedwater pump steam stop valve may have contributed to the cause of this event, it was not the primary cause, and the event would not have happened if other equipment problems had not also existed. The primary causes of the event were loss of the 2C turbine-driven feedwater pump due to foreign material (part of a plastic tie wrap) in a valve test circuit relay and loss of the 2B turbine-driven feedwater pump due to a faulty speed control circuit card. The foreign material and circuit card problems were both attributed to age related degradation. Thus, the inspectors determined that a performance deficiency was not a significant contributor to this event and it is not considered a finding. The licensee entered the event into its corrective action program as CRs 188874, 189406, and 189536. Documents reviewed as part of this inspection are listed in the Attachment. The licensee intended to submit an LER for this event in February 2004.

4OA4 Cross-Cutting Aspects of Findings

A finding described in Section 1R15 of this report involved a licensee deficiency in problem identification and resolution in that the licensee identified a problem with the containment gaseous radiation monitors but the issue was not adequately resolved until questioned by the NRC inspectors.

4OA5 Other

.1 Unit 2 Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles ([Temporary Instruction] TI 2515/150, Revision 2)

a. Inspection Scope

The inspectors conducted a review of the licensee's activities in response to the requirements of Order EA-03-009, "Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," (NRC Accession Number ML030410402), issued on February 11, 2003. To support the evaluation of the licensees' activities implemented in accordance with Order EA-03-009, TI 2515/150, Revision 2, "Reactor Pressure Vessel Head and Vessel Head Penetration Nozzles (NRC Order EA-03-009)," was issued August 4, 2003. The licensee did not perform a bare metal visual examination of the Unit 2 vessel head and penetration nozzles this outage. As a result this review does not constitute credit for this TI.

A calculation of effective degradation years (EDY) that accounts for the reactor pressure vessel (RPV) head time and temperature operating history is the basis for the susceptibility ranking. For Unit 2, the licensee's EDY calculation of 1.7 placed the unit in the primary water stress corrosion cracking susceptibility category of "Low" (plants with a calculated value of EDY less than 8, with no previous inspection findings requiring classification as High). Based on the "Low" category the licensee performed a bare metal visual examination of 100 percent of the RPV head surface (including 360 degrees around each RPV head penetration nozzle) during the last (A2RO9) refueling outage. Based on the requirement to perform a bare metal visual examination of 100 percent of the RPV head surface every three refueling outages or 5 years, the licensee plans to perform the next inspection during A2R12 per the order for low susceptibility plants.

b. Findings

No findings of significance were identified.

.2 Unit 2 Reactor Pressure Vessel LHP Nozzles (NRC Bulletin 2003-02) (TI 2515/152, Revision 1)

a. Inspection Scope

The inspectors conducted a review of the licensee's activities in response to Bulletin 2003-02, which was issued on August 21, 2003. To support the evaluation of the licensees' activities implemented in accordance with Bulletin 2003-02, TI 2515/152, "Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02)," was issued September 5, 2003, and revised November 5, 2003.

Summary

The licensee did not identify any signs of leakage from the RPV LHP nozzles, or degradation of the RPV lower head.

b. Findings

No findings of significance were identified. In accordance with requirements of TI 2515/152, Revision 1, the inspectors evaluated and answered the following questions:

For each of the examination methods used during the outage, was the examination:

1. Performed by qualified and knowledgeable personnel? (Briefly describe the personnel training/qualification process used by the licensee for this activity.)

   Yes. The inspectors verified that the remote visual examination of the Unit 2 LHP nozzles was performed by qualified and certified ASME Level II and Level III Visual Testing (VT)-2 examiners. Additionally, the licensee's inspection staff were trained on Electric Power Research Institute Report TR 1000975, "Boric Acid Corrosion Evaluation."

2. Performed in accordance with demonstrated procedures?

   Yes. The remote visual examination of the vessel bottom head and the penetration nozzles using a crawler and a camera was performed in accordance with procedure ER-AP-335-1012, Revision 0, "Visual Examination of PWR Reactor Head Penetrations." The inspectors reviewed the videotape of the licensee's demonstration of color acuity and visual resolution and noted that it was consistent with the procedure requirements.

3. Able to identify, disposition, and resolve deficiencies?

   Yes. The inspectors verified that the licensee was able to identify, disposition, and resolve deficiencies.

4. Capable of identifying pressure boundary leakage as described in the bulletin and/or RPV lower head corrosion?

   Yes. The inspectors verified that the bare metal visual examinations of the Unit 2 bottom mounted instrumentation nozzles were conducted in accordance with ER-AP-335-1012, "Visual Examination of PWR Reactor Vessel Head Penetrations," Revision 0.

5. Could small boric acid deposits, as described in the Bulletin 2003-02, be identified and characterized, if present by the visual examination method used?

   Yes. Through review of the videotape documentation, the inspectors verified that small boric acid deposits, as described in the Bulletin 2003-02, could be identified and characterized. However, the licensee did not identify any leakage from the J-groove welds of the 58 LHP nozzles during the bare metal visual examination of the Unit 2 reactor vessel bottom head.

6. How was the visual inspection conducted (e.g., with video camera or direct visual by the examination personnel)?

   The licensee performed a visual examination of the Unit 2 vessel bottom head and the LHP nozzles using a remotely powered and controlled crawler (Inuktun Micro Variable Geometry Tracked Vehicle) which had variable lights, camera focus and camera tilt. The inspection was performed in accordance with procedure ER-AP-335-1012, "Visual Examination of PWR Reactor Vessel Head Penetrations," Revision 0. Each LHP nozzle was examined for 360 degrees and the entire examination was recorded on a videotape.

7. How complete was the coverage (e.g., 360 around the circumference of all the nozzles)?

   Each LHP nozzle was examined for 360 degrees and the entire examination was recorded on a videotape.

8. What was the physical condition of the RPV lower head (e.g., debris, insulation, dirt, boric acid deposits from other sources, physical layout, viewing obstructions)? Did it appear that there are any boric acid deposits at the interface between the vessel and the penetrations?

   The Unit 2 bottom head has vertical insulation panels that cover the sides of the vessel. Horizontal insulation panels are mounted to the vertical insulation panels. A minimum clearance of approximately 8 inches existed between the bottom radius of the vessel and the horizontal insulation panels. Access to the space between the horizontal insulation panels and the bottom of the vessel is provided through twelve removable, horizontal, periphery panels.

A remotely powered and controlled crawler (Inuktun Micro Variable Geometry Tracked Vehicle) which had variable lights, camera focus and camera tilt was placed on top of the horizontal panel insulation allowing access for a bare metal visual examination. Each of the 58 Unit 2 LHP nozzles was examined for 360 degrees and the entire examination was recorded on a videotape.

The bottom head had rust colored stains (approximately 1 to 2 mils thick) on the J-groove welds and adjacent areas from previous reactor cavity seal leakage (see Question 12 response). There was no indication of leakage from the J-groove welds, which would have easily penetrated the stained surface.

9. What material deficiencies (i.e., cracks, corrosion, etc.) were identified that required repair?

   None. The bare metal remote visual inspections did not identify any material deficiencies associated with the 58 LHP nozzles that required repair.

10. What, if any, impediments to effective examinations, for each of the applied nondestructive examination methods, were identified (e.g., insulation, instrumentation nozzle distortion)?

    There were no impediments to the remote visual examinations. Access to the Unit 2 LHP nozzles was provided through one of the twelve bottom head periphery removable insulation panels. A minimum clearance of approximately 8 inches existed between the bottom radius of the vessel and the horizontal insulation panels.

11. Did the licensee perform appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components above the RPV lower head?

    Yes. There were rust colored stains (1 to 2 mil thick) on J-groove welds and adjacent areas from previous cavity seal leakage. The inspectors verified that the source of leakage were identified and appropriately attributed to reactor cavity seal leakage.

12. Did the licensee take any chemical samples of the deposits? What type of chemical analysis was performed (e.g., Fourier Transform Infrared, what constituents were looked for (e.g., boron, lithium, specific isotopes), and what were the licensee's criteria for determining any boric acid deposits were not from RCS leakage (e.g., Lithium-7, ratio of specific isotopes, etc.)?

    Yes. There were no deposits from which to take samples, only slight boric acid stains. Lower Head Penetration Nozzle #45 showed the most boric acid staining of the 58 LHPs. A swipe of the boric acid staining was taken from LHP #45 for smear radionuclide analysis. The smear of the boric acid stain showed no signs of RCS activity and the isotopic analysis identified only natural background isotopes. The activity and isotopic analysis results were consistent with the Unit 2 reactor cavity leakage.

13. Is the licensee planning to do any cleaning of the head?

    Yes. The licensee pressure washed the bottom head (3000 pounds per square inch pressure washer) and conducted a bare metal visual examination to set up a baseline for future examination.

14. What are the licensee's conclusions regarding the origin of any deposits present and what is the licensee's rationale for the conclusions?

    The inspectors verified that the sources of leakage were identified and appropriately attributed to reactor cavity seal leakage. In 1996, an inflatable reactor cavity seal failed which led to staining of the LHPs and lower head. The smear of the worst boric acid staining located on nozzle #45 showed no signs of RCS activity and the isotopic analysis identified only natural background isotopes. The activity and isotopic analysis results were consistent with reactor cavity leakage.

.3 Units 1 and 2 Reactor Containment Sump Blockage (TI 2515/153)

a. Inspection Scope

The inspectors reviewed the licensee's response to NRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors." This bulletin addressed issues associated with potential post-accident debris blockage that may prevent operation of the containment emergency sumps during the recirculation mode. The NRC was tracking final resolution of this industry concern under Generic Safety Issue 191, "Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance."

The inspectors verified that the licensee's compensatory actions were either implemented or were scheduled to be implemented consistent with the response. This was accomplished, as applicable, by reviewing training records, procedures, CRs, and interviewing plant operators. The inspectors also evaluated the post-accident emergency sump availability, by reviewing the applicable engineering calculations and licensee documentation of sump inspections. Regarding the calculations, the inspectors did not validate the overall conclusion, but specifically determined whether the assumptions and methodology used in the calculations were generally consistent with existing NRC guidance. For the sump inspections, the inspectors verified that any identified deficiencies were entered into the licensee's corrective action program for resolution. Those documents specifically reviewed during this inspection are listed in the Attachment. This TI was not a part of the baseline inspection program and was therefore not considered a sample. The TI is considered complete for both units.

b. Findings

No findings of significance were identified. Answers to the questions in the TI are detailed below:

a. For units that entered refueling outages after August 31, 2002, and subsequently returned to power: Was a containment walkdown to quantify potential debris sources conducted by the licensee during the refueling outage?

   Yes. The licensee entered the tenth Unit 1 refueling outage on April 15, 2003, and the tenth Unit 2 refueling outage on November 4, 2003. During both refueling outages, the licensee conducted walkdowns of the emergency recirculation sumps. The inspectors accompanied the licensee during both of these walkdowns. The results of the Unit 1 walkdown were documented in NRC Inspection Report 05000456/2003003; 05000457/2003003. The results of the Unit 2 walkdown were documented in Section 1R20 of this report.

b. For units that are currently in a refueling outage, is a containment walkdown to quantify potential debris sources being conducted during the current refueling outage?

   Yes. See above.

c. For units that have not entered a refueling outage between September 1, 2002 and the present, will containment walkdown to quantify potential debris sources be conducted during the upcoming refueling outage?

   Not applicable.

d. Did the walkdowns conducted check for gaps in the sumps' screened flowpath and for major obstructions in containment upstream of the sumps?

   Yes. During both the Unit 1 and Unit 2 refueling outages, the inspectors performed containment walkdowns to evaluate the licensee's process for maintaining containment cleanliness, in particular, for identifying and removing potential sources of debris that could clog the emergency sumps. No significant problems were identified during these walkdowns.

   The licensee performed the walkdowns of the emergency sumps in accordance with station Engineering Surveillance Requirement procedure BwVSR 3.5.2.8, "Visual Surveillance of Containment Recirculation Sumps," Revision 2. This procedure required, in part, that licensee engineering staff evaluate the overall material condition of the sumps by looking for gaps or tears in the screened flowpath and for potential blockage in the internal sump piping. The inspectors did not identify any significant gaps or tears in the screened flowpath or potential blockage of the internal sump piping during the Unit 1 and 2 sump walkdowns.

e. Are any advanced preparations being made at the present time to expedite the performance of sump-related modifications, in case it is found to be necessary after performing the sump evaluation?

No. The licensee did not plan to make any modifications to the sumps. The licensee concluded that existing controls in conjunction with interim compensatory measures (as described in the response) would ensure the operability of the emergency sumps.

Note: During the Unit 2 walkdown, the inspectors noted that the sump outer trash rack was covered with unsecured metal plates. The outer trash rack was a single enclosure encompassing both the train A and B emergency recirculation sumps. This design was also used for the Unit 1 sumps. Because the trash rack was located adjacent to reactor coolant piping, the inspectors questioned whether the sump would survive a high energy line break event. Specifically, the inspectors were concerned that if the reactor coolant piping broke, the metal plates could be blown off, blocking one or both recirculation sumps. After additional review, the inspectors concluded that the sump design met the licensee's design basis, at the time of original construction, as stated in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems. Revision 0."

.4 Spent Fuel Material Control and Accounting At Nuclear Power Plants (TI 2515/154)

a. Inspection Scope

The inspectors interviewed the station and Exelon corporate special nuclear material custodians. The inspectors reviewed licensee procedures regarding the movement and accountability of special nuclear material. The inspectors also reviewed a sample of recent inventories of nuclear fuel and special nuclear material. Documents reviewed as part of this TI are listed in the Attachment. This TI was not a part of the baseline inspection program and was therefore not considered a sample. The TI is considered complete.

b. Findings

No finding of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. M. Pacilio and other members of licensee management at the conclusion of the inspection on January 7, 2004. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- Licensed Operator Requalification Testing for Calendar Year 2003 and Applicability of NRC IMC 0609, Appendix I, "Operator Requalification Human Performance SDP," with Mr. D. Burton on October 6, 2003;

- Radiation Protection inspection with Mr. M. Pacilio on November 14, 2003; and

- Inservice Inspection, Temporary Instruction TI 2515/150, Revision 2, and Temporary Instruction TI 2515/152, Revision 1, with Mr. T. Joyce on November 17, 2003.

- Emergency Preparedness inspection with Mr. S. McCain on December 19, 2003.

4OA7 Licensee-Identified Violations

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

Cornerstone: Barrier Integrity

1. Condition 2.C(1) of the Units 1 and 2 Operating Licensee, required that the reactor core power levels not exceed 3586.6 MWt (100 percent rated power). As discussed in Section 4OA3.1 of this report, on August 20, 2003, the licensee identified that both units had exceeded the maximum thermal power output by 0.4 MWt, owing to an error in the Westinghouse calculations used to determine the net heat value. This item was entered in the licensee's corrective action system as CR 173182.

2. Condition 2.C(1) of the Unit 2 Operating Licensee, required that the reactor core power levels not exceed 3586.6 MWt (100 percent rated power). As discussed in Section 4OA3.2 of this report, on August 31, 2003, the licensee identified that Unit 2 had exceeded its licensed power level by 0.39 percent, owing to an incorrectly measured feedwater flow. This item was entered in the licensee's corrective action system as CRs 173182, 173548, and 173819.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Pacilio, Site Vice President
T. Joyce, Plant Manager
D. Burton, Licensed Operator Requalification Training Lead Instructor
D. Chrzanowski, ISI Coordinator
G. Dudek, Operations Manager
C. Dunn, Site Engineering Director
C. Gayheart, Shift Operations Superintendent
R. Gilbert, Nuclear Oversight Manager
T. Green, NDE Level III
F. Lentine, Design Engineering Manager
R. Linthicum, Engineering Programs - Probabilistic Risk Assessment
S. McCain, Corporate Emergency Preparedness Manager
D. Meyers, Training Director
J. Moser, Radiation Protection Manager
A. Ronstadt, Site Maintenance Rule Coordinator
K. Root, Regulatory Assurance Manager
M. Sears, Engineering Programs
B. Spahr, Operations Training Manager
E. Stefan, Regulatory Assurance - NRC Coordinator
B. Stoffels, Maintenance Manager

Nuclear Regulatory Commission

M. Chawla, Project Manager, Office of Nuclear Reactor Regulation
A. Stone, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000456/2003008-01; 05000457/2003008-01	NCV	Failure to Identify That Containment Atmosphere Radiation Monitors Were Inoperable (Section 1R15)

Closed

05000456/2003008-01; 05000457/2003008-01	NCV	Failure to Identify That Containment Atmosphere Radiation Monitors Were Inoperable (Section 1R15)

05000456/2003-003-00	LER	Licensed Maximum Power Level Exceeded Due to an Error in a Westinghouse Supplied Calorimetric Calculation Constant (Section 4OA3.1)
05000457/2003-002-00	LER	Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination (Section 4OA3.2)
05000456/2002-003-01; 05000457/2002-003-01	LER	Isolated Loop RCS Boron Sample Outside Technical Specification Frequency Requirement Due to Misapplication of the Implementing Procedure (Section 4OA3.3)
05000456/02-02-03; 05000457/02-02-03	URI	Ability of Radiation Monitors to Detect RCS Leakage (Section 1R15)
05000456/2003006-03; 05000457/2003006-03	URI	Potential Error in Reactor Thermal Power Calculation Due to Incorrect Heat Input Data (Section 4AO3.1)
05000457/2003006-02	URI	Potential Error in Reactor Thermal Power Calculation Due to Feedwater Flow Signal Noise (Section 4AO3.2)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

BwOP HT-1T1; Heat Tracing Locations; Revision 4

BwOP HT-E3; Electrical Lineup - Unit 0 Chemical and Volume Control System 0HT03J Operating; Revision 1E1

CR 133739; BwOP CC-1 Minimum Temperature Limit - Leakage Past SX007s; December 2, 2002

CR 139752; All Heating Units in Fuel Handling Building Outer Trackway Not Working; January 16, 2003

CR 140655; Refueling Water Storage Tank Low Temperature BwAR Requires Refueling Water Storage Tank on Recirculation With No Procedures; January 23, 2003

CR 178645; Station Heat Unavailable With Freezing Temperature Predicted; October 1, 2003

CR 180886; Nuclear Oversight Identified Enhancements to Winter Readiness Review; October 14, 2003

CR 186645; Various Issues Identified During Freezing Temperature PM; November 6, 2003

OP-AA-108-109; Seasonal Readiness; Revision 1

WO 500954 01; Unit 0 Station Heat Area Heaters Annual Surveillance; July 6, 2003

WO 500955 01; Unit 0 Freezing Temperature Equipment Protection Annual Surveillance; August 3, 2003

WO 636168 01; Replace Thermal Overload for 1AP48E-DC, 1SI03P [Engineering Change] EC 345729 Refueling Water Storage Tank Heating System Pump; November 13, 2003

NOL 20-03-034; Memo from Gilbert, Braidwood Nuclear Oversight Manager to Pacillio, Braidwood Site Vice President and Joyce, Braidwood Plant Manager; Nuclear Oversight Assessment of Braidwood Station Winter Readiness; October 15, 2003

1R04 Equipment Alignment

BwOP FW-E1; Electrical Lineup - Unit 1 Operating; Revision 6

BwOP FW-M1; Operating Mechanical Lineup Unit 1; Revision 12

1R05 Fire Protection

CR 180882; Construction of Temporary Radiation Protection Hut Started But Not Authorized; October 14, 2003 [NRC-Identified]

Byron/Braidwood Fire Protection Report; Revision 20

Braidwood Station Pre-Fire Plans

OP-AA-201-004; Fire Prevention for Hot Work; Revision 5

OP-AA-201-009; Control of Transient Combustible Material; Revision 2

MA-AA-716-010; Maintenance Planning; Revision 3

1R06 Flood Protection Measures

BB-PRA-012l; Internal Flooding Analysis Notebook; Revision 1

BwAP 1110-3; Plant Barrier Impairment Program; Revision 11

0BwOS WF-3; Auxiliary Building Leak Detection Sump Alarm Function Operability Surveillance; Revision 2

CC-AA-201; Plant Barrier Control Program; Revision 3

CR 049952; B4 Trend Code: 2LS-WF024 Found Out Of Tolerance; January 10, 2001

CR 143296; B4 Trend Code: Instrument Out Of Tolerance 1LS-WF016; February 6, 2003

CR 154690; Missed Hourly Watch on Opened Flood Seal 1-1 (Venture); April 18, 2003

CR 154739; Nuclear Oversight Identified Missing Hourly Plant Barrier Impairment Log Entries (Venture); April 19, 2003

CR 155933; Nuclear Oversight Identified Hourly Fire Watch Posting Not Signed (Venture); April 26, 2003

CR 173604; Outdated Portions of Calculation SI-90-01 Not Superseded; August 28, 2003

CR 175367; 1LS-WF014 Cover Gasket Missing; September 11, 2003

CR 182293; 2LS-WF018 Cover Gasket Missing; October 22, 2003

CR 183942; Debris Identified in 1B AF Pump Room Leak Detection Sump; October 30, 2003 [NRC-Identified]

Regulatory Guide 1.102; Flood Protection for Nuclear Power Plants; Revision 1

WO 363523; Auxiliary Building Leak Detection Sump Alarm Function Operability Surveillance; February 21, 2003

1R08 Inservice Inspection

EXE-UT-350; Westinghouse Procedure on Ultrasonic Examination of Austenitic Piping Welds; dated March 11, 2002

EXE-PDI-UT-2; Westinghouse Procedure on Ultrasonic Examination of Austenitic Piping Welds; dated August 27, 2003

CR 183357; FME (foreign material exclusion) - 2B Feedwater Pump Suction Strainer Damaged with FME

CR 186293; FME - Foreign Object Inspection Results from 2D Steam Generator

CR 185519; FME - U2 Steam Generator 2C Preheater Inspection Results

CR 185810; 2B Steam Generator Preheater Visual Inspection Results

CR 186210: 2A Steam Generator Preheater Inspection Foreign Objects

1R11 Licensed Operator Requalification Program

OP-AA-101-111; Roles and Responsibilities of On-Shift Personnel; Revision 0

OP-AA-103-102; Watchstanding Practices; Revision 1

OP-AA-103-103; Operation of Plant Equipment; Revision 0

OP-AA-103-104; Reactivity Management Controls; Revision 0

OP-AA-104-101; Communications; Revision 0

1R12 Maintenance Effectiveness

CR 126093; 2FW035C Failed Stroke Time 2BwOSR 3.6.3.5.FW-5 Unplanned Limiting Condition for Operation; October 7, 2002

CR 128792; EH Leak on 1B Main Feedwater Pump; October 24, 2002

CR 131194; Repeat Maintenance - Process I&C Cabinet Power Supply Failure

CR 131214; Unplanned Load Reduction (60MW) for 2A Feedwater Pump Swap (~1 hour); November 12, 2002

CR 133502; 2C Feedwater Pump Low Oil Reservoir Level; November 29, 2002

CR 155262; Repeat Maintenance - Improper Sequencing of Containment Spray During 1A Diesel Generator Test; April 23, 2003

CR 152596; 2FW035A Failed to Open Fully During 1BwOSR 3.6.3.5.FW-5; April 7, 2003

CR 156920; Emergent Work Required on 1FW009A (Small Packing Leak); May 2, 2003

CR 157091; 1PA13J Timer T2A Would Not Repeat; April 25, 2003

CR 160362; Engineered Safety Features Sequence Timer for the 1B Containment Spray Failed to Operate; May 23, 2003

CR 160672; Missed/Differing Limiting Condition for Operations Entries for 1/2PA14J Work; May 27, 2003

CR 160683; 1PA14J Eagle Timer Surveillance Failed (1B Chemical and Volume Control Pump); May 28, 2003

CR 164513; Feedwater Flow Perturbation to 1D Steam Generator; June 23, 2003

CR 173548; AMAG Technology, Inc. Equipment Problems Identified at Byron Unit 1 - Applicability to Braidwood; August 28, 2003

CR 179156; 2B Feedwater Pump Adverse Condition Monitoring Plan; October 3, 2003

CR 182097; Unplanned Limiting Condition for Operation - 1PA14J Timer Sequencing Relay Failure; October 21, 2003

CR 184995; Slave Latch Erratic Behavior (K604A and K609B); November 4, 2003

CR 185510; A2R10 Lessons Learned for Feedwater Isolation/Safety Injection [SI]/Phase A Tests

EACE 154329; Relay Timer in 1PA14J As Found Out of Tolerance; May 27, 2003

ER-AA-310; Implementation of the Maintenance Rule; Revision 2

ER-AA-2202; System Health Indicator Program; Revision 3

High Safety Significant Status of In-Scope Functions (User Parameters); Feedwater

Feedwater System Work Order Backlog

Monthly Ship System Report; Unit 1 Main Feedwater; October 2003

Monthly Ship System Report; Unit 2 Main Feedwater; September 2003

Maintenance Rule - Performance Criteria; Feedwater

Maintenance Rule - Performance Monitoring User Parameters; Unit 1 and Unit 2 Train A, B, and C; Oct 1, 2001 through September 30, 2003

Maintenance Rule Expert Panel Scoping Determination; Feedwater

Maintenance Rule - Evaluation History; October 1, 2002 through October 31, 2003

1R13 Maintenance Risk Assessments and Emergent Work Control

1BwOSR 3.3.2.8-608B; Unit 1 Engineered Safety Features Actuation System Instrumentation Slave Relay Surveillance (B Train Automatic SI - K608); Revision 0

CR 180924; 2A Chemical and Volume Control Pump Cubicle Cooler Fan Tripped Causing Yellow Status; October 14, 2003

CR 180956; Chemical and Volume Control Pump Availability with One Cubicle Cooler Fan Broken; October 14, 2003

CR 180987; DC Bus 111 Batter Charger Voltage was Lower Than Expected; October 15, 2003

CR 182097; Unplanned Limiting Condition for Operation - 1PA14J Timer Sequencing Relay Failure; October 21, 2003

CR 182117; Appropriateness of Limiting Condition for Operation 3.3.2 for Timer Sequencing Relay Failure to Cycle; October 21, 2003

IN 93-17, Revision 1; Safety Systems Response to Loss of Coolant and Loss of Offsite Power; March 25, 1994

Drawing 20E-1-4030EF02; Schematic Diagram Engineered Safety Features Sequencing & Actuation Cabinet Train B 1PA14J; January 6, 1982

Shift Manager Turnover; Wednesday, October 15, 2003; Oncoming Shift 2

1R14 Operator Performance During Non-Routine Evolutions and Events

CR 181793; Unexpected Refueling Water Storage Tank Level Loss; October 19, 2003

CR 181823; Incorrect Procedure Identified During Pre-Job Brief; October 19, 2003

1BwGP 100-4; Power Descension; Revision 19

Reactivity Maneuver Form; Braidwood Unit 1 Cycle 11; July 30, 2003

Fragnet Plan for 1FW009A Repair, December 31, 2003

Testing and Troubleshooting Activities Plan for 1FW009A, December 31, 2003

1R15 Operability Evaluations

2BwOA SEC-7; Auxiliary Feedwater Check Valve Leakage Unit 2; Revision 4A

CR 179714; Westinghouse NSAL 03-09 - Unit 2 Steam Generator Narrow Range Level Uncertainties; October 6, 2003

CR 188384; Repeat Maintenance - 2AF014A Check Valve Leakage; November 27, 2003

CR 189014; 2AF014A Backleakage Issue Not Repaired in A2F37; December 3, 2003

CR 183954; Failed Fuel Canister Located in Spent Fuel Pool Region II Rack; October 30, 2003

EC 344976; Evaluation of Westinghouse NSAL 03-9 and WCAP-16115-P (Unit 2 Steam Generator Narrow Range Level Uncertainties/Setpoints); October 8, 2003

OP-AA-108-11; Adverse Condition Monitoring and Contingency Plan 2AF014A Back Leakage Monitoring; November 29, 2003

Westinghouse Letter from Humphries to Lentine; Failed Fuel Rod Storage Basket Safety Evaluation; March 13, 1992

Westinghouse Fuel Rod Storage Canister Criticality Analysis; October 1994

ComEd Letter from Krich to NRC; Request for an Amendment to Technical Specifications to Support Installation of New Spent Fuel Pool Storage Racks at Byron and Braidwood Stations; March 23, 1999

NFM:PSS:00-064; Failed Fuel Rod Storage Basket Criticality Analysis Impact; October 9, 2000

Nuclear Safety Westinghouse Electric Company Advisory Letter NSAL-03-9; Steam Generator Water Level Uncertainties - Setpoint Analysis; September 22, 2003

1R16 Operator Workarounds

BwAR 1-2-D7; Containment Penetration Cooling Flow High/Low; Revision 7

Braidwood Archival Operations Narrative Logs; September 11-12, 2003

CR 140843; BwEPs are Not Consistent with Containment Spray Technical Specification Bases 3.7.6; November 20, 1986

CR 152460; Operating Experience Review of Operating Event 14519 Potentially Applicable to Braidwood; September 12, 2002

CR 175447; 1PA40J Trouble Alarm (Breaker Not Closed - Unknown Why); September 11, 2003

CR 175457; Loss of Cooling to Unit 1 and Unit 2 Computer Rooms; September 12, 2003

EC 336997 001; Swap Location of 1TI-WO065 with 1TS-WO084 in Order to Improve Operation of the Oil Reservoir Heater on the 1B Containment Chiller; March 31, 2003

EC 337061 000; Utilize Different Contacts for Reactor Trip Breaker (Cell 1C) STB Relay; May 9, 2002

EC 344077 000; Provide Instrument Optimization Values for Unit 1 SI Accumulator; August 8, 2003

OP-AA-102-103; Operator Work-Around Program; Revision 0

Unit 1 Aggregate Review - July 2003

Unit 2 Aggregate Review - July 2003

1R17 Permanent Plant Modifications

EC 337149; Install Temporary Monitoring Equipment on the 1B AF Diesel in Order to Capture Data in the Event of a Failed Start Attempt; Revision 2

EC 344306: Unit 1 Auxiliary Feed Pump Diesel Governor Oil Reservoir Relocation; Revision 0

EC 344307; Unit 2 Auxiliary Feed Pump Diesel Governor Oil Reservoir Relocation; Revision 0

EC 344306 Revision 0 Work Planning Instructions

CR 183836; Scaffold Plant Close to Top of 3/4 Inch Essential Service Water Line; October 30, 2003 [NRC-Identified]

CR 185564; Governor Oil Reservoir Sight Glass 1LI-AF8000 Off Scale High; November 9, 2003

CR 187101; Missing and Loose Nuts From 1B AF Diesel Air Box Solenoid L2; November 18, 2003 [NRC-Identified]

WO 608391; 1AF01PB Relocate Governor Oil Reservoir Per EC 344306; October 16, 2003

1BwOSR 3.7.5.4-2; Unit One Diesel Driven AF Pump Surveillance; Revision 7

BwOP AF-7; AF Pump B (Diesel) Startup on Recirculation; Revision 23

Engineering Memorandum; Plan for Returning 1AF01PB to Normal Monthly Starts; October 23, 2003

CR 181765; New Metal Equipment Part Number Tags not Included with 1B AF Work Scope; October 18, 2003

1RST Post-Maintenance and Surveillance Testing - Pilot

ASME OMA CODE-1996; ISTC 4.5 Inservice Exercising Tests for Category C Check Valves; Subsection ISTC

BwISR 3.3.3.2-202; Surveillance Calibration of Wide Range Reactor Coolant Hot Let (Outlet) Temperature and Loop 2C Hot Leg Pressure; Revision 5

BwOP AF-7T1; Diesel Driven AF Pump Operating Log; Revision 4

1BwOSR 3.3.2.8-611B; Unit One Engineered Safety Features Actuation System Instrumentation Slave Relay Surveillance (B Train Automatic SI - K611); Revision 2

1BwOSR 3.7.5.4-2; Unit One Diesel Driven AF Pump Surveillance; Revision 7

1BwOSR 3.8.1.2-2; Unit 1 1B Diesel Generator Operability Monthly and Semi-Annual Surveillance; Revision 8

BwVSR 3.6.7.5.2; Containment Spray Additive Flow Rate Verification “Train B”; Revision 2

2BwVSR 3.8.1.11-1; Loss of Engineered Safety Features Bus Voltage with No SI Signal; Revision 3

2BwVSR 3.8.1.19-1; 2A Diesel Generator Emergency Core Cooling System Sequencer Surveillance; Revision 3

1BwVSR 5.5.8.CS.2; ASME Surveillance Requirements for 1B Containment Spray Pump and Check Valves 1CS003B, 1CS011B; Revision 3

BwVSR 5.5.8CV.4; SI System Charging Check Valve Stroke Test [Conducted November 6, 2003]; Revision 3

Pre-Job Briefing for BwVSR 5.5.8.CV.4; SI System Charging Check Valve Stroke Test A2R10; Conducted November 6, 2003

CR 179201; Safety Concerns Identified by NRC During Containment Spray Additive Test; October 3, 2003 [NRC-Identified]

CR 179514; 2B Containment Spray Additive Flow Rate Verification Test Aborted (2CS026B); October 3, 2003

CR 185225; Chemical and Volume Control Emergency Core Cooling System Injection Flow Exceeded Flow Imbalance Criteria; November 7, 2003

CR 185370; NRC Questions Regarding Chemical and Volume Control Full Flow Testing; November 6, 2003 [NRC-Identified]

CR 185556; Nuclear Oversight Identified Potential Deficient Post Maintenance Testing of Valve 2CV8481B; November 9, 2003

CR 185580; 2CC01A: Tube Plugs Missing at As-Found Inspection - FME; November 9, 2003

CR 185616; Tube Sheet Damage Removing CC Heat Exchanger Tube Plugs; November 9, 2003

CR 185956; Wrong Tube Marked for Removal on the 2CC01A Heat Exchanger; November 11, 2003

CR 186322; Cosmetic Indications on the Existing CC Heat Exchanger Tubes; November 13, 2003

CR186798; A2R10 Lessons Learned - Valve Lubrications Not Requiring Post Maintenance Testing; November 16, 2003

CR 186827; A2R10 Lessons Learned - Continuity Check for Removed Overload; November 16, 2003

MA-AA-716-012; Post Maintenance Testing; Revision 1

MA-AA-733-1001; Guidance for Check Valve Inspection; Revision 1

WO 433514 01; 2P-0455 Cold Overpressure; October 28, 2003

WO 445093 01; Unit 2 AF Diesel Prime Mover Performance Surveillance; November 16, 2003

WO 489246 01; M4-2CV8481B Disassembly/Inspection per MA-AA-733-1001; August 26, 2003

WO 489246 03; Perform Disassembly and Inspection Per MA-AA-733-1001; November 5, 2003

WO 492496-20; Final Tubesheet Leak Check; November 13, 2003

WO 599799 01; Unit 1 Train B Slave Relay Surveillance K611; October 21, 2003

WO 610541 01; Unit 2 Diesel Driven AF Pump ASME Quarterly Surveillance; November 15, 2003

WO 617367 01; 1B Diesel Generator Operability Monthly; October 21, 2003

1R20 Refueling and Other Outage Activities

Action Tracking Number 175484; Potential Adverse Trend Associated with Fuel Handling Work Activities; October 13, 2003

2BwGP 100-6, "Refueling Outage;" Revision 7

2BwOSR 3.4.11.3; Pressurizer Power Operated Relief Valve Instrument Air Accumulator Check Valve Test and Accumulator System Pressure Integrity Test; Revision 6

CR 183675; Spent Fuel Pool Rack Cell Panel Failure; October 29, 2003

CR 183767; Concerns with Heavy Load Handling Around the Spent Fuel Pool; October 29, 2003

CR 184534; Unit 2 Digital Electro-Hydraulic Controller Shifted to Manual During Ramp Offline; November 3, 2003

CR 184556; Paint Chips Near Reactor Cavity Area from Reactor Containment Fan Cooler Risers; November 4, 2003

CR 198873; Posting of Protected Equipment; November 3, 2003 [NRC-Identified]

CR 185184; NOS Identified No Protected Equipment Signs on B Train Equipment; November 6, 2003

CR 185286; Protected Equipment Posting Adherence challenge; November 6, 2003 [NRC-Identified]

CR 186776; NOS Identified Colored Dots Being Used in the Main Control Room as Equipment Status Tags; November 15, 2003

186976; Unplanned LCO 3.0.3 Entry Due to Loss of 1RF008 and 2PR11J; November 17, 2003

Exelon Nuclear Procedure OU-AA-103; Attachment 1; Shutdown Safety Approval; approved October 20, 2003

Engineering Memorandum; A2R10 Engineering Programs Personnel Mode 3 Leakage Exams Plan and Expectations

Engineering Memorandum; A2R10 Mode 3 Walkdown Results; November 4, 2003

Reactivity Maneuver Form; Braidwood Unit 2 Initial Power Ascension Following A2R10; Revision 1, November 17, 2003

Reactivity Maneuver Form; Braidwood Unit 2 Cooldown Following Shutdown for A2R10; Revision 0

WO636370 01; Low Power Physics Test Program With Dynamic Rod Worth Measure BwVS 500-6; November 18, 2003

1R23 Temporary Plant Modifications

EC 344820 000; Freeze Required to Support Operations Out-of-Service On Line 1CP04-6" to Repair Corroded Section of Line 1CP04-6" Near Column H-1, 401," September 24, 2003

Shift Manager Turnover; Friday, October 17, 2003; Oncoming shift 2

1EP4 Emergency Action Level and Emergency Plan Changes

Exelon Nuclear Radiological Emergency Plan Annex for Braidwood Station; Revisions 10, 11, 12, and 13

1EP6 Drill Evaluation

Braidwood Station 2003 Emergency Plan Exercise; October 1, 2003

CR 180003; Missing Gasket on Connection in Technical Support Center; October 1, 2003 [NRC-Identified]

CR 180219; Emergency Preparedness Exercise Failed Facility Objective; October 2, 2003

Nuclear Accident Reporting System (NARS) Utility Messages 1, 2, and 3 (Drill); October 1, 2003

Reactor Plant Event Notification Worksheets 1, 2, and 3 (Drill); October 1, 2003

20S1 Access Control to Radiologically Significant Areas

RP-AA-550; Attachment 2; Hot Spot Exposure/Corrective Action History; Revision 0

RP-AA-460; Controls for High Radiation Area and Very High Radiation Areas; Revision 2

RP-AA-460-1001; Additional High Radiation Exposure Controls; Revision 0

RP-AA-220; Bioassay Program; Revision 1

RP-AA-221; Whole Body Count Data Review; Revision 0

RP-AA-222; Methods for Estimating Internal Exposure From In Vivo and In Vitro Bioassay Data; Revision 1

Braidwood Station Year 2003 Radiation Exposure

Monthly CR Summary for PI; November 6, 2003

Electric Power Research Institute Exposure Data A1R01 - A1R08 and A2R01 - A2R09

Braidwood Station - Permanent Shielding Data; November 7, 2003

2OS2 As Low As Is Reasonably Achievable Planning And Controls

2003-2005 Exposure Reduction Plan

4OA1 Performance Indicator Verification

Braidwood's Archival Operations Narrative Logs (Selected Days with Power Changes of Greater than 20 Percent); July 1, 2002, to September 30, 2003

Braidwood Process Book Plot of 8-Hour Average of Nuclear Power for Unit 1 and Unit 2; July 1, 2002, to September 30, 2003

4OA3 Event Followup

CR 172182; Potential to Exceed Rater Thermal Power Limits (0.4 Mwt); August 26, 2003

CR 173548; AMAG Technology, Inc. Equipment Problems Identified at Byron Unit 1; August 28, 2003

CR 173819; AMAG Technology, Inc. Equipment Signal Noise Potential Effect on Reactor Power Calorimetric; August 31, 2003

CR 182779; RCS Temperature Alignment Affected by AMAG Technology, Inc. Equipment Removal; October 21, 2003

CR 184790; Inadvertent Engineered Safety Feature Actuation of AF During Clearance Order Hang; November 5, 2003

CR 188874; Unit 2 Reactor Trip on 2D Steam Generator Low Level; December 3, 2003

CR 189406; Tac Pac Circuit Boards for Feedwater Controls Not Available; December 6, 2003

CR 189536; Foreign Material Exclusion a Contributing Factor in 2FW01PC Trip; December 8, 2003

LER 2002-003-00; Isolated Loop Reactor Coolant System Boron Sample Outside of Technical Specification Frequency Requirement Due to Misapplication of the Implementing Procedure; November 26, 2003

LER 2003-002-00; Licensed Maximum Power Level Exceeded Due to Inaccuracies in Feedwater Ultrasonic Flow Measurements Caused by Signal Noise Contamination; September 30, 2003

LER 2003-003-00; Licensed Maximum Power Level Exceeded Due to an Error in a Westinghouse Supplied Calorimetric Calculation Constant; October 3, 2003

NRC Event Number 40298; Braidwood 2 AF Support System Actuation During Outage; November 5, 2003

NRC Event Number 40370; Plant Had an Automatic Reactor Trip from 100 Percent Power due to Steam Generator Low Level; December 3, 2003

NSAL-03-6; Nuclear Safety Advisory Letter; High Net Heat Input; August 20, 2003

OP-AA-101-403.1 Operator Aid Review and Approval 99-044; Feedwater Flow Constants - Unit 1 Constants Used in the Feedwater Flow Calculations of the Calorimetrics; Valid Period August 28, 2003 through December 7, 2003; Revision 13

OP-AA-101-403.1 Operator Aid Review and Approval 99-045; Feedwater Flow Constants - Unit 2 Constants Used in the Feedwater Flow Calculations of the Calorimetrics; Valid Period August 28, 2003 through December 9, 2003; Revision 15

Westinghouse Electric Company Technical Bulletin TB-03-6; Crossflow Ultrasonic Flow Measurement System Signal Issues; September 5, 2003

4OA5 Other

TI 2515/152

ER-AA-335-015; VT-2 Examination, VT-2 Visual Examination; dated November 22, 2002

SSI-A2R10-RV HEAD; Visual Inspection of Braidwood Unit 2 Reactor Vessel Head; dated March 24, 2003

NES-MS-10.02 Revision 0; Standard for Determining PWR RPV Head Penetration Inspection Categories; dated March 23, 2003

CR 185413; Boric Acid Accumulation on a Unit 2 In-core Instrument Nozzle

TI 2515/153

Analysis BRW-98-0100M; Containment Sump Zone of Influence for Failed Coatings; Revision 3

Analysis CS-5; NPSHA for Residual Heat Removal and Containment Spray Pumps; Revision 3A

2BwCA-1.1; Loss of Emergency Coolant Recirculation Unit 2; Revision 102 WOG 1C

CR 185472; Emergency Core Cooling System Sump Blockage Response - Potential Enhancements; October 30, 2003

CR 185568; Gaps in Outer Screen Containment Recirculation Sump Structural Members; November 9, 2003 [NRC-Identified]

CR 187587; Questions Regarding Emergency Core Cooling System Recirculation Sump Design; November 17, 2003 [NRC-Identified]

Drawing S-1065; Containment Building Recirculation Sump Screens Plans, Sections, and Details Units 1 and 2; Revision P

Drawing M-196; Reactor Coolant Loop Piping Arrangement Unit 2; Revision G

Regulatory Guide 1.82; Sumps for Emergency Core Cooling and Containment Spray Systems; June 1974

<u>TI 2515/154</u>

WO 00595897; Fuel Pool Physical Inventory; completed July 8, 2003

Nuclear Materials Control Procedure No. 3; Controlling Movements of Special Nuclear Material Within a Station; dated April 9, 1999

Nuclear Material Control Procedure No. 4, Physical Inventories of Nuclear Fuel and Other Special Nuclear Material Items; dated December 9, 1998

Exelon Nuclear Procedure; NF-AA-30; Special Nuclear Material Control Process Description; Revision 0

Exelon Nuclear Procedure NF-AA-300; Special Nuclear Material Control; Revision 4

Exelon Nuclear Procedure NF-AA-300-1000; Special Nuclear Material Control and Periodic Reporting; Revision 2

Exelon Nuclear Procedure NF-AA-310; Special Nuclear Material and Core Component Movement; Revision 6

Exelon Nuclear Procedure NF-AP-311-3.7.16; Special Nuclear Material Movement Requirements for Byron and Braidwood; Revision 1

Exelon Nuclear Procedure NF-AA-330; Special Nuclear Material Physical Inventories; Revision

Exelon Nuclear Procedure NF-AP-330-5201; Special Nuclear Material Inventory Requirements for Byron and Braidwood; Revision 0

Exelon Nuclear Procedure NF-AA-600; Integrated Spent Fuel Management; Revision 0

Exelon Nuclear Procedure NF-AA-610; On-Site Wet Storage of Spent Nuclear Fuel; Revision 2

Braidwood Failed Fuel Rod Storage Rack Map; dated December 2, 2003

Braidwood Station Nuclear Component Transfer List; completed May 5 and 6, 1994

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AF	Auxiliary Feedwater
ASME	American Society of Mechanical Engineers
BwAP	Braidwood Administrative Procedure
BwAR	Braidwood Annunciator Response Procedure
BwCA	Braidwood Contingency Action Procedure
BwEP	Braidwood Emergency Procedure
BwOA	Braidwood Abnormal Operating Procedure
BwOP	Braidwood Operating Procedure
BwOS	Braidwood Operating Surveillance Procedure
BwOSR	Braidwood Operating Surveillance Requirement Procedure
BwVSR	Braidwood Engineering Surveillance Requirement Procedure
CC	Component Cooling
CFR	Code of Federal Regulations
Ci/gm	Curies per Gram
CR	Condition Report
EC	Engineering Change
ECT	Eddy Current Testing
EDY	Effective Degradation Years
FME	Foreign Material Exclusion
gpm	Gallons per Minute
IR	Inspection Report
IMC	Inspection Manual Chapter
LER	Licensee Event Report
LHP	Lower Head Penetration
MWt	Megawatts Thermal
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
SDP	Significance Determination Process
SI	Safety Injection
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VT	Visual Testing
WO	Work Order