July 22, 2002

EA-02-147

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 50-456/02-06; 50-457/02-06

Dear Mr. Skolds:

On June 30, 2002, the 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 July 1, 2002, with Mr. J. von Suskil 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, the NRC has determined that one Severity Level IV Violation of NRC requirements occurred. Specifically, a procedure change associated with residual heat removal system operation resulted in a more than minor increase in the consequences of an accident previously evaluated in the final safety analysis report. Your staff did not obtain NRC approval prior to implementing the procedure change as required by 10 CFR 50.59. However, because the violation was non-willful and non-repetitive and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

The NRC also identified one violation of NRC requirements for which the final risk significance remains to be determined at a later date. This finding which involves implementing fire watches for degraded fire barriers does not present an immediate safety concern because the condition was corrected. Finally, the NRC identified one issue that was evaluated under the risk significance determination process as having a very low safety significance (Green). That issue was determined not to involve a violation of NRC requirements.

If you contest the subject or severity of the 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 Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector at the Braidwood facility.

The NRC has increased security requirements at the Braidwood Station in response to terrorist acts on September 11, 2001. Although the NRC is not aware of any specific threat against nuclear facilities, the NRC issued an Order and several threat advisories to commercial power reactors to strengthen licensees' capabilities and readiness to respond to a potential attack. The NRC continues to monitor overall security controls and will issue temporary instructions in the near future to verify by inspection the licensee's compliance with the Order and current security regulations.

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 (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.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 50-456/02-06;
50-457/02-06

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Regulatory Assurance Manager - Braidwood
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
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REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report Nos:	50-456/02-06; 50-457/02-06
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	35100 S. Route 53 Suite 84 Braceville, IL 60407-9617
Dates:	April 1 through June 30, 2002
Inspectors:	S. Ray, Senior Resident Inspector N. Shah, Resident Inspector Z. Falevits, Senior Reactor Inspector D. Pelton, Senior Operator Licensing Examiner M. Holmberg, Reactor Inspector R. Schmitt, Radiation Specialist D. Nelson, Radiation Specialist D. Funk, Physical Security Inspector J. Maynen, Observer Y. Diaz-Castillo, Observer J. Roman, Illinois Department of Nuclear Safety
Approved by:	Ann Marie Stone, Chief Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000456-02-06, 05000457-02-06; Exelon Generation Company, LLC; on 04/01-06/30/02, Braidwood Station; Units 1 & 2. Fire Protection, Refueling and Other Outage Activities, and Identification and Resolution of Problems.

This report covers a 1-quarter period of baseline resident inspection and announced baseline inspections on radiation protection and security. In addition, an inspection in accordance with Temporary Instruction 2515/145, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” was completed for Unit 2. The inspection was conducted by Region III inspectors and the resident inspectors. One Severity Level IV Non-Cited Violation (NCV), one Green finding, and one finding with significance to be determined were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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. Inspection Findings

Cornerstone: Initiating Events

Green. A finding of very low safety significance was identified through a self-revealing event when an operator inadvertently performed steps to isolate heater drain pump flow on Unit 1, which was operating at full power, instead of Unit 2, which was shutdown at the time. The primary cause of this finding was related to the cross-cutting area of Human Performance. Despite several unit-specific visual indications that were available, the operator did not perform adequate self-checking to ensure that he was on the correct unit.

This finding was more than minor because it increased the likelihood of a reactor trip event due to low steam generator level and also could have affected the availability of the main feedwater mitigating system because the motor-driven main feedwater pump, if it had been operating, could have tripped on low suction pressure. The finding was only of very low safety significance because the exposure time was short, all other mitigating systems were available, and the main feedwater system could have been recovered by fairly simple operator actions. (Section 1R20.1)

Cornerstone: Mitigating Systems

TBD. An apparent violation of Technical Specification Fire Protection Program requirements was identified by the inspectors. The licensee removed two fire rated barriers (floor plugs) in the auxiliary building, and left them off for over six months, without establishing the required compensatory firewatches. The primary cause of this apparent violation was related to the cross-cutting area of Human Performance. The licensee Fire Marshall failed to identify that the floor plugs were rated fire barriers, despite labels indicating that the 10 CFR 50, Appendix R, program applied to them, before authorizing their removal.

The issue was more than minor because a fire in one elevation of the auxiliary building could have spread to other elevations and therefore affected redundant trains of several mitigating systems. The NRC will conduct a more detailed review of the significance of this issue. (Section 1R05.1)

NCV. The licensee identified that a change to the operating procedure for the residual heat removal pumps required prior NRC approval because the change could have caused a more than minor increase in the consequences of a steam line break accident. The thyroid dose to the control room operators could have increased by more than a minor amount above that previously analyzed. This was determined to be a Severity Level IV Non-Cited Violation of 10 CFR 50.59.

Because the issue affected the NRC's ability to perform its regulatory function, it was evaluated with the traditional enforcement process. The finding was determined to be of very low safety significance because, although the procedure change could have resulted in a delay in cooling down the reactor to stop a radiation release, the effect of the change would have occurred late in the accident scenario when the technical support staff would have been available to develop methods of reducing the radiation release time and to monitor and reduce operator exposure. (Section 4OA2.1)

B. Licensee Identified Violations

A violation of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. This violation is listed in Section 40A7 of this report.

Report Details

Summary of Plant Status

Unit 1 operated at or near full power throughout the inspection period except that power was reduced to about 70 to 75 percent, for a few hours each time, on June 2, 15, 16, and 17, 2002, for load-following. Unit 2 operated at or near full power until the unit was shut down for a refueling outage on April 20, 2002. Unit 2 was made critical for physics testing on May 10, 2002, and was then shutdown for repairs on emergent conditions. The reactor was restarted and the generator placed on-line on May 12, 2002. Over the next few days, Unit 2 was gradually brought to full power. Unit 2 reached full power on May 15, 2002. On May 28, 2002, Unit 2 was reduced to about 25 percent power to allow work on a feedwater system containment isolation valve. The unit was returned to full power on May 29, 2002, and operated at or near full power for the remainder of the inspection period except that power was reduced to about 85 percent for a few hours on June 8, 2002, for load-following.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors verified that the licensee had completed its seasonal preparations for hot weather in a timely manner before the hot weather actually presented a challenge. The inspectors reviewed the licensee's completed high temperature annual surveillance and verified that it adequately covered risk-significant equipment and insured that the equipment was in a condition to meet the requirements of Technical Specifications (TSs), the Technical Requirements Manual (TRM), and the Updated Final Safety Analysis Report (UFSAR) with respect to protection from high temperatures. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action system by reviewing the associated condition reports (CRs). Based on their importance for availability of mitigating systems, the inspectors conducted more detailed system reviews and walkdowns for the following two systems:

- During the week of April 15, 2002, a period of unseasonably high temperatures, the inspectors walked down all accessible areas of the auxiliary building to verify that the auxiliary building ventilation system was maintaining room temperatures less than the limits in the TRM. On May 1, 2002, the inspectors walked down the auxiliary building ventilation system chillers to verify that they were in a condition where they could be used if necessary.

- On May 1, 2002, the inspectors walked down the Unit 1 and Unit 2 refueling water storage tank circulating and heating systems and the power supplies for the heaters to verify that the heaters had been taken out-of-service and that the circulation system was lined up for hot weather operation in accordance with the licensee's mechanical lineup procedures.

As part of these inspections, the inspectors reviewed the documents listed at the end of this report.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdowns

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of trains of risk-significant mitigating systems equipment during times when the trains were of increased importance due to the redundant trains or other related equipment being unavailable. The inspectors utilized the valve and electric breaker checklists listed at the end of this report 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 CRs associated with the trains 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 UFSAR to determine the functional requirements of the systems.

The inspectors verified the alignment of the following trains:

- Units 1 and 2 component cooling water (CC) trains on April 12, 2002;
- 1B centrifugal charging (CV) pump on April 15, 2002; and
- 2A CV pump on June 13, 2002.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

On May 24, 2002, the inspectors performed a complete system alignment inspection of the Unit 2 residual heat removal (RH) system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspection consisted of the following activities:

- a review of plant procedures (including selected abnormal and emergency procedures), drawings, and the UFSAR to identify proper system alignment;

- a review of outstanding or completed temporary and permanent modifications to the system;
- a review of control room operator log entries from September 1, 2001, through May 24, 2002, to identify potential system issues; and
- an electrical and mechanical walkdown of the system to verify proper alignment, component accessibility, availability, and current condition.

During the refueling outage, the inspectors also observed the alignment of the Unit 2 RH system to perform shutdown cooling and to drain down the reactor cavity to the refueling water storage tank following core reload.

The inspectors also reviewed selected issues documented in CRs, to determine if they had been properly addressed in the licensee's corrective actions program. Documents reviewed during this inspection are listed at the end of this report.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Fire Barriers in Auxiliary Building Removed For Extended Time With No Compensatory Measures

a. Inspection Scope

During a fire protection walkdown the week of June 10, 2002, the inspectors noted that three sets of large concrete floor plugs in the general area of the auxiliary building were removed. This created openings between the 346, 364, 383, and 401 foot elevations of the auxiliary building. The openings had been used during the Unit 2 outage to move a new essential service water pump into place. The inspectors questioned why the floor plugs had not been replaced at the end of the outage to restore the integrity of the fire barriers.

b. Findings

An apparent violation of fire protection regulations was identified in that the licensee failed to implement required compensatory measures for two missing fire barriers for over 6 months. The finding is greater than minor but is unresolved pending completion of a significance determination.

On June 21, 2002, as a result of its review of the inspectors' concern, the licensee determined that two of the three floor plugs were considered 3-hour rated fire barriers for protection of safe shutdown equipment. Those were the plugs between the 346 and 364 foot elevations and between the 364 and 383 foot elevations. The plugs between the 383 and 401 foot elevations were not considered a fire barrier. The licensee further determined that, at the time the plugs were removed on January 7, 2002, it had failed to identify that they were fire barriers and therefore had failed to implement the fire watches required as compensatory measures. The fire watches were not implemented until June 21, 2002.

The plant design depended on fire barriers between the various elevations of the auxiliary building for separation of electrical cables for redundant trains of several systems required for safe shutdown. Preliminary reviews by the inspectors determined that a fire spreading between the 346, 364, and 383 foot elevations of the general area of the auxiliary building could affect redundant trains of the essential service water, CV, CC, and auxiliary feedwater (AF) systems for both units.

The inspectors determined that failing to identify the removed fire barriers and establish the required compensator firewatches 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 Inspection Reports," Appendix B, "Issue Disposition Screening," issued on April 29, 2002. The finding involved the attribute of protection against external factors (fire) as well as human performance and could have affected the mitigating systems objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences because a fire on one elevation of the auxiliary building could have spread to other elevations containing redundant equipment cables. The finding also affected the cross-cutting area of human performance because the licensee Fire Marshall failed to identify that the floor plugs were rated fire barriers, despite labels indicating that the plugs were part of the 10 CFR 50, Appendix R requirements, before authorizing their removal.

In order to determine the significance of the finding, the NRC will need to conduct a more detailed review of the cable locations, fire detection and suppression capabilities in the areas in question, and possible fire scenarios. This will require the involvement of a regional fire protection specialist, and the issue will remain unresolved until completion of that review. The licensee entered the issue into its corrective action program as CR 112775 on June 21, 2002. On June 27, 2002, the licensee replaced the floor plugs, sealed the joints, and placed more informative labels on the plugs. Documents reviewed by the inspectors during this inspection are listed at the end of this report.

Technical Specification 5.4.1 required, in part, that written procedures shall be established, implemented, and maintained covering fire protection program implementation. One of the procedures established to meet this requirement was Braidwood Administrative Procedure BwAP 1110-1, Fire Protection Program System Requirements," which required, in Step E.7.a.3), with one or more required fire rated sealing devices unavailable, within 1 hour either establish a continuous firewatch on at least one side of the affected assembly, or verify the availability of fire detectors on at least one side of the unavailable assembly and establish an hourly firewatch patrol. However, on January 7, 2002, fire rated sealing devices (floor plugs) were removed from between the 346 and 364 foot elevations and from between the 364 and 383 foot elevations of the auxiliary building. The required firewatches were not established until June 21, 2002. This is an Unresolved Item (URI 50-456/457/02-06-01).

.2 Other Fire Protection 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 verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and 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 documents listed at the end of this report were also used by the inspectors to evaluate this area.

The following areas were inspected by walkdowns:

- Unit 1 upper cable spreading rooms on April 17, 2002;
- Unit 2 upper cable spreading rooms on April 17, 2002;
- Unit 1 lower cable spreading rooms on April 17, 2002;
- Unit 2 lower cable spreading rooms on April 17, 2002;
- auxiliary diesel generator and day tank rooms on June 12, 2002;
- 1B emergency diesel generator and day tank rooms on June 17, 2002; and
- 1B diesel driven AF pump room on June 20, 2002.

b. Findings

No findings of significance were identified.

.3 Fire Drill Observation

a. Inspection Scope

On June 27, 2002, the inspectors observed the licensee's response to a simulated fire in the old radiation protection office area. The specific fire drill scenario was 20.6.10.02, "Old RP Building," dated June 10, 2002. The inspectors chose to observe this scenario, because an actual fire in this area could cause a plant transient if equipment located in the adjacent turbine building were damaged. Prior to the drill, the inspectors performed a walkdown of the simulation with the licensee's Fire Marshall to identify the specific hazards and the drill objectives to be addressed by the fire brigade. The inspectors also performed a walkdown of the appropriate fire brigade storage cage to verify that fire fighting equipment was properly maintained. During the drill, the inspectors observed the following specific aspects of the fire brigade response:

- the fire brigade responded in a timely manner upon being notified of the fire;
- the fire brigade members protective equipment was in good, working order and was properly donned;
- fire hoses were properly laid out, charged and tested prior to entering the fire area of concern;
- fire fighting equipment was properly staged and used; and
- the fire brigade leader maintained appropriate command and control and had good radio communication with the responders.

The inspectors also attended the post-drill critique to determine whether the pre-planned drill scenario was appropriately followed and whether the specific drill acceptance criteria was met.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

During the Unit 2 refueling outage, the inspectors reviewed heat exchanger (HX) performance testing of the Unit 0 CC HX and eddy current testing of the Unit 2 CC HX. Both HXs were located in the auxiliary building and were considered highly risk-significant. The inspectors observed portions of the testing and reviewed the test results and the acceptance criteria. Specifically, the inspectors determined whether the testing was performed consistent with industry guidance as stated in Electric Power Research Institute TR-107397, "Service Water HX Testing Guidelines," dated March 1998. The inspectors also discussed the test results with the licensee's engineering staff to determine whether the HXs met their design basis heat removal rate or if there had been potential degradation in HX performance. Additionally, the inspectors reviewed selected issues that the licensee had entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance. Documents reviewed by the inspectors during this inspection are listed at the end of this report.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On June 5, 2002, the inspectors observed an operating crew during an "out-of-the-box" requalification examination on the simulator using Scenario BR-2, "Respond to an Anticipated Transient Without Scram and Miscellaneous Malfunctions," Revision 9. The inspectors evaluated crew performance in the areas of:

- 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 following documents:

- OP-AA-101-111, “Rules and Responsibilities of On-Shift Personnel,” Revision 0;
- OP-AA-103-102, “Watchstanding Practices,” Revision 0;
- OP-AA-103-103, “Operation of Plant Equipment,” Revision 0;
- OP-AA-103-104, “Reactivity Management Controls,” Revision 0; and
- OP-AA-104-101, “Communications,” Revision 0.

The inspectors verified that the crew completed the critical tasks listed in the above 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.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed systems to verify that the licensee properly implemented the maintenance rule, 10 CFR 50.65, for structures, systems, or components (SSCs) with performance problems. This evaluation included the following aspects:

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

The above aspects were evaluated using the maintenance rule scoping and report documents listed at the end of this report. For each SSC reviewed, the inspectors also reviewed the significant WOs and CRs listed at the end of this report to verify that failures were properly identified, classified, and corrected, and that unavailable time had been properly calculated.

The inspectors reviewed the licensee's implementation of the maintenance rule requirements for the following SSCs:

- Units 1 and 2 solid state protection and engineered safety features actuation (ESF) system on April 12, 2002; and
- Units 1 and 2 nuclear instrumentation system on May 17, 2002.

b. <u>Findings</u>

No findings of significance were identified.

1R13 <u>Maintenance Risk Assessments and Emergent Work Control</u> (71111.13)

a. <u>Inspection Scope</u>

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, and the documents listed at the end of this report were used by the inspectors to verify that the equipment configurations had been properly listed, that protected equipment had been identified and was being controlled where appropriate, 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 Nuclear Station Procedure WC-AA-101, "On-Line Work Control Process," Revision 6.

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

The inspectors reviewed the following activities:

- 2A RH pump maintenance work window on April 2, 2002;
- 1B AF pump undervoltage simulated start surveillance on April 12, 2002;
- troubleshooting and testing subsequent to the licensee's discovery of an emergent issue with the 1B AF pump governor oil reservoir level on May 1, 2002;
- repair of a valve cap seal weld leak on Unit 2 head vent valve 2RC8070 on May 10-12, 2002; and
- replacement of power supplies associated with Unit 1 control room annunciator cabinet 1PA19J on June 11, 2002.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Non-routine Plant Evolutions and Events (71111.14)

a. Inspection Scope

On June 28, 2002, while the inspectors were performing a plant status walkdown in the control room, Unit 1 experienced a steam generator system transient. Specifically, the steam flow controlling channel for the 1C steam generator failed, resulting in the plant operators having to enter station abnormal procedure 1BwOA INST-2, “Operation With A Failed Instrument Channel–Unit 1,” Revision 57B. The inspectors monitored control room instrumentation to verify that plant response was as expected and observed whether operators correctly followed 1BwOA INST-2. The transient was terminated when the operators switched to the alternate controlling channel and restored plant conditions to normal. The licensee subsequently identified a failed circuit card as the reason for the channel failure. This event was documented in CR 113642, “Failure of 1F-0532 1 Steam Generator Steam Flow Channel.” dated June 28, 2002.

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 TS and UFSAR to the licensee’s evaluations presented in the CRs and documents listed at the end of this report to verify that the components or systems were operable.

The inspectors reviewed the following operability evaluations:

- snubber 2RH05003S failed to meet functional testing acceptance criteria;
- 480 volt molded case circuit breakers tripping at above their setpoints;
- “as found” testing of Unit 1 pressurizer power operated relief valve, check valves;
- the 2B CV pump inboard bearing seal oil leak; and
- Fisher Model 67CFR air regulators which may not allow have air operated valves to go to their fail safe position.

Additionally, on June 10, 2002, the inspectors completed a review of the effect on control room habitability from hazardous chemicals stored on-site. This review was performed after the inspectors identified some concerns in this area during a routine plant walkdown on May 15, 2002. Specifically, the inspectors identified that there was no documented analysis, for potential effects on control room habitability, for several cylinders of Freon stored in the auxiliary building. The inspection consisted of interviews with plant personnel, a review of applicable operability and design criteria, and a walkdown of the on-site hazardous chemical storage areas.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

As part of its follow up to an unresolved issue identified at Byron Station, the licensee instituted new operator actions which may have to be taken following a failure of a reactor coolant pump thermal barrier heat exchanger. On June 19, 2002, the inspectors verified that the new operator actions had been incorporated into procedures, that operators had been informed of the changes, that ladders and other equipment were available and staged as necessary to complete the actions, and that adequate personnel resources would be available to perform the actions during an event. The following documents were reviewed as part of this inspection:

- CR 110964, "Response to Task Interface Agreement 2001-009," June 6, 2002
- Supporting Operability Determination for CR 110964, dated June 14, 2002;
- Operations Daily Orders for June 18, 2002;
- 1BwOA PRI-6, "Component Cooling Malfunction Unit 1," Revision 101; and
- 2BwOA PRI-6, "Component Cooling Malfunction Unit 2," Revision 102.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors evaluated the following permanent plant modifications:

- increasing the size of the throttling orifice plates on the CC discharge line from the Unit 1 reactor coolant pump motor lower bearing oil coolers on April 12, 2002.
- removal of the nuclear instrumentation power range negative flux rate reactor trip on Unit 2 on April 25, 2002; and

The change to the throttling orifice plates was necessary in order to increase the cooling water flow to the reactor cooling pump lower bearing oil coolers, to within the range recommended by the pump manufacturer. Although the pumps were operable prior to the change, the cooling water flow was slightly below the vendor recommendations.

The removal of the Unit 2 negative flux rate trip eliminated an unnecessary trip function, thereby reducing the potential for a spurious trip signal, which could challenge safe plant operation. Originally, the negative flux rate trip was intended to protect against departure from nuclear boiling due to an unexpected reactivity event caused by a dropped control rod or control rod bank. However, several industry analyses have determined that this trip is not required, as there is sufficient departure from nuclear boiling margin in the Westinghouse type design, regardless of the reactivity worth of the dropped control rod or control rod bank.

For each modification, the inspectors determined if potential unresolved safety questions and/or risk evaluations were evaluated by the licensee, if the associated design and licensing documents and/or station procedures were being revised, and if the modification was correctly installed. Documents reviewed as part of this inspection are listed at the end of this report.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post maintenance testing activities associated with maintenance or modification of important mitigating, barrier integrity, and support systems to ensure that the testing adequately verified system operability and functional capability with consideration of the actual maintenance performed. The inspectors used the appropriate sections of the TS and UFSAR, as well as the documents listed at the end of this report, 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.

Testing subsequent to the following activities was observed and evaluated:

- refurbishing the electrical breaker for motor-operated valve 2CV8355A on April 24, 2002;
- refurbishing the electrical breaker for motor-operated valve 2CV8355D on April 24, 2002;
- replacement of the 2A essential service water pump on April 25, 2002;
- completion of the power uprate modification on Unit 2 on June 6, 2002;
- replacement of the fuel shutoff solenoid on the 1B AF pump on June 20, 2002; and
- preventative maintenance on the 2B CC water pump on June 21, 2002.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Operator Error Involving Activity on the Wrong Unit

a. Inspection Scope

On April 20 2002, while observing activities in the control room during the Unit 2 shutdown for refueling, the inspectors observed plant and operator response to an activity that was inadvertently performed on the wrong unit.

b. Findings

A Green finding was identified after a self-revealing event where an operator isolated all heater drain tank pump flow to Unit 1, which was operating at full power, when he intended to perform the activity on Unit 2, which was shutdown at the time. The finding was not considered a violation of regulatory requirements. The finding increased the probability of a reactor trip initiating event while also potentially degrading the main feedwater mitigating system.

During the Unit 2 shutdown for refueling, while performing actions in accordance with Braidwood Operating Procedure BwOP CD/CB-4 “Condensate/Condensate Booster System Drain”, Revision 15, Step F.25.b.1, an operator mistakenly closed manual valves 1CB026A and 1CB026B instead of 2CB026A and 2CB026B. These were the combined heater drain pump discharge valves, which caused a loss of about one third of the suction flow to the Unit 1 main feedwater pumps. The loss of heater drain flow could have resulted in a low suction pressure trip of the motor-driven main feedwater pump, if it had been running, and could have led to a low steam generator level reactor trip. Numerous annunciators were received in the Unit 1 control room including a low feedwater pump suction alarm. The inspectors were in the control room at the time of the event, and observed operator and plant response. Control room operators ramped power down about 20 megawatts in accordance with 1BwOA Sec-1, ”Secondary Pump Trip Unit 1,” Revision 100, in order to reduce feedwater flow demand, and operators were sent to determine the cause of the transient and reopen the valves. Heater drain flow was restored to normal and the plant was stabilized shortly thereafter.

The inspectors determined that the operator’s action in performing activities on the wrong unit was a performance deficiency warranting a significance evaluation in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Disposition Screening,” issued on April 29, 2002. The inspectors determined that the finding was more than minor because it involved the configuration control and human performance attributes of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability during power operations because it increased the likelihood of a reactor trip on low steam generator level. To a lesser extent, the finding involved the same attributes in the mitigation systems cornerstone and affected the cornerstone objective of ensuring the availability of the main feedwater system mitigation function because the motor-driven feedwater pump, if it had been operating, could have tripped on low suction pressure. The inspectors determined that the error by the operator also affected the cross-cutting area of human performance because, despite several unit-specific visual indications that were available, such as color coding of procedures and components, the operator did not perform adequate self-checking to ensure that he was performing the activity on the correct unit.

The inspectors determined that the finding could be evaluated using the Significance Determination Process (SDP) in accordance with IMC 0609, “Significance Determination Process,” because the finding was associated with an increase in the likelihood of an initiating event and with the availability of a train of a mitigating system as discussed above. For the Phase 1 screening, the inspectors answered “yes” to Question 2 under the Initiating Events column because the finding contributed to the likelihood of a reactor trip and the likelihood that mitigating equipment would not be available. Thus a Phase 2 evaluation was required.

Using the Risk-Informed Inspection Notebook for Braidwood Nuclear Power Station Units 1 and 2, Revision 0, dated December 8, 2000, the inspectors determined that the issue’s exposure time was less than 3 days (actually only a few minutes) and increased the likelihood of the reactor trip and loss of power conversion system transient events. The listed Table 1 likelihood ratings for those two events were raised one decade, to a point value of 2, because of the increased likelihood. For the Reactor Trip SDP worksheet, the inspectors assumed that all mitigating capability was available except for one train of the motor-driven main feedwater pumps, and that it was recoverable by fairly simple operator action. This resulted in two core damage sequences of 10 points and one of 14 points. For the Loss of Power Conversion System SDP worksheet, the inspectors assumed that all mitigating capability was available. This resulted in two sequences of 7 points and one of 11 points. Those were the only events and sequences affected by the finding. Thus, the counting rules did not apply and the final SDP determination for the issue was 7 points or Green.

The operator was performing activities in accordance with a procedure for a nonsafety-related system. The procedure was not one required by or 10 CFR 50, Appendix B. Thus, no violation of regulatory requirements occurred. The licensee entered the event into its corrective action system as CR 104628, “Heater Drain Flow Isolated Due To Personnel Error,” April 20, 2002.

.2 Other Refueling Outage Activities

a. Inspection Scope

The inspectors observed the licensee’s performance during the ninth Unit 2 refueling outage (A2RO9) conducted between April 20 and May 13, 2002.

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 reactor coolant system 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.

Additionally, the inspectors performed the following specific activities:

- on April 19 and 20, 2002, the inspectors observed the control room staff perform the Unit 2 shutdown and initial cooldown;
- on April 20, 2002, the inspectors observed the operators align the RH system for shutdown cooling;
- on April 22, 2002, the inspectors observed the control room staff drain the reactor vessel to the flange;
- on April 25, 2002, the inspectors monitored a pre job briefing for the upcoming fuel handling evolution;
- on April 25 and 26, 2002, the inspectors performed a walkdown of the auxiliary building to verify the placement of clearance orders on the Unit 2B AF, Units 1 and 2 CC water, and the Unit 2B essential service water systems;
- on April 27, 2002, the inspectors observed testing of the Unit 2 pressurizer air accumulator check valves;
- on April 27 and 28, 2002, the inspectors observed core unloading and fuel shuffling activities in the reactor containment and refueling buildings, respectively;
- on April 28, 2002, the inspectors observed the alignment of the Units 1 and 2 fuel pool cooling systems;
- on April 29, 2002, the inspectors performed a walkdown of the control room and turbine building to verify the Unit 2 safety-related electrical alignment after the direct current battery charger 212 and 4kV electrical bus 242 were taken out-of-service for routine work;
- on May 7, 2002, the inspectors performed a closeout inspection of the Unit 2 containment including a review of the results of the emergency core cooling sump inspection that had been performed earlier by the licensee. As part of this inspection, the inspectors also verified that all discrepancies noted during the walkdown were recorded and corrected);
- on May 10, 2002, the inspectors observed portions of low power physics testing and initial dilution to criticality; and
- on May 12, 2002, the inspectors observed a control rod withdrawal to criticality and portions of the plant power ascension.

In particular, during fuel movement, the inspectors verified that spent fuel pool cooling operation was performed in accordance with the NRC's safety evaluation report supporting the full power uprate of Unit 2. Documents reviewed during these inspection activities are listed at the end of this report.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed selected surveillance testing and/or reviewed test data to verify that the equipment tested using the surveillance procedures met the TS, the TRM, the UFSAR, and licensee procedural requirements, and demonstrated that the equipment was capable of performing its intended safety functions. The activities were selected based on their importance in verifying mitigating systems capability and barrier integrity. The inspectors used the documents listed at the end of this report to verify 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 following tests were observed and evaluated:

- performance testing of the 1A emergency diesel generator on April 12, 2002;
- performance testing of the 1B AF pump on April 12, 2002;
- full flow and equipment response time testing of the Unit 2 motor and diesel driven AF pumps on April 15, 2002;
- local leak rate testing of the Unit 2 containment equipment hatch double gasket on May 2, 2002; and
- Unit 2 primary containment integrity verification of isolation devices inside and outside containment on May 6, 2002.

The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns, Radiological Boundary Verification, Radiation Work Permit (RWP) Reviews and Observations of Radiation Worker Performance

a. Inspection Scope

The inspectors conducted walkdowns of selected radiologically controlled areas within the plant to verify the adequacy of radiological boundaries and postings. Specifically, the inspectors walked down several radiologically significant work area boundaries (high and locked high radiation areas) in the Units 1 and 2 auxiliary building, the radwaste building, the spent fuel pool and Unit 2 containment, and performed confirmatory radiation measurements to verify if these areas and selected radiation areas were properly posted and controlled in accordance with 10 CFR Part 20, licensee procedures, and the TS. The inspectors also reviewed the radiological conditions within those work areas walked down, to assess the radiological housekeeping and contamination controls.

The inspectors reviewed selected A2R09 RWPs for various engineering, operations, radiation protection (RP) and maintenance activities. The RWPs were evaluated for protective clothing requirements, respiratory protection concerns, electronic dosimetry alarm set points, RP hold points, and As-Low-As-Is-Reasonably-Achievable (ALARA) considerations, to verify that work instructions and controls had been adequately specified and that electronic dosimeter set points were in conformity with survey indications. The inspectors also observed radiation workers performing the activities described in Section 2OS2.2, evaluated their awareness of radiological work conditions, and verified the implementation of radiological controls specified in applicable RWPs and ALARA plans.

b. Findings

No findings of significance were identified.

2OS2 As-Low-As-Is-Reasonably-Achievable (ALARA) Planning and Controls (71121.02)

.1 Radiological Work/ALARA Planning

a. Inspection Scope

The inspectors examined the station's procedures for radiological work/ALARA planning and scheduling, and evaluated the dose projection methodologies and practices implemented for A2R09, to verify that sound technical bases for outage dose estimates existed. The inspectors examined selected A2R09 radiologically significant RWP/ALARA planning packages to verify that adequate person-hour estimates, job history files, lessons learned, and industry experiences were utilized in the ALARA planning process. In addition, the inspectors reviewed the Total Effective Dose Equivalent (TEDE) ALARA evaluations developed for the decontamination of the upper/lower reactor cavity to assess the licensee's analysis for the potential use of respiratory protection equipment during the evolution.

The inspectors reviewed the exposure results for selected A2R09 activities to evaluate the accuracy of exposure estimates in the ALARA plans for those activities. The inspectors compared the actual exposure results versus the initial exposure estimates, the estimated and actual dose rates as well as the estimated and actual man-hours expended. The inspectors reviewed the exposure history for each activity to determine if management had monitored the exposure status of each activity, to determine if in-progress ALARA job reviews were needed and performed, if additional engineering/dose controls had been established, and if required corrective documents had been generated.

b. Findings

No findings of significance were identified.

.2 Verification of Exposure Estimate Goals and Exposure Tracking System

a. Inspection Scope

The inspectors reviewed the methodology and assumptions used by the licensee for its A2R09 exposure estimates and exposure goals. Actual job exposure data was compared with estimates to verify that the licensee could project and, thus, control radiological exposure. The inspectors also reviewed the licensee's exposure tracking system to verify that the level of exposure tracking detail, exposure report timeliness, and exposure report distribution were sufficient to support control of collective exposures. The inspectors reviewed the job dose history files and dose reductions anticipated through lessons learned to verify that they were appropriately used to forecast outage doses. The inspectors evaluated how the licensee had identified problems with it's exposure estimates for some jobs, the processes being utilized to revise dose estimates, and methods to improve it's dose forecasting procedures to verify that the licensee could adequately track dose.

b. Findings

No findings of significance were identified.

.3 Job Site Inspections, Radiation Worker Performance, and ALARA Controls

a. Inspection Scope

The inspectors observed A2R09 Unit 2 Containment work activities performed in radiation areas, high radiation areas and locked high radiation areas to evaluate the use of ALARA controls. Specifically, the inspectors reviewed the adequacy of RWPs, radiological surveys, attended pre-job radiological briefings, and assessed job site ALARA controls, in part, for the following work activities:

- removal/replacement stuck reactor pressure vessel (RPV) closure stud # 35; and
- reactor cavity decontamination activities.

The inspectors examined worker instruction requirements, which included protective clothing, engineering controls to minimize dose exposures, the use of predetermined low dose waiting areas, as well as the on-the-job supervision by the work crew leaders, to verify that the licensee had maintained the radiological exposure for these work activities ALARA. The inspectors evaluated RP technician performance for each of the aforementioned work evolutions, as well as observing and questioning workers at each job location, to determine that they had adequate knowledge of radiological work conditions and exposure controls. Enhanced job controls, including RP technician use of electronic teledosimetry and remotely monitored cameras, were also evaluated to assess the licensee's ability to maintain real time doses ALARA in the field. Additionally, the inspectors evaluated the implementation of the licensee's dosimetry placement guidance necessitated by significant dose rate gradients during both the removal/replacement stuck RPV closure stud # 35 and the upper/lower reactor cavity decontamination activities (i.e., per the requirements of RWPs).

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Identification and Resolution of Problems</u>

a. <u>Inspection Scope</u>

The inspectors reviewed a 2002 focus area self-assessment of ALARA Program and Outage Readiness and Preparation to evaluate the effectiveness of the self-assessment process to identify, characterize, and prioritize problems. The inspectors also reviewed corrective action documentation to verify that previous access control and ALARA related issues were adequately addressed. The inspectors also selectively reviewed January - June 2002 CRs that addressed access control and ALARA program deficiencies to verify that the licensee had effectively implemented the corrective action program.

b. <u>Findings</u>

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

.1 Identification of Radiological Monitors Associated With High/Very High Radiation Areas

a. Inspection Scope

The inspectors completed walkdowns and reviewed calibration records to verify the accuracy and operability of radiation monitoring instruments used for the protection of occupational workers. Instrumentation included area radiation monitors (ARMs), continuous air monitors (CAMs), portable survey meters, the whole body counter, and portal monitors.

The UFSAR was reviewed to identify those ARMs that were associated with transient high and very high radiation areas. These monitors included, but were not limited to, the following:

- fuel building handling incident;
- main steamline;
- incore seal table elevation 401;
- containment fuel handling incident; and
- high range containment.

CAMs were identified from the UFSAR in the following locations:

- containment atmosphere; and
- containment purge.

The inspectors performed a walkdown of selected ARMs and CAMs in order to verify that locations were as described in the UFSAR.

b. Findings

No findings of significance were identified.

.2 Calibration and Operability of Radiological Instrumentation

a. Inspection Scope

The inspectors reviewed the most recent calibrations and alarm set points for selected ARMs and CAMs. A representative sample of current calibration records were also reviewed for the whole body counter, personnel contamination monitors, portable radiation survey instruments, electronic dosimeters, and whole body frisking monitors. The inspectors observed the calibration of several portable survey instruments, reviewed source check data and observed source checks of instruments staged in the Unit 1 and 2 auxiliary building to verify compliance with procedures.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed a RP department focus-area self-assessment of radiological instrumentation and self-contained breathing apparatus (SCBA) controls, and CRs covering radiological incidents involving personnel internal contamination events and radiological instrumentation, to verify that the licensee could identify, track, and correct radiological problems in these areas.

b. Findings

No findings of significance were identified.

.4 Respiratory Protection - SCBA

a. Inspection Scope

The inspectors reviewed the status and surveillance records for SCBA that was located in various areas onsite, including those units reserved for fire brigade and control room personnel. In addition, the inspectors verified that applicable emergency response and control room personnel were properly trained, mask fit, and medically qualified in the use of SCBA.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization Program (Behavior Observation Only) (71130.01)

a. Inspection Scope

The inspectors interviewed five supervisors and five non-supervisors (both licensee and contractor employees) to determine their knowledge level and practice of implementing the licensee's behavior observation program responsibilities. Selected procedures pertaining to the Behavior Observation Program and associated training activities were also reviewed. Also licensee fitness-for-duty semi-annual test results were reviewed. In addition, the inspectors reviewed a sample of licensee self-assessments, audits, and security logged events. The inspectors also interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP2 Access Control (Identification, Authorization and Search of Personnel, Packages, and Vehicles) (71130.02)

a. Inspection Scope

The inspectors reviewed the licensee's protected area access control testing and maintenance procedures. The inspectors observed licensee testing of all access control equipment to determine if testing and maintenance practices were performance based. On two occasions, during peak ingress periods, the inspectors observed in-processing search of personnel, packages, and vehicles to determine if search practices were conducted in accordance with regulatory requirements. Interviews were conducted and records were reviewed to verify that security staffing levels were consistently and appropriately implemented. Also, the inspectors reviewed the licensee's process for limiting access to only authorized personnel to the protected area and vital equipment by a sample review of access authorization lists and actual vital area entries. The inspectors reviewed the licensee's program to control hard-keys and computer input of security-related personnel data.

The inspectors reviewed a sample of licensee self-assessments, audits, maintenance request records, and security logged events for identification and resolution of problems. In addition, the inspectors interviewed security managers to evaluate their knowledge and use of the licensee's corrective action system.

b. Findings

No findings of significance were identified.

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspectors reviewed Revision 54 (dated January 8, 2002) and Revision 55 (dated April 17, 2002) to the Braidwood Nuclear Power Station Physical Security Plan to verify that the changes did not decrease the effectiveness of the security plan. The referenced revisions were submitted in accordance with 10 CFR 50.54(p).

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

40A1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

On May 1, 2002, the inspectors performed a review of the data submitted by the licensee for the first quarter 2002 performance indicators for any obvious inconsistencies prior to its public release in accordance with MC 0608, "Performance Indicator Program."

b. Findings

No findings of significance were identified.

The inspectors noted that the licensee's submittal for the Emergency Response Organization Drill Participation Performance Indicator was in error because the indicator value was greater than 100 percent. The numbers for participating key personnel and total key personnel were transposed. The licensee submitted a corrected report before the data was released to the public. This error was considered a minor issue. The issue was entered into the licensee's corrective action system as CR 106671, "Incorrect Data entered for Performance Indicator for Emergency Response Organization Drill Participation," April 15, 2002.

.2 Reactor Coolant System Leakage

a. Inspection Scope

On May 10, 2002, the inspectors reviewed the performance indicator data submitted by the licensee for the Reactor Coolant System Leakage Performance Indicator for both units for the period of April 1, 2001, through March 31, 2002. The inspectors reviewed the results of Reactor Coolant System inventory balance surveillances recorded in the electronic control room logs for a sampling of several months during that period to verify that the highest identified leakage value found during the month was reported, as required by Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 2. In performing this inspection, the following documents were reviewed:

- Braidwood Operability Surveillance Requirement Procedure 1BwOSR 3.4.13.1, "Unit One Reactor Coolant System Water Inventory Balance 72 Hour Surveillance," Revision 2;
- 2BwOSR 3.4.13.1, "Unit Two Reactor Coolant System Water Inventory Balance 72 Hour Surveillance," Revision 2;
- 3.4.13, "Reactor Coolant System Operational Leakage; and
- Braidwood Nuclear Station Operator Logs (electronic) for April 1, 2001, through March 31, 2002.

b. Findings

No findings of significance were identified.

.3 Public Radiation Safety

a. Inspection Scope

The inspectors verified the licensee's assessment of its performance indicators for public radiation safety. Since no reportable elements were identified by the licensee for the 2nd, 3rd, and 4th quarters of 2001 and the 1st quarter of 2002, the inspectors reviewed the licensee's data to verify that there were no occurrences concerning the public radiation safety cornerstone during those quarters.

b. Findings

No findings of significance were identified.

.4 Physical Protection

a. Inspection Scope

The inspectors verified the data for the Physical Protection Performance Indicators pertaining to Fitness-For-Duty Personnel Reliability, Personnel Screening Program, and Protected Area Security Equipment. Specifically, a sample of plant reports related to security events, security shift activity logs, fitness-for-duty reports, and other applicable security records were reviewed for the period between October 2000 and March 2002.

b. Findings

No findings of significance were identified.

40A2 Identification and Resolution of Problems (71152)

.1 (Closed) Unresolved Item (URI) 50-456/457/01-13-01: Potential inadequate 10 CFR 50.59 evaluation for 1B RH Pump.

The licensee identified a Severity Level IV Non-Cited Violation (NCV). This issue was evaluated through the SDP and determined to be of very low safety significance. The licensee failed to obtain prior NRC approval for a change to the procedure for operation of the RH pumps which would have required a license amendment in accordance with 10 CFR 50.59.

This issue was previously discussed in Inspection Report 50-456/457/01-13, Section 1R15. It involved the licensee's discovery that a change made to the operating procedure for the RH pumps could have caused a more than minimal impact on the consequences of certain accidents, and therefore, may not have been allowed without prior NRC approval. Specifically, BwOP RH-6 "Placing the RH System In Shutdown

Cooling" was revised to delay placing RH in shutdown cooling until the reactor coolant system was below 260 degrees Fahrenheit in order to minimize the transient and prevent failure of the RH pumps. However, it did not identify that the UFSAR Section 5.4.7.2.7 assumed that RH system was placed into service for shutdown cooling at 350 degrees Fahrenheit for the natural circulation without letdown analysis. The reduced temperature at which shutdown cooling would be placed in service would increase steaming to the atmosphere during some accident scenarios resulting in release of greater amounts of activity and thus impact calculated off site and control room dose rates. The issue was unresolved pending the licensee's determination of the extent of the potential consequences.

On May 7, 2002, the licensee completed its evaluation, as documented in CR 106807, and concluded that the only accident the procedure change would have adversely affected to the point of being more than an minor effect was the steam line break. Assuming a concurrent loss of offsite power, a single failure of one of the steam generator power operated relief valves to close, worst-case pre-accident fuel leakage, decay heat rate, and other initial parameters, the control room thyroid dose for the accident initiated spike case would have been a more than minimal increase (more than a 10 percent increase from the previously analyzed dose). Therefore, in accordance with 10 CFR 50.59, the licensee needed to obtain NRC approval prior to implementing the procedure change.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the SDP. Typically, a Severity Level would be assigned after consideration of appropriate factors for the particular regulatory process violation in accordance with the NRC Enforcement Policy. However, the SDP is used, if applicable, in order to consider the associated risk significance of the finding prior to assigning a severity level. Using MC 0612, Appendix B, "Issue Dispositioning Screening," the inspectors determined that the finding was more than minor because it involved the attributes of quality of a post event operating procedure for a mitigating system and the availability of equipment in the mitigating systems cornerstone and affected the cornerstone objective by reducing the capability of the RH system to be used to prevent undesirable consequences (i.e. operator thyroid dose).

The procedure change was in effect for 11 days before the licensee discovered the issue regarding the inadequate safety review. Corrective actions were completed to revise the procedure again within about 2 months. During that time period, there were no operations which involved placing the RH system in service. Had a steam line break accident occurred during that period, it is likely that the Technical Support Staff would have developed or revised procedures to minimize the time for cooling down the plant after the accident. In addition, the control room dose would have been monitored during the accident, and actions could have been taken to minimize personnel exposures. Using MC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," the inspectors answered "no" to all five screening questions in the Phase 1 Screening Worksheet under the Mitigating Systems column. The inspectors concluded the issue was of very low safety significance.

Part 50.59 of 10 CFR states, in part, that the licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would: ... (iii) Result in more than a minimal increase in the consequences of an accident previously evaluated in the final safety analysis report (as updated).

However, between November 15, 2001, and January 22, 2002, the licensee implemented a change to BwOP RH-6, "Placing the RH System in Shutdown Cooling", which would have resulted in more than a minimal increase in the consequences of a steam line break accident previously evaluated in the final safety analysis report (as updated) without obtaining a license amendment. The result of the violation was determined to be of very low safety significance; therefore, this violation of 10 CFR 50.59 was classified as a Severity Level IV violation. This Severity Level IV violation is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 50-456/457-02-06-02). This violation is in the licensee's corrective action program as CR 106807. This URI is closed.

.2 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 report.

b. Findings

No finding of significance were identified.

.3 Selected Issue Follow-up Inspection

Effectiveness of Problem Identification

a. Inspection Scope

In the past year, the inspectors have observed an adverse trend pertaining to molded case circuit breaker (MCCB) testing and operation at Braidwood. Several inadvertent breaker trips during motor operated valve (MOV) operations have occurred and a number of MCCBs failed to trip at their high test current setpoint values during testing. To assess this issue, the inspectors reviewed inspection reports, CRs, corrective action documents, test procedures, maintenance work orders and industry standards pertaining to Westinghouse model HFB molded case circuit breaker operation, maintenance and testing requirements. This was done to determine if problems relative to MCCBs at Braidwood were being identified at the proper threshold; entered into the corrective action process; evaluated for cause, operability and extent of condition; and addressed in a timely manner commensurate with the safety significance of the issues. The documents listed at the end of this report were used during the review.

b. Issues

Westinghouse Model HFB Breakers - Background Information

Braidwood's breakers used to provide the power feed to the MOVs are Westinghouse Model HFB magnetic only MCCB. These breakers provide short circuit fault protection for the MOV power circuits and coordination within the Auxiliary Power distribution system. This type of breaker has an adjustable magnetic only trip element and provides only an instantaneous trip of the breaker. The trip setting of the MOV must coordinate with the trip setting of the associated upstream breakers. The trip setting must be higher than the motor starting currents to prevent inadvertent trips during motor start. The licensee determined that the largest size of the HFB adjustable magnetic only breaker used at Braidwood was a 25 ampere breaker.

The instantaneous trip on the MCCBs at Braidwood was originally set to a specific trip current setpoint, which was based on 10 times the motor rated full load current +/- 25 percent tolerance. Subsequently, in 1996 the licensee established new setting guidance for MCCBs in standard NES-EIC-10.01, Revision O, "Molded Case Circuit Breaker Selection and Setting Design Standard," which specified settings of 2 times motor locked rotor current, but not to exceed 17 times full load current. This allowed a higher acceptable band for a given setting which should not trip the MCCB prematurely on valve inrush current but trip it at the higher setpoint (maximum actuation current). This setting defined the minimum and maximum actuation current for the instantaneous trip for this model breaker. If the current through the breaker was below the minimum actuation current, the instantaneous trip should not actuate. If the current through the breaker was above the maximum actuation current, the instantaneous trip should actuate and trip the breaker. The actual pickup current could vary between individual breakers of the same type and with the same settings, but was expected to be within the setpoint tolerance, which was the basis of the tolerance.

The inspectors noted that many of these safety-related HFB breakers were installed in the plant in the 1970's and had not been exercised or tested routinely since installation. The vendor, industry and regulatory documents recommend periodically mechanically exercising the MCCBs which will materially increase their reliability. The licensee has recently started exercising and testing these breakers on 6-year intervals. At the time of the inspection, most MOVs still had their original breaker instantaneous current design settings.

Observation of Ongoing MCCB Testing Activities

On April 21, 2001 the licensee initiated EC#335908, Revision 1, to revise the MCCBs instantaneous trip setpoints per the latest Exelon established methodology for calculating MCCB settings, since the original criteria of 10 times full load current was found by the licensee to be too restrictive for optimal valve operation. The licensee has started to implement the new setpoints in the field.

On April 24, 2002, the inspectors observed licensee electricians perform testing on two HFB molded case circuit breakers feeding reactor coolant pump (RCP) 2A and 2D seal injection isolation valves 2CV8355A and 2CV8355D. The licensee tested the breakers to verify that they would actuate at their expected setting tolerance by injecting a test current into the breaker at a specified value above the maximum actuation current for the design settings. Both breakers failed the trip setting (trip test) set values in the procedure and tripped at currents higher than the established setpoints. Both breakers were replaced and retested. Subsequent testing on a number of other HFB MCCBs resulted in similar breaker failures to trip at the higher trip setting (trip test) setpoints. The breakers had to be replaced with new breakers and retested. Subsequently, between April 21 and April 29, 2002 the licensee initiated CR 104750, 105053, 105186, 105426, 105256, 105397, 105399, 105670 and 105984 to document the failures of the tested HFB breakers to trip at the high current setpoint. Some of the breakers tripped at close to twice the specified high instantaneous trip setpoint current values. Extent of condition review indicated that these type breakers are used throughout Braidwood and Byron Units 0, 1 and 2.

On April 26, 2002, the licensee initiated CR 105657 to document a potential common mode failure associated with the instantaneous trip settings on the Westinghouse HFB MCCBs. The CR documented that since June 2001, there have been 14 CRs written where model HFB breakers failed the acceptance criteria for the trip setting. The CR was issued to identify the potential common mode failure and to recommend that a common cause analysis be performed to review this issue. The CR noted that there was industry experience with age related degradation of the breaker trip mechanism in MCCBs if they were not operated as part of a preventive maintenance program. However, it stated that a common cause analysis was required to determine what caused the breakers trip mechanism at Braidwood to degrade.

MCCB Related Issues Captured in NRC Reports

Condition Report 74717, dated September 10, 2001, documented that during performance of the quarterly stroke surveillance for 1SI8804B, the valve failed to stroke and tripped its breaker open. The valve did not open. The B Train of the emergency core cooling system was declared inoperable. Condition Report 074714 was initiated to determine corrective actions and extent of condition. The cause of tripping was identified by the licensee to be low instantaneous breaker current trip settings after the breaker was replaced in June 2001 due to unsatisfactory test results. During the apparent cause evaluation, the licensee determined that the starting characteristics for some MOV motors under normal plant operating voltages may not be bound by the original guidance used to determine the MOV feed breaker settings. Therefore, the guidance used during the original plant design may have established breaker settings that can result in breaker trips when the valves are operated under normal conditions. The licensee identified approximately 80 valves having incorrect instantaneous current trip setpoints. Little action was taken to correct these setpoints until additional failures to pass testing acceptance criteria were identified in April 2002. This issue was considered more than minor and was documented in Inspection Report 50-456/457-2001-010 as an NCV.

Condition Reports 96945 and 96746, dated February 25, 2002 documented that while performing breaker surveillance it was discovered that the breaker and overload did not pass its acceptance criteria and the breaker for valve 2SX016B tripped. The breaker was replaced with like for like breaker; however, valve 2SX016B did not pass its subsequent operational test and failed to stroke on demand due to improper setting of the instantaneous current trip setpoint on the breaker. This issue was considered more than minor and was documented in Inspection Report 50-456/457-2002-005 as an NCV.

<u>Corrective Action and Safety Significance of Nonconforming MCCBs</u>

In general, the licensee has identified MCCB related problems and entered them into the corrective action program by initiating CRs. The threshold for initiating a CR appeared appropriate. The licensee performed an apparent cause evaluation and operability evaluation for the nonconforming condition and determined that the degraded condition did not prevent accomplishment of its specified safety function. The licensee determined that the range of expected fault currents on the motor control center loads was above the potential actuation points of the MCCBs based on the degradation seen and therefore the breakers were capable of providing the required fault protection and isolation of the motor starter and feed cables from damage as a result of short circuits. The licensee examined the coordination scheme and concluded that coordination was maintained for the degradation seen with the breakers. The licensee stated that, although most tested breakers were found to be out of tolerance on the high side, they tripped when tested and would have tripped on a fault condition with no impairment to its operation, but not at the desired setting. The licensee performed an engineering evaluation and concluded that since the MCCBs were tripping at higher current values than specified and not tripping prematurely at the low end of the band they were considered operable. In addition, an inadvertent breaker trip does not affect the ability to manually operate the MOV.

4OA3 <u>Event Follow-up</u> (71153)

.1 <u>(Closed) LER 50-456/02-001-00</u>: Set Point Drift Causes Two of Three Pressurizer Safety Valve Lift Tests to Exceed TS Tolerance.

On April 8, 2002, the licensee discovered that two of the three Unit 1 pressurizer safety valves had “as found” setpoints outside of the $\pm$ 1 percent tolerance required by 3.4.10. These valves had been removed during the ninth Unit 1 refueling outage and subsequently tested at an offsite facility. During testing, one valve had a lift setpoint 1.1 percent high, the other was 1.4 percent low. The license identified no material condition issues with the valves and attributed the test failures to the inability of the valves to perform within the required close tolerance. This behavior was consistent with industry data as described in Electric Power Research Institute Test Report TR-105872, “Safety and Relief Valve Testing Guide,” dated August 1996. Additionally, the licensee determined that even with the safety valves being out-of-tolerance, the limits of the applicable UFSAR accident scenarios would not have been exceeded. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the test failures in CR 102884, "Pressurizer Safety Valves Set Test Out of Tolerance," dated April 8, 2002. This LER is closed.

.2 (Closed) LER 50-456/01-001-01: Three Main Steam Safety Valves Exceeded The TS Limit By Greater Than 3 Percent.

The specifics of this issue were described in Revision 0 of this LER and were discussed in NRC Inspection Report 50-456/456/01-13. Revision 1 of this LER, contained supplemental information regarding the licensee's root cause investigation. This investigation identified that oxide bonding between the valve nozzle and disk seating surfaces may have caused the high lift setpoints. This was based, in part, on industry experience, showing that long continuous runs (i.e., no thermal cycle between tests) may contribute to oxide bonding. However, the cause of this bonding was not well understood by the industry. Prior to being tested, these valves had completed a 535-day continuous run. The licensee's analysis results were reviewed by the inspectors and no findings of significance were identified. The licensee documented the failure of the safety valves and the subsequent analysis in CR 75897, "1MS016B, 1MS017B, and 1MS014D Exceeds 3 percent Criteria," dated September 19, 2001. This LER is closed.

.3 (Closed) LER 50-457/02-001-00: Multiple Test Failures of Pressurizer Safety Valves and Failure to Report Those Failures Due To Management Weakness In Applying Reportability Requirements.

On April 8, 2002, the licensee discovered that the failure of multiple pressurizer safety valves on Unit 2 had not been reported as required by 10 CFR 50.73(a)(2)(i)(B). Specifically, the pressurizer safety valves were removed during the seventh (May 7-27, 1999) and eighth (October 21-November 16, 2000) Unit 2 refueling outages, respectively, for subsequent testing at an offsite facility. In each of the outages, two of the three safety valves had "as found" setpoints outside of the $\pm$1 percent tolerance allowed by TS 3.4.10. The license attributed the test failures to the inability of the valves to perform within the required close tolerance, consistent with industry data, and determined that the applicable UFSAR accident analyses were still met. However, station management did not recognize that this event was reportable until subsequent, similar testing of the Unit 1 pressurizer safety relief valves (see LER 50-456/02-001-00 discussed above). Corrective actions included providing training to all applicable licensee personnel on determining reportability. Although the failure to report the previous failures of multiple pressurizer safety valves as required by 10 CFR 50.73(a)(2)(i)(B) should be, and was, corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. The LER was reviewed by the inspectors and no findings of significance were identified. The licensee documented the failure to report the Unit 2 test failures in CR 102884. This LER is closed.

The inspectors also reviewed Technical Report 01823-TR-001, "Evaluation of Dresser 3700 Series Safety Valve Iconel X-750 Disc," Volume 1 of 1, Revision 0, dated December 2001.

4OA4 Cross-Cutting Findings

.1 A finding described in Section 1R05.1 of this report had, as its primary cause, a human performance deficiency, in that, the licensee Fire Marshall, despite labels indicating that 10 CFR 50, Appendix R requirements applied, failed to identify that floor plugs in the auxiliary building were rated fire barriers before authorizing their removal.

.2 A finding described in Section 1R20.1 of this report had, as its primary cause, a human performance deficiency, in that, an operator, despite several unit-specific visual indications that were available, failed to perform adequate self-checking and isolated heater drain tank pump flow on the wrong unit.

4OA5 Other Activities

.1 Circumferential Cracking of RPV Head Penetration Nozzles (Temporary Instruction 2515/145)

a. Inspection Scope

Braidwood Station, Unit 2, is in the sub-population of plants (Bin 4) that have a low-susceptibility for head penetration nozzle cracking. The licensee responded to NRC Bulletin 2001-01, "Circumferential Cracking Of RPV Head Penetration Nozzles," by describing their past practice of performing head inspections above the metal reflective horizontal insulation. Because the insulation was not removed during these inspections, no documented examinations existed for the vessel head-to-nozzle interface under the head insulation.

The licensee conducted a visual examination of all head penetration nozzles on the vessel head from under the insulation layer during the A2R09 refueling outage. The inspectors observed the licensee's remote visual examination of one half of the bare metal head conducted under the head insulation on April 23 and 24, 2002. On April 29, 2002, inspectors reviewed digital pictures of the second half of this head inspection.

The following documents were reviewed during this inspection:

- Procedure CEDI-A2R09-RV HEAD-A, "Visual Inspection of Braidwood Unit 2 Reactor Vessel Head," dated April 19, 2002;
- Drawing 8758D33, Sheet 7, "Coordinates and Elevation of Closure Head Penetrations," ECN 23603;
- Drawing DR-4359D-10, "Reactor Vessel Top Dome Insulation Layout Plan," Sections and Details, Revision A; and
- Drawing 701J782, Sheets 1 and 2, "General Assembly Diagram of the 4-Loop Integrated Head Package," Original Revision.

b. <u>Evaluation of Inspection Requirements</u>

(1) Was the examination:

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

Yes. The remote visual examination of the head was observed by knowledgeable licensee personnel certified to Level II or III as VT-2 examiners in accordance with programs meeting the American Society for Nondestructive Testing Recommended Practice SNT-TC-1A. Additionally, these personnel had completed self-guided training, which consisted of a video tape and Technical Report 1006296, “Visual Examination For Leakage Of PWR [Pressurized Water Reactor] Reactor Head Penetrations,” supplied by the Electric Power Research Institute Materials Reliability Project.

(b) Performed in accordance with approved and adequate procedures?

Yes. The visual examinations were conducted in accordance with CEDI-A2R09-RV HEAD-A, “Visual Inspection of Braidwood Unit 2 Reactor Vessel Head.” The inspection scope included all vessel head penetrations and was intended to meet visual quality standards established for remote VT-2 examinations as defined in Section XI of the American Society of Mechanical Engineers Code.

(c) Adequately able to identify, disposition, and resolve deficiencies?

Yes. The visual inspection procedure was qualified by demonstrating the ability to resolve characters of dimensions identified in American Society of Mechanical Engineers Code Section XI for VT-2 standards. The remote visual inspection was conducted with a camera mounted to a hand held pole. Inspectors reviewed the tape of the qualification of the visual system and considered the lighting and picture quality to be adequate. The results of the head examination were required by procedure CEDI-A2R09-RV HEAD-A to be recorded on videotape and documented in a report.

(d) Capable of identifying the primary water stress corrosion cracking phenomenon described in the bulletin?

Yes. This examination was capable of detecting cracking for the penetrations in which the annulus gap (space between the head bore and the outside diameter of the penetration nozzle) was sufficient to allow nozzle leakage to reach the head surface. However, the licensee had not performed an analysis to determine which penetrations had sufficient gaps to allow nozzle leakage to be identified.

(2) What was the condition of the reactor vessel head (debris, insulation, dirt, boron from other sources, physical layout, viewing obstructions)?

The reactor vessel head is covered with reflective metal insulation panels installed on a steel support structure. The remote camera visual inspection was conducted under the insulation support structure and the as-found head condition was generally clean (free of debris, insulation, dirt). The uphill side of the annulus gap on a few penetrations contained loose debris, which was removed with a vacuum and did not hinder the licensee's inspection. For the half of the vessel head inspection observed, the licensee achieved a complete visual inspection of each head penetration with the exception of the head vent. For the head vent penetration, a steel support member for the horizontal metal insulation obstructed the examination in one quadrant (e.g. approximately 45 degrees could not be viewed). Additionally, the head was covered with a protective coating (reportedly zinc based) that precluded direct observation of the vessel head material. The licensee did not have the records for this coating which was reportedly applied by the vessel head vendor.

(3) Could small boron deposits, as described in the bulletin, be identified and characterized?

Yes. Small boron deposits as described in the bulletin could be identified due to the cleanliness of the head. However, the licensee had not conclusively demonstrated that a leakage path would exist for any of the penetration nozzles. Therefore, the licensee was relying on the expectation that sufficient gaps were present to detect leakage.

No indications of boron deposits (indicative of penetration leakage) were found during the portion of the head examination (approximately one-half) completed on April 24, 2002. From the digital pictures of the second half of the head inspection, boric acid deposits were identified at head penetration 76. These deposits appeared to be from a previous coolant leak (May of 1994) at a nearby conoseal connection which is located above the head insulation layer. This leak reportedly had run down through the insulation layer at this penetration resulting in a small buildup of boric acid on the uphill side of this head penetration. The licensee cleaned off this boric acid deposit and no evidence of head degradation was observed.

(4) What materiel deficiencies (associated with the concerns identified in the bulletin) were identified that required repair?

None.

(5) What, if any, significant items that could impede effective examinations and/or ALARA issues were encountered?

No significant impediments to the examination were identified. The projected dose for the head inspections was 956 mrem. The actual dose received was tracking with the projected dose at the conclusion of this inspection.

.2 Review of Institute of Nuclear Power Operations Report

The inspectors completed a review of the final report for the Institute of Nuclear Power Operations May 2001 Evaluation dated January 14, 2002.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. J. von Suskil and other members of licensee management at the conclusion of the inspection on July 1, 2002. 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

The results of the Temporary Instruction 2515/145 inspection for Braidwood Unit 2 inspection were presented to Mr. K. Schwartz and other members of licensee management at the conclusion of the inspection on April 24, 2002. The inspectors did not receive any information identified as proprietary during this inspection.

The results of the safeguards inspection were presented to Mr. J. von Suskil and other members of licensee management at the conclusion of the inspection on May 24, 2002. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

The results of the RP inspection were presented to Mr. T. Joyce and other members of licensee management at the conclusion of the inspection on June 7, 2002. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following violation of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600 for being dispositioned as an NCV

1. Prior to March 2001, 10 CFR 50.59(a)(1) stated, in part, that the licensee may make changes in the facility, as described in the safety analysis report, without prior Commission approval, unless the proposed change involved an unreviewed safety question. In May 2002, the Braidwood licensee was notified by Byron station personnel of a violation which was identified by the NRC during an inspection at the Byron station. Specifically, the NRC identified that Safety Evaluation 6G-98-0200, “Editorial Clarification to Byron/Braidwood UFSAR,

Section 9.2.2.4.4," (dated July 28, 1998) was inadequate, in that, the change made to the facility as described in this safety evaluation could result in an unanalyzed, unisolable, containment-bypassing loss of coolant accident which was considered a violation of 10 CFR 50.59 (a)(1). This violation was determined to be of very low safety significance; therefore, was classified as a Severity Level IV violation and because it was captured in the Braidwood corrective action program (CR 110964), it is considered a Non-Cited Violation consistent with Section VI.A.1 of the NRC Enforcement Policy.

KEY POINTS OF CONTACT

Licensee

J. von Suskil, Site Vice President
T. Joyce, Plant Manager
G. Baker, Site Security Manager
R. Blaine, Radiation Protection Manager
D. Chrzanowski, Inservice Inspection Coordinator
G. Dudek, Operations Manager
C. Dunn, Site Engineering Director
A. Ferko, Regulatory Assurance Manager
J. Harvey, Nuclear Oversight Manager
J. Kuchenbecker, Maintenance Manager

Nuclear Regulatory Commission

M. Chawla, Project Manager, NRR
A. Stone, Chief, Reactor Projects Branch 3
D. Chyu, Regional Fire Protection Specialist

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-456/457/02-06-01	URI	Failure to Establish Compensatory Firewatches for Two Removed Fire Rated Barriers (Section 1R05.1)
50-456/457/02-06-02 EA-02-147	NCV	Inadequate 10 CFR 50.59 Evaluation for Procedure Change to RH Operating Procedure (Section 4OA2.1)

Closed

50-456/457/02-06-02 EA-02-147	NCV	Inadequate 10 CFR 50.59 Evaluation for Procedure Change to RH Operating Procedure (Section 4OA2.1)
50-456/457/01-13-01	URI	Potential inadequate 10 CFR 50.59 evaluation for 1B RH Pump (Section 4OA2.1)
50-456/02-001-00	LER	Set Point Drift Causes Two of Three Pressurizer Safety Valve Lift Tests to Exceed Tolerance (Section 4OA3.1)
50-456/01-001-01	LER	Three Main Steam Safety Valves Exceeded The Limit By Greater Than 3 Percent (Section 4OA3.2)
50-457/02-001-00	LER	Multiple Test Failures of Pressurizer Safety Valves and Failure to Report Those Failures Due To Management Weakness In Applying Reportability Requirements (Section 4OA3.3)

Discussed

50-456/457/01-10-02	NCV	Failure to Have Procedure Appropriate to Circumstances (Section 4OA2.3)
50-456/457/02-05-01	NCV	Failure to Use the Correct Instantaneous Current Trip Setpoint (Section 4OA2.3)

LIST OF ACRONYMS AND INITIALISMS USED

A2RO9	Unit 2, Refueling Outage 9
ADAMS	Agencywide Documents Access and Management System
AF	Auxiliary Feedwater
ALARA	As-Low-As-Reasonably-Achievable
ARM	Area Radiation Monitor
BwAP	Braidwood Administrative Procedure
BwGP	Braidwood General Procedure
BwMP	Braidwood Maintenance Procedure
BwOA	Braidwood Abnormal Operating Procedure
BwOP	Braidwood Operating Procedure
BwOSR	Braidwood Operability Surveillance Requirement
BwVP	Braidwood Engineering Procedure
BwVS	Braidwood Engineering Surveillance
CAM	Continuous Air Monitor
CC	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
CV	Centrifugal Charging Pump
EA	Escalated Action
ESF	Engineered Safety Features
HX	Heat Exchanger
MCCB	Molded Case Circuit Breaker
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations
OOS	Out-of-Service
PARS	Publicly Available Records
RCP	Reactor Coolant Pump
RH	Residual Heat Removal
RP	Radiation Protection
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SDP	Significance Determination Process
SCBA	Self-Contained Breathing Apparatus
SI	Safety Injection
SSC	Structures, Systems, or Components
SX	Essential Service Water
TEDE	Total Effective Dose Equivalent
TRM	Technical Requirements Manual
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation
WO	Work Order
WR	Work Request

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

0BwOS XHT-A1	High Temperature Equipment Protection Annual Surveillance	Revision 4, Completed 4/19/02
1BwOSR 0.1-,2,3	Unit One - Modes 1,2, and 3 Shiftly and Daily Operating Surveillance Data Sheet	Revision 15
BwOP SI-M1	Operating Mechanical Lineup Unit 1	Revision 13
BwOP SI-M2	Operating Mechanical Lineup Unit 2	Revision 12
CR 106679	Inadequate Documentation of Surveillance Discrepancies (NRC Identified)	May 3, 2002
TRM Table T3.7.d-1	Area Temperature Monitoring	Revision 1
WO 99283474 01	High Temperature Equipment Protection	March 31, 2002

1R04 Equipment Alignment

CR 085292	Unexpected Unit 2 RH Pump Discharge Pressure Increase	December 6, 2001
CR 087705	Instrument Pre-calibration for 1A RH ASME [American Society of Mechanical Engineers] Not Per Procedure	December 18, 2002
CR 091373	RH System Recommended (a)(1) By Expert Panel	January 14, 2002
CR 096841	Spare RH Motor Not Stored Properly	February 26, 2002
CR 101826	RH Suction Relief Not Popped Per ASME Code Requirements	September 28, 2001
CR 102721	1FI-SX105 (1B CV Pp Cubicle Cooler Essential Service Water Flow) Read 0 GPM	April 9, 2002
CR 107738	2BwGP 100-1 Step 37 Concern with Commitment for RH Pump Shutdown	May 9, 2002
2BwOA PRI-10	Loss of RH Cooling Unit 2	Revision 100
2BwOA S/D-2	Shutdown Loss of Coolant Accident Unit 2	Revision 100
BwOP CC-P11	Operating Mechanical Line-Up, Unit 1	Revision 13
BwOP CC-P12	Operating Mechanical Line-Up, Unit 2	Revision 12

BwOP CV-E1	Electrical Lineup - Unit 1 Operating	Revision 5
BwOP CV-E2	Electrical Lineup - Unit 2 Operating	Revision 4
BwOP CV-M1	Operating Mechanical Lineup Unit 1	Revision 15
BwOP CV-M2	Operating Mechanical Lineup Unit 2	Revision 16
BwOP RH-E2	Electrical Lineup - Unit 2 RH System Operating Electrical	Revision 1
BwOP RH-M3	Operating Mechanical Lineup 2A Train	Revision 5
BwOP RH-M4	Operating Mechanical Lineup 2B Train	Revision 4
BwOP RH-6	Placing the RH System in Shutdown Cooling	Revision 26
BwOP RH-9	Pump Down of the Reactor Cavity to the RWST [Refueling Water Storage Tank]	Revision 17
WO 00421145 01	Unit Two Emergency Core Cooling System Venting and Valve Alignment	May 5, 2002
	Operator Log Entries (Selected) from September 1, 2001 through May 2, 2002	

<u>1R05</u> <u>Fire Protection</u>

CR 112775	Failure to Properly Identify Fire Barrier during Plant Barrier Impairment Review (NRC and Licensee Identified)	June 21, 2002
BwAP 1110-1	Fire Protection Program System Requirements	Revision 15
BwAP 1110-3	Plant Barrier Impairment Program	Revision 11
CC-AA-201	Plant Barrier Control Program	Revision 3
Fire Protection Report Section 2.3 (selected subsections)	Fire Area Analysis	Amendment 18
Fire Protection Report Section 2.4	Safe Shutdown Analysis for Braidwood-1 and Braidwood-2	Amendment 18
Table	Summary of Braidwood Fire Induced Core Damage Frequency Results	Based on Calculation BRW-97-0502-N
CC-AA-211	Fire Protection Plan	Revision 0

OP-AA-201-008	Pre-Fire Plans	Revision 0
Fire Protection Report Figure 2.3.7	Upper Cable Spreading Area	Amendment 18
Fire Protection Report Figure 2.3.9	Lower Cable Spreading Area	Amendment 18
Fire Protection Report Sections 3.2.B-1 through 3.2.E-1	Fire Hazards Analysis Lower Cable Spreading Rooms Unit 1	Amendment 19
Fire Protection Report Sections 3.2.B-2 through 3.2.E-2	Fire Hazards Analysis Lower Cable Spreading Rooms Unit 2	Amendment 19
Fire Protection Report Sections 3.3.A-1 through 3.3.D-1	Fire Hazards Analysis Upper Cable Spreading Rooms Unit 1	Amendment 19
Fire Protection Report Sections 3.3.A-2 through 3.3.D-2	Fire Hazards Analysis Upper Cable Spreading Rooms Unit 2	Amendment 19
CR 112271	Fire Dampers Not Periodically Surveilled (NRC Identified)	June 17, 2002
Dwg. M-1294	Station Auxiliary Diesel Generator & Fuel Tank Room Vent System	Revision F
	Memo From Gene O'Donnell, Fire Protection Design Engineer, x2550, "Response to Fire Protection Requests/Questions Regarding Security Diesel Room	June 12, 2002
TR 148	Fire and Hose Stream Test of TC0-003 High Density Silicone Elastomer, TC0-049 High Density Silicone Gel, TC0-050 Silicone foam, and TC0-029 Pre-Fab Aluminized Boot Seals for Mechanical Penetrations	March 14, 2985
TR-198	Fire and Hose Stream Test of TC0-010 Ceramic Blanket Single Wrap, TC0-019 Ceramic Blanket Spiral Wrap, and TC0-029 Pre-Fab Aluminized Boot Seals for Mechanical Penetrations, and TC0-001 Cement for Electrical Penetrations	March 19, 1985

1R07 Heat Sink Performance

CR A2001-01169	1VA03SB: Eddy Current Testing Deferred	April 20, 2001
CR 00112490	1VA02SB - 1B RH PP Cubicle Cooler Thermal Performance Test	June 18, 2002
CR 098719	2nd Deferral Generated for 0CC01A HX Inspection	March 11, 2002
CR 105327	Inner Diameter Tube Pitting Found in Unit 2 CC HX (2CC01A) During A2R09	April 24, 2002
WO 00340761 01	1VA02SB Thermal Performance Test of the 1B RH Pump Room Cubicle Cooler	June 18, 2002
WO 99160303 01	Thermal Performance Test of the Unit 0 CC HX	April 20, 2002
WO 990046618 01	Thermal Performance Test of CC HXs	March 18, 2000
Memorandum	Braidwood Station Unit 2 CC Water HX (2CC01A) Summary of A2R09 Activities and Basis for Continued Operation Through A2R10	May 3, 2002

1R12 Maintenance Rule Implementation

CR 084531	Possible Repetitive Maintenance Rule Functional Failure for Criterion EF2 (OCC01A)	August 27, 2001
CR 074130	Failure of Power Range Channel N44	September 1, 2001
CR A2001-01597	NR-42 Inoperable Due To Axial Flux Difference Calibration Difficulties	May 29, 2001
CR A2001-01884	Failure of Power Range Nuclear Instrument N-44	June 25, 2001
	Maintenance Rule Expert Panel Meeting Minutes	August 29, 2001
Maintenance Rule - Evaluation History	System EF	
Maintenance Rule - Expert Panel Scoping Determination	System EF	

Maintenance Rule - (a)(1) Action Plan	System: NR	November 1, 2001
Maintenance Rule - Performance Criteria	System: NR	
Maintenance Rule - Expert Panel Scoping Determination	System: NR	
Maintenance Rule - Evaluation History	System: NR	

1R13 Maintenance Risk Assessments And Emergent Work Control

BwOP AN-10	1PA19J Annunciator Cabinets Power supply Energization/De-Energization	Revision 2
BwOP AN-11	2PA19J Annunciator Cabinets Power Supply Energization/De-Energization	Revision 3
1BwOSR 3.3.2.3	Unit One Undervoltage Simulated Start of 1A AF Pump Monthly Surveillance	Revision 0E1
1BwOSR 3.7.5.3-2	Unit One Diesel Driven AF Pump Monthly Surveillance	Revision 0E2
CR 111478	Multiple Problems with the 1PA19J Planned Maintenance (NRC & Licensee Identified)	June 11, 2002
CR 108842	Aux Bldg Painting Not Turned Over to Oncoming Shift (NRC Identified)	May 20, 2002
CR 106374	1B AF Diesel Governor Oil Reservoir Found Empty on Ultrasonic Test Exam	May 1, 2002
CR 107744	2RC8070 Class Boundary Issues Requiring Evaluation	May 11, 2002
CR 107570	Rework-2RC8070 Valve Found Leaking During Post Maintenance Test at Normal Operating Temperature and Pressure	May 10, 2002
WO 00440447 05	Body to Bonnet Leak - Tighten	May 10, 2002
WO 99193303-01	1PA10J Annunciator Logic Cabinet	June 11, 2002
50.59 Screening No. BWR-S-2002-178	Temporary Leak Sealant Injection of Valve 2RC8070	Revision 0
	Exelon Nuclear Radiological Emergency Plan Annex for Braidwood Station	January 3, 2002

1R15 Operability Evaluations

CR 111324	2B CV Pump Oil Leak	June 9, 2002
CR 111809	NOS [Nuclear Oversight] Identified Lack of Familiarity of Illinois Department of Nuclear Safety Diesel Fuel Tank	June 12, 2002
CR 101417	Snubber 2RH05003S Failed to Meet Functional Test Acceptance Criteria	March 28, 2002
Reg Guide 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	June 1974 Revision 0
Reg Guide 1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	December 2001 Revision 1
BRW-SE-2000-1264	Design Change Package (DCP) Type #D20-0-00-378 DCP #9900561	December 6, 2000
EN-MW-501-0002	Chemical Control Evaluation Guidance	Revision 0
BwOA ENV-6	Operation During Chlorine/Toxic Chemical Incident Unit 0	Revision 1
MAD Form 13.1	On-Site Hazardous Chemicals Effect on Control Room Habitability	Revision 0 June 24, 1976
ComEd Memo to Mr. J. Meister	Safety Significance of Reduced Long Term Charging Pump Operation	July 26, 1996
CR 112290	Air Operated Valves With 67CFR Air Regulators May Not Bleed on Loss of Instrument Air	June 18, 2002

1R17 Permanent Plant Modifications

CC-AA-102	Design Impact Screening	Revision 2
CC-AA-103	DCP 9900571 (EC 42844) NIS Power Range Negative Flux Rate Trip Elimination	August 24, 2001
CC-AA-107	Design Change Test, Acceptance Criteria, and Results Acceptance	Attachment B

EC 0000 331965 000	Remove Diodes Instead of Lifting Leads	July 25, 2001
EC 00000 42813 000	Increase Size of CC Throttling Orifices Downstream of RCP	April 13, 2002
EC 00000 42844 001	NIS [Nuclear Instrumentation System] Power Range Negative Flux Rate Trip Elimination	August 21, 2001
DCP 990571 Attachment D	DCP Instructions, "Addendum #2, Braidwood Unit 2, D20-2-00-386," (EC42844)	May 4, 2001 and August 24, 2001
	10 CFR 50.59 Applicability Review Form	May 2, 2001
20E-2-4030EF21	Schematic Diagram "Reactor Protection Power Range Trip Logic Train A"	Revision E
20E-2-4030EF65	Reactor Protection Power Range Trip Logic Train B	Revision E
20E-2-4030AN093	Schematic Diagram "Demultiplexer Control Cabinet ZPA17J, PT. 3"	Revision 4
20E-2-4030AN053	Schematic Diagram "Trip Status Lights and Bypass-Permissive Lights"	Revision G
BRW-01-0162-M	Develop a Flow vs Pressure Graph for square Edge Orifices of Various Sizes	Revision 0
BwOP CC-17	Balancing CC System Flows to the Reactor Coolant Pumps	Revision 0E5
Dwg. 8881D48	CS Motor Bearing Assembly Lower	October 24, 1977

1R19 Post Maintenance Testing

CR 00112588	Daily Schedule Error	June 6, 2002
ER-AA-321	Administrative Requirements for Inservice Testing	Revision 2
TP-EXE-IST-01-04 Final Status	Exelon IST [Inservice Testing] Program Technical Position Pump Testing Set Value Tolerance Range	Revision 0 August 10, 2001
BwVP 850-5	Vibration Monitoring Program	Revision 0
WO 00425097 01	Unit 1 Diesel Driven AF Pump ASME Quarterly Surveillance	June 24, 2002

WO 00447691 01	ASME Surveillance Requirements for CC Pump	June 21, 2002
WO 00448817 01	Diesel Driven AF Pump Monthly Surveillance	June 20, 2002
WO 98102827 08	Braidwood Generating Station Pump, 2A Essential Service Water ASME	April 25, 2002
WO 99223287 03	2CV8355A Thermal Overload Preventive Maintenance	April 24, 2002
WO 99217697 03	2CV8355D Thermal Overload Preventive Maintenance	April 24, 2002
WO 99222778 01	2CV8355A Trip Test Molded Case Circuit Breaker	April 24, 2002
WO 99222779 01	2CV8355D Trip Test Molded Case Circuit Breaker	April 24, 2002
WO 99222779 03	2CV8355D Adjust Magnetic Trip	April 24, 2002
BwVP 850-22	Braidwood Power Uprate Project Pre and Post Installation Electrical Output Test	Revision 1

1R20 Refueling and Outage Activities

CR 104618	Heater Drain Flow Isolation Due To Personnel Error	April 20, 2002
CR 105577	NRC Question Regarding Main Steam Supports (NRC Identified)	April 23, 2002
CR 106097	Control Rod Drive Guide Funnel 40 Found on Top of the Upper Internals	April 27, 2002
CR 107772	Critical Position Estimated to be Outside Administrative Limit	May 12, 2002
BwAP 370-3A12	Fuel Handling Guidance for Fuel Movement from the Reactor Core to the Spent Fuel Pool	Revision 4
BwAP 370-3A13	Fuel Handling Guidance for Fuel Movement from the spent Fuel Pool to the Reactor Core	Revision 4
BwAP 370-3A14	Fuel Handling Guidance for Fuel Movement in the spent Fuel Pool	Revision 0

BwAP 2364-3	Safeguarding and controlling Movements of Nuclear Fuel Within a Station	Revision 4E1
BwAP 2364-9	controlling Movements of Nuclear Fuel Into the spent Fuel Racks	Revision 5
2BwGP 100-1T5	Containment Integrity Checklist	Revision 8
2BwGP 100-2	Plant Startup	Revision 16
2BwGP 100-3	Plant Ascension 5% to 100%	Revision 22
2BwGP 100-4	Power Decension	Revision 16
2BwGP 100-5	Plant Shutdown and Cooldown	Revision 20
BwMP 3100-092	Installation and Removal of Temporary Containment Penetration Covers	Revision 2
BwOP AP-E7	Electrical Lineup - Unit 2 Operating Lineup for Safety Related 4160V [volt] Busses, 480V Switchgear Busses, and 480V MCC's [motor control centers]	Revision 3E3
BwOP AP-64	Bus 242 Outage While in Mode 6 or Defueled	Revision 3
BwOP AP-64T1	Bus 242 Outage Checkoffs	Revision 1
BwOP AP-64T5	Bus 242 Outage Instrument Bus Temporary Power Installation and Removal	Revision 0
BwOP CC-8	Isolation of CC Between Units 1 and 2	Revision 14
BwOP CC-12	Alignment of the "0" HX to a Unit	Revision 6E1
BwOP DC-1-212	125V DC [direct current] ESF [engineered safety feature] Battery Charger 212 Start-Up	Revision 2E2
BwOP DC-2-212	125V DC ESF Battery Charger 212 Shutdown	Revision 1E3
BwOP DC-7-212	125V DC ESF Bus 212 Cross-Tie/Restoration	Revision 2E2
BwOP FC-1	Fuel Pool Cooling System Start-Up	Revision 15
BwOP FC-E2	Electrical Lineup - Unit 2 Lineup Operating	Revision 1E1
BwOP FC-M2	Operating Mechanical Lineup Unit 2	Revision 6E3
BwOP RC-4	Reactor coolant system Drain	Revision 22

BwOP RH-11	Securing the System From Shutdown Cooling	Revision 17
2BwOSR 3.1.1.1-1	Shutdown Margin Daily Verification During Shutdown	Revision 5
2BwOS 3.4.3.1	Reactor Coolant System Pressure/Temperature Limit Surveillance	Revision 3
2BwOS TRM 2.5.b.1	Unit Two containment Loose Debris Inspection	Revision 0
OOS Checklist 00006545	Diesel Driven Aux Feed Pump 2B Assembly	April 25, 2002
OOS Checklist 00008674	Essential Service Water Pump 2B Crosstie Valve Assembly	April 26, 2002
WO 00433222 01	Perform Unit Restart Review	April 20, 2002
WO 99225666 01	Unit 2 Containment Penetration Status Weekly	April 20, 2002
WO 99228680 01	Visual Inspection of Containment sumps	April 29, 2002
NF-AA-200-1520	Reactivity Maneuver (ReMa) Form Page 1 of 1- Coastdown Guidance Revision 0 (3/7/02)	Revision 0
NF-AA-200-153	Reactivity Maneuver Form Page 1 of 1- End of Life Load Drop for End Turn Vibration Revision 0 (4/8/02)	Revision 0
Reactivity Maneuver Form	Shutdown Unit 2 for A2R09 on 4/19/02	Revision 0
Pre-Job Briefing Worksheet	Fuel Handling Outage Offload/Insert Shuffle/Reload	
Installation Instruction NPSI-II-2	Variable Springs	Revision 3
Memorandum	NRC Walk Through [of Unit 2 Containment]	May 7, 2002
Memorandum	Containment Walkdown [by licensee management]	May 7, 2002
GL 88-17	Loss of Decay Heat Removal 10 CFR 50.54(f)	October 17, 1988
GL 98-02	Loss of Reactor Coolant Invent Associated Potential for Loss of Emergency Mitigation Functions While in a Shutdown Condition	May 28, 1998

1R22 Surveillance Testing

BwOP PC-1	Local Leak Rate Flowmeter Monitor Operation	Revision 2
2BwOSR 3.6.1.1-6	Primary Containment Type B Local Leakage Rate Test of Equipment Hatch Double Gasket	Revision 2
2BwOSR 3.6.3.3	Unit Two Primary Containment Integrity Verification of Isolation Devices Outside Containment	Revision 4
2BwOSR 3.6.3.4	Unit Two Primary Containment Integrity Verification of Isolation Devices Inside Containment	Revision 2
1BwOSR 3.8.1.2-1	Unit One 1A Diesel Generator Operability Monthly and Semi-Annual Surveillance	Revision 4
2BwVS 800-014	Heightened Level of Awareness Briefing Worksheet	April 17, 2002
BwVS 900-8	Diesel Generator Engine Analysis	Revision 7
2BwVSR 3.6.1.1.25	Summation of Type “B” & “C” Tests For Acceptance Criteria	Revision 3 Completed 5/7/02
CR 105963	Nuclear Oversight Identified Local Leak Rate Test Enhancements	April 28, 2002
CR 106417	Local Leak Rate Test Box Gives Suspected Results After Test is Completed	May 2, 2002
WO 99223480 01	U2 Full Flow Test and Equipment Response Time of AF Pumps	April 15, 2002
WO 990054871 01	Full Flow Test and Equipment Response Time of AF Pumps	October 16, 2000

20S1 Access Control to Radiologically Significant Areas

RWP 10000871	NRC Surveillance and Tours	Revision 2
RP-AA-376	Radiological Posting, Labeling, and Markings	Revision 0
RP-AA-403	Attachment #3, RWP Acknowledgment and Access to the RPA [radiological protected area]	Revision 0

RPJS-ADM-39	High Radiation Area and Locked High Radiation Area Barrier Guidance, "Defense in Depth Strategy"	Revision 0
CR 072322	Operator Didn't Use Designated Portal Monitor - NOS Issue	August 15, 2001
CR 076090	High Radiation Area Control Violation	October 22, 2001
CR 104735	RPA Work (Snubber and Inservice Inspections) Done Under Wrong RWP	April 20, 2002
CR 105615	Electronic Dosimeter Failed to Collect Dose	April 26, 2002
CR 106376	Violation of Radiological Posting - Operations Personnel	May 1, 2002

2OS2 ALARA Planning and Control

BwRP 6020-2	Radiological Air Sampling Program	Revision 8
BwRP 6020-2A1	Air Sample Pre-screen Volume Graph for Co-60	Revision 2
RP-AA-401	Operational ALARA Planning and Controls	Revision 2
RWP 10000872	A2R09 Venture Work in all Areas	Revision 1
RWP 10000877	A2R09 Inservice Inspection Activities	April 21, 2002 and April 23, 2002
RWP 10000883	Reactor Cavity Decon Activities	Revision 2
RWP 10001067	A2R09 RCP Seal Replacement and Motor Inspection Work, Work-in Progress Review	April 26, 2002
RWP 10001078	A2R09 Shielding, Installation and Removal, Work-in Progress Review	April 22, 2002 and April 23, 2002
RWP 10001104	Unit 2 Outage: Reactor Stuck Removal	Revision 1and Revision 2
RWP 10001104	ALARA Plan: Unit 2 Outage: Reactor Stuck Removal	Revision 1
RP-AA-401	Attachment #3, ALARA Briefing Checklist	Revision 2
RP-AA-401	Attachment #3, ALARA Briefing Checklist, Reactor Upper/Lower Decontamination Evolution	Revision 2

RP-AA-401	Attachment #8, Recognized Risk Personnel Contamination Dose Assessment Form	Revision 2
RP-AA-401	Attachment #8, Recognized Risk Personnel Contamination Dose Assessment Form, RWP 10000883, Reactor Cavity Decontamination Evolution	April 11, 2002
RP-AA-441	Attachment #2, TEDE ALARA Evaluation Screening Worksheet	Revision 1
RP-AA-441	Attachment #2, TEDE ALARA Evaluation Screening Worksheet, RWP 10000883, Activities After Initial Rinse-down in Upper Reactor Cavity	April 12, 2002
	A2R09 Dose (i.e., Overall, by Project, and Work Groups) Projection Graphics	May 2, 2002
	A2R09 Refuel Outage, Braidwood Nuclear Station, Nine-day Schedule	April29-May7, 2002
	A2R09 Reactor Cavity Scrubbing Decon Plan, RWP 10000883, Sequence of Cavity Decon	May 3, 2002
	ALARA Plan Amendment Form, RWP #10000883, Wide Range Monitor Dosimetry Requirement for Reactor Cavity Decontamination	May 2, 2002
	Attachment #2, ALARA Plan, Decontamination of Reactor Cavity Floor and Wall Surface Areas, RWP 10000883	April 18, 2002
	Contamination Survey, Upper Reactor Cavity Upon Draindown/Decontamination	May 2, 2002
	Micro ALARA Plan, Addendum for RWP 10000883, Emergent Work on #35 Bolt Hole	May 2, 2002
	“Outage Experiences, A2R09"	April 2002
2002-005	Focus Area Self-Assessment of the ALARA Program and Outage Readiness and Preparation	April 7 - April 8, 2002
	Common Cause Analysis Report, A Large Number of Events Involving Exposure Estimating Issues at Braidwood Station are Analyzed for a Common Cause Trend	March 27, 2002

20S3 Radiation Monitoring Instrumentation

BwISR 3.3.3.2-212	Surveillance Calibration of Main Steam Line Radiation Monitors	Revision 4
BwISR 3.3.3.2-213	Surveillance Calibration of High Range Containment Radiation Monitors	Revisions 3 and 4
BwISR 3.3.6.6-201	Surveillance Calibration of Containment Fuel Handling Incident ARMs _AR11J and _AR12J	Revision 5
0BwISR 3.3.8.3-201	Surveillance Calibration of Fuel Building Handling Incident ARMs 0AR55J and 0AR56J	Revision 2E1
BwISR 3.4.15.4-201	Surveillance Calibration of GA [GA Technologies] Gaseous Effluent Radiation Monitors	Revision 2
BwIS RETS 2.2B-201	Surveillance Calibration of GA Effluent Gaseous Radiation Monitors	Revision 6
WR 970084100 01	2R-AR23A Calibration of Area Accident Radiation Monitor (Main Steam Line)	February 3, 1999
BwRP 5510-13	Operation, Use, and Inspection of SCBA	Revision 8
BwRP 5510-13T4	ISI Viking SCBA Checklist	Revision 0
LS-AA-2150	Monthly Performance Indicator (PI) Data Elements for RETS/ODCM [Radiological Effluents /Offsite Dose Calculation Manual] Radiological Effluent Occurrences	Revision 2
WO 970108631 01	2AR20J Calibration of Area Accident Radiation Monitor (Containment)	April 19, 1999
WO 970113531 01	2PR-011; Gaseous Effluent Radiation Monitor (Containment Atmosphere)	May 16, 1999
WO 990008395 01	2AR22JJ Calibration of Area Accident Radiation Monitor (Main Steam Lines)	July 24, 2000
WO 990022267 01	0AR056J Calibration of ARM (Fuel Handling Building)	August 24, 2000
WO 990050250 01	2PR01J Calibration of Effluent Gaseous Radiation Monitor (Containment Purge System)	October 25, 2000

WO 990136877 01	2AR-AR011, Replace High Voltage Power Supply and Calibration Detector (Containment)	October 26, 2000
CR 072322	Operator Didn't Use Designated Portal Monitor - NOS Issue	May 15, 2002
CR 106862	NOS Identified RP Instrument Control Log Deficiencies	May 15, 2002
98734	Braidwood RP Department Focus-Area Self-Assessment Final Report	May 15 - May 21, 2002
NOA-BW-01-3Q	Nuclear Oversight Continuous Assessment Report Braidwood Generating Station July - September 2001	October 31, 2001
N-GRS2 04	Training Administrative System Nuclear-Respiratory Systems 2 (N-RS2X) Course Completion Report (From January 2001 to May 21, 2002)	May 21, 2002
	Year 2002 Crews	Revision 2
	Respiratory Qualifications Report (TE001)	May 23, 2002
TIMD412	Personnel Qualifications and Exam Data	May 24, 2002
FO 101538-29	Controls for RP Instrumentation (CR 106862)	May 6, 2002
FO 101538-32	Essential Elements: P1H-1, 2, and 3 (CR106708)	May 3, 2002
BWIP 2505-008	Calibration of GA Technologies ARMs	Revision 3E1
BwRP 5822-8	Operation and Calibration of the IPM-7/8/8D Whole Body Frisking Monitor	Revision 5
BwRP 5822-22	Operation and Calibration of the Eberline Model PRM-6	Revision 4
BwRP 5823-2	Operation and Calibration of the Merlin Gerin RAM Ion Dose Rate Meter	Revision 2
BwRP 5823-6	Operation and Calibration of the Eberline Model 6112 Teletector	Revision 2
BwRP 5824-4	Operation and Calibration of the Merlin Gerin CDM-21 Calibrator	Revision 4
RP-AA-103	Controls for Radiation Instrumentation	Revision 0

IR-PR011	Containment Atmosphere CAM Calibration	March 31, 2000
0RE-AR016	Primary Sample Room ARM Calibration	July 8, 1997
0RE-AR038	Fuel Handling Building Elevation 401 ARM Calibration	May 15, 1998
1RE-AR001	Containment ARM Elevation 426 Calibration	October 15, 1996
2RE-AR001	Containment ARM Elevation 426 Calibration	May 8, 1999
1RE-AR003	Incore Seal Table Elevation 401 ARM Calibration	October 15, 1996
2RE-AR003	Incore Seal Table Elevation 401 ARM Calibration	May 8, 1999
1RE-AR011	Containment Fuel Handling Incident ARM Calibration	March 23, 2000
2RE-AR011	Containment Fuel Handling Incident ARM Calibration	October 24, 2000
0RE-AR055	Fuel Building Fuel Handling Incident ARM Calibration	July 11, 2001
0RE-AR056	Fuel Building Fuel Handling Incident ARM Calibration	August 22, 2000
1RE-AR020	High Range Containment Elevation 514 ARM Calibration	March 24, 2000
2RE-AR020	High Range Containment Elevation 514 ARM Calibration	October 26, 2000
1RE-AR021	High Range Containment Elevation 514 ARM Calibration	March 24, 2000
1RE-AR021	High Range Containment Elevation 514 ARM Calibration	October 26, 2000
FASTSCAN	Whole Body Counter #2 Calibration	November 17, 2000
1587	Eberline Model PRM-6 Calibration	August 8, 2001
2094-063	Merlin Gerin Ram Ion Dose Rate Meter Calibration	August 8, 2001
37453	Eberline Model 6112 Teletector Calibration	August 8, 2001
140865, 140906, 140620, 137723	Merlin Gerin CDM-21 Calibrations	August 8, 2001
CR A2001-00548	Inconsistent Use of AMS-4	February 5, 2001

CR A2001-01822	Source Separated from Source Jig	June 18, 2001
	Braidwood Station Emergency Plan	January 2000
BwRP 5510-13T1	ISI Magnum Self-Contained Breathing Apparatus Checklist	June 7, 2001
	SCBA Mask Fits & Qualifications	August 8, 2001
	RP Self-Assessment Report 4th Quarter 2000	
	RP Self-Assessment Report 1st Quarter 2001	
	RP Self-Assessment Report 2nd Quarter 2001	
NOA-20-99-005	Braidwood Station Nuclear Oversight Assessment of the RP Program	January 8, 2001

3PP1 Access Authorization (AA) Program

SY-AA-102	Exelon's Nuclear Fitness-for-Duty Program	Revision 5
SY-AA-102-201	Call-Outs for Unscheduled Work	Revision 3
SY-AA-102-203	FFD [fitness-for-duty] Follow-up Testing	Revision 3
SY-AA-102-205	Fitness-for-Duty Appeal	Revision 2
SY-AA-102-221	Processing Fitness-for-Duty Allegations	Revision 1
SY-AA-103-512	Continual Behavioral Observation Program	Revision 3
TQ-AA-118	Nuclear General Employee Training-N-GET	Revision 3
Focus Area Self-Assessment	Fitness-for-Duty	May 29, 2001 - July 17, 2001
Security Event Reports		April, 2001 - May, 2002
Exelon Fitness-for-Duty Performance Data Reports - Second Period - 2001		March 1, 2002

3PP2 Access Control

SY-AA-101-112	Searching Personnel and Packages	Revision 5

SY-AA-101-115	Controlling Gates	Revision 2
SY-AA-101-117	Processing Visitors and Vehicles	Revision 5
SY-AA-101-119	Control of Receiving Warehouse	Revision 2
SY-AA-101-120	Control of Security Keys and Cores	Revision 1
SY-AA-101-122	Testing Security Equipment	Revision 5
SY-AA-101-123	Searching Vehicles and Cargo/Material	Revision 6
SY-AA-103-511	Request for Unescorted Access	Revision 7
SY-AA-103-514	Fabrication of Security Badges	Revision 6
SY-AA-103-518	Outprocessing of Personnel (Employee and Contractor)	Revision 4
LS-AA-125	Corrective Action Program (CAP) Procedure	Revision 2
CR 109325	Security Perimeter Performance Indicator	May 24, 2002
CR 109351	Alternate Methods of Form DD214 Verification Not Specified	May 24, 2002
CR 108172	Discrepancies Identified on Security Key Issuance Logs	April 25, 2002
Nuclear Oversight Continuous Assessment Report NOA-BW-02-1Q		January - March 2002
Focus Area Self-Assessment	Access Authorization	February 25, 2002 - March 22, 2002
Security Event Reports		April, 2001 - May, 2002

4OA2 Identification and Resolution of Problems

CR 074717	1SI18804B Trips Breaker When Trying to Stroke - Unplanned Limiting Condition for Operations	September 10, 2001
CR 096746	Parts Failed During Motor Control Center Breaker and Overload Surveillance	February 25 2002
CR 096945	Potential Rework of 2SX016B Molded Case Breaker Replacement	February 25, 2002

CR 104750	Molded Case Breakers Failed Test for 2FW009B	April 21, 2002 April 22, 2002
CR 105053	Westinghouse Breaker HFB3110ML Failed Surveillance - 2S18802B	April 21, 2002
CR 105186	Westinghouse Breaker for 2RH8716B Failed Acceptance Criteria	April 23, 2002
CR 105256	Breaker for 2CV8355D (2AP21E-B3) Failed Acceptance Criteria	April 24, 2002
CR 105053	Westinghouse Breaker HFB3110ML Failed Surveillance - 2SI8802B	April 21, 2002
CR 105397	Westinghouse Breaker for 2CV8355A Did Not Meet Acceptance Criteria	April 24, 2002
CR 105426	Westinghouse Molded Case Breaker Failed Test for 2FW009C	April 22, 2002
CR 105657	Potential Common Mode Failure with Westinghouse HFB Breakers	April 26, 2002
CR 105670	Westinghouse Molded Case Breaker Out of Tolerance - 2CV8100	April 26, 2002
CR 105984	Breaker Out of Tolerance 2AP28E-C3	April 29, 2002
WO 99222357 01	Trip Test Molded Case Breaker - Motor Control Center 232X4 Cubicle A1 2SX016 Molded Case Breaker RCFC 2B & 2D SX Inlet Valve	February 25, 2002
WO 99222778 01	2CV8355A Trip Test Molded Case Breaker	April 24, 2002
NEMA Standards Publication AB 4-1996	Guidelines for Inspection and Preventive Maintenance of Molded Case Circuit Breakers Used in Commercial and Industrial Applications	
EC 335908	Molded Case Circuit Breaker Instantaneous Trip Setting Changes	Revision 1
MA-AA-EM-4-00-00405	Molded Case Circuit Breaker Testing	Revision 1
DCP/EC 335668	Change the Instantaneous Setting of the Circuit Breaker for 2SX016B	February 26, 2002
LS-AA-105-1001	CR 96945 Supporting Operability Documentation: Auxiliary Power and Various System MOVs	Revision 0

Nuclear Operations Notification	Failure of MOV 1SI18804B to Stroke During Performance of 1BwOSR 5.5.8.SI-1B	September 10, 2001