September 13, 2000

Mr. Oliver D. Kingsley
President, Nuclear Generation Group
Commonwealth Edison Company
ATTN: Regulatory Services
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BRAIDWOOD- NRC INSPECTION REPORT 50-456/2000011(DRP); 50-457/2000011(DRP)

Dear Mr. Kingsley:

On August 21, 2000, the NRC completed an inspection at your Braidwood Units 1 and 2 reactor facilities. The results were discussed with Mr. Tulon and other members of your staff. The enclosed report presents the results of that inspection.

The inspection was an examination of 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. Within these areas the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel. Specifically, this inspection focused on resident inspection activities.

Based on the results of this inspection, three issues of very low safety significance (Green) were identified. One of these issues was determined to involve a violation of NRC requirements. However, the violation was not cited due to it's very low safety significance and because it was entered into your corrective action program. If you contest this non-cited violation or the severity level of the non-cited violations, 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 20555-001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the Braidwood facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be 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/

Michael J. Jordan, Chief
Reactor Projects Branch 3

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

Enclosure: Inspection Report 50-456/2000011(DRP);
50-456/2000011(DRP)

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U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report Nos:	50-456/2000011(DRP); 50-457/2000011(DRP)
Licensee:	Commonwealth Edison Company (ComEd)
Facility:	Braidwood Nuclear Power Station, Units 1 and 2
Location:	35100 S. Route 53 Suite 84 Braceville, IL 60407-9617
Dates:	June 30 through August 21, 2000
Inspectors:	C. Phillips, Senior Resident Inspector J. Adams, Resident Inspector D. Pelton, Resident Inspector N. Shah, Resident Inspector B. Kemker, Resident Inspector J. Belanger, Senior Physical Security Inspector J. Roman, Illinois Department of Nuclear Safety
Approved by:	Michael J. Jordan, Chief Reactor Projects Branch 3 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.

SUMMARY OF FINDINGS

NRC Inspection Report 50-456/2000011(DRP); 50-457/2000011(DRP), Commonwealth Edison Company, Braidwood Nuclear Power Station, Units 1 & 2, conducted between June 30 and August 21, 2000. The inspection was conducted of the following baseline activities: Equipment Alignment, Maintenance Risk Assessments And Emergency Work Control. The inspection was conducted by resident inspectors and a regional physical security inspector. This inspection identified three green issues, one of which was a non-cited violation. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the Significance Determination Process.

REACTOR SAFETY

Cornerstone: Mitigating Systems

- GREEN. The inspectors identified a Non-cited Violation because maintenance personnel violated station administrative procedures for equipment control of plant barriers and ran welding cables through a ventilation damper above the Unit 1A Safety Injection Pump Room Door without a pre-approved plant barrier impairment evaluation.

  The risk significance for running the cables through the ventilation damper was determined to be of very low safety significance because their placement was later determined not to impact the auxiliary building ventilation system's ability to perform it's safety function.

- GREEN. The inspectors identified that the operations management personnel took the 1B Essential Service Water Pump out-of-service for planned maintenance and were not aware of the risk impact due to having the 1A Motor Driven Feed Pump out-of-service for emergent work at the same time.

  The actual risk to the plant of having the 1B Essential Service Water Pump out-of-service in combination with the 1A Motor Driven Feed Pump did not change the plant risk status and therefore this issue was determined to be of very low safety significance.

Cornerstone: Initiating Event

- GREEN. The inspectors identified that operations management personnel were not aware of the impact on risk from the realignment of the Unit 2 heater drain pump flow control valves due to emergent work, which could have impacted the risk associated with planned maintenance.

  The actual risk to the plant of having the Unit 2 heater drain pump flow control valves realigned did not change the plant risk status and therefore this issue was determined to be of very low safety significance.

Report Details

Plant Status: Unit 1 operated at or near full power for the entire period. Unit 2 operated at or near full power until July 12, 2000, when a unit power reduction to 20 percent was commenced at 5:00 p.m. to perform repairs to a valve in main turbine electro-hydraulic control system. The unit was restored to full power at about 2:00 a.m. on July 14, 2000. Unit 2 operated at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Mitigating Systems, and Initiating Events

1R04 Equipment Alignment

.1 Equipment Alignment Verification of the 2A and Unit 0 Component Cooling Water (CC) Trains

a. Inspection Scope

The inspectors verified the system alignment of the Unit 0 and 2A CC trains while the 2B CC train was out-of-service for maintenance. The Unit 0 CC pump and heat exchanger were aligned to Unit 2 to support the unavailability of the 2B CC pump. The inspectors reviewed the system drawings and the following procedures to determine the correct system alignment:

- Braidwood Operating Procedure (BwOP) CC-10, “Alignment of the ‘0' CC Pump to a Unit,” Revision 13;
- BwOP CC-12, “Alignment of the ‘0' Heat Exchanger to a Unit,” Revision 6E1; and
- BwOP CC-14, “Post Loss of Coolant Accident Alignment of the CC System,” Revision 8.

The inspectors performed walkdowns of the accessible portions of the system and verified the system lineup and each of the system operating parameters (i.e., temperature, pressure, flow, etc.). In addition, the inspectors reviewed the Updated Final Safety Analysis Report and Technical Specifications.

b. Findings

There were no findings identified.

.2 Equipment Alignment of 1A Safety Injection (SI) Pump

a. Inspection Scope

The inspectors observed welding cables were routed through the ventilation damper above the 1A SI pump room door during maintenance activities on June 29, 2000. The inspectors interviewed operations shift management personnel and the ventilation system engineer. The inspectors also reviewed:

- CC-AA-201, “Plant Barrier Control Program,” Revision 3.

In addition, the inspectors evaluated the licensee’s corrective actions for plant barrier impairment issues documented in the following corrective action documents:

- Condition Report (CR) A2000-02731, “Welding Cable Through The Vent Damper Above The 1A SI pump room;”

- A2000-03195, “Corrective Actions For Welding Cable CR (A2000-02731) Are Too Narrowly Focused;” and

- Nuclear Tracking System Item 456-100-97-00504, “NRC Violation For Failure To Include A Safety Evaluation For An Auxiliary Building Doorway That Had Been Blocked Open.”

b. Findings

On June 29, 2000, maintenance personnel routed welding cables through the damper above the 1A SI pump room door without first getting an approved plant barrier impairment for the damper. Therefore the damper was not evaluated for impact on fire, security, ventilation, flood, high energy line break, radiation, missile and other design requirements as appropriate. Technical Specification 5.4.1.a states in part, “Written procedures shall be established, implemented, and maintained covering the following activities: a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.” Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 1c references administrative procedures for equipment control. Procedure CC-AA-201, “Plant Barrier Control Program,” Revision 3, Step 1.3, stated, “This procedure applies to plant barriers such as doors, floors, walls, roofs, gypsum board, structural fire proofing, electrical raceway fire barriers, heating, ventilation, and cooling [HVAC] plenum, ducting, floor drain seals, electrical, mechanical and instrument and control penetration seals, security barrier, lift out slabs or plugs, manhole covers, and spare penetrations. Those barriers SHALL be evaluated for impact on fire, security, ventilation, flood, high energy line break, radiation, missile and other design requirements as appropriate.” Contrary to the above, on June 29, 2000, welding cables were routed through an HVAC damper above the door to the 1A SI pump room without being evaluated for system impact. This was considered a **non-cited violation** of Technical Specification 5.4.1 (**50-456/2000011-01(DRP)).** This violation was entered into the licensee’s corrective action program as CR A2000-02731.

The inspectors verified that the placement of the welding cables did not impact the safety function of the 1A SI pump room ventilation damper to allow air flow from the room to the filtered ventilation system, but could have affected air flow depending on how the cables were placed through the damper. This issue was reviewed using the significance determination process assuming the cables would have made the auxiliary building ventilation inoperable in this room and was considered to be of very low safety significance (green).

The inspectors reviewed the approved corrective actions associated with the CR for the issue discussed above:

- A2000-02731, “Welding Cable Through The Vent Damper Above The 1A SI Pump Room.”

The apparent cause evaluation stated that neither the supervisor, nor the crew were aware of the procedural requirement to have an approved plant barrier impairment for running cables through the HVAC damper. The corrective actions consisted of discussing the incident with the mechanical maintenance shop. The inspectors considered the corrective action too narrowly focused. Without knowing why the mechanical maintenance personnel were unaware of procedural requirements a similar problem may exist in the electrical and instrument maintenance departments as well. The licensee placed this concern in the corrective action program in CR A2000-03195, “Corrective Actions For Welding Cable CR (A2000-02731) Are Too Narrowly Focused.”

1R05 Fire Protection

Unit 1 Lower Cable Spreading Room

a. Inspection Scope

The inspectors evaluated the licensees fire protection controls for the Unit 1 Lower Cable Spreading Room (licensee fire zone 3.2A-1). This area was selected, because it had a high, associated fire induced core damage frequency. Specifically, the inspectors performed a walkdown of the area to observe conditions related to the control of transient combustibles and ignition sources; the material condition, operational lineup, and operational effectiveness of fire protection systems, equipment and features; and the material condition and operational status of fire barriers. The inspectors verified that the area (including associated fire protection and mitigation equipment) was as described in the Braidwood Fire Protection Plan, dated December 1988. The inspectors also verified that appropriate compensatory measures were taken on August 15, 2000, when the gaseous fire protection system was declared inoperable due to ongoing work. The inspectors reviewed the following documents:

- Braidwood Fire Protection Plan, dated December 1988.

- CR A2000-03241, “Housekeeping Deficiency Identified By NRC,” which was written because of observations made by the inspectors during the inspection.

- TRANSCO Test Report No. TR-159, “Fire and Hose Stream Tests of TCO-001

Cement Used in Electrical Conduit Penetrations."

- TRANSCO Test Report No. TR-161, "Fire and Hose Stream Tests of TCO-001 Cement, TCO-002 Medium Density Silicone, and TCO-007 Silicone Adhesive Used in Electrical Conduit and Blockout Penetrations."

- TRANSCO Seal Detail Drawing Nos. BR-E-05A and BR-E-01

- Braidwood Seal Penetration Package Nos. E1454043A0, E1371022B2, E1361043, and E1457043A0

b. Findings

There were no findings identified.

1R12 Maintenance Rule Implementation

.1 Maintenance Rule Implementation of Deficiencies Associated With the 2B Centrifugal Charging Pump

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, as it pertained to identified performance problems with the 2B Centrifugal Charging Pump that had been documented in the following problem identification forms (PIFs) and CRs:

- PIF A1999-02946, "2CV01PB Increased Seal Leakage When Shutdown;"

- PIF A1999-03489, "2B CV [Chemical and Volume Control] Pump Inboard Seal Leakage During Return to Service;"

- PIF A1997-03657, "2B CV Pump Seal Line Leak;"

- CR A2000-00027, "2B CV Pump Outboard Seal Leakage;"

- CR A2000-00272, "Maintenance Rule Unavailability for CV-2 Criteria is Approaching Limit;"

- CR A2000-00533, "Potential Rework 2B CV Pump Seal Injection Leaking on Return to Service;" and

- CR A2000-00551, "2B CV Pump Seal Flush Line Leak on Return to Service."

During this inspection, the inspectors evaluated the licensee's monitoring and trending of performance data, verified that performance criteria were established commensurate with safety, verified that the equipment problems were appropriately evaluated in accordance with the maintenance rule, and evaluated the appropriateness of a(1) goals and corrective actions. The inspectors interviewed the station's maintenance rule coordinator and reviewed Nuclear Station Procedure ER-AA-310, "Maintenance Rule," Revision 0.

b. Findings

There were no findings identified.

.2 Maintenance Rule Implementation of Deficiencies Associated With the Unit 1 Instrument Air (IA) System

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, as it pertained to the improper installation of a solenoid valve in the IA system. The inspectors reviewed the following:

- CR A2000-00965, "Potential rework - Solenoid Valve Installed Backwards on Unit 1 Air Dryer;"
- CR A2000-00649, "U-1 IA Dryer Malfunction;"
- Braidwood maintenance rule performance criteria for the IA system;
- Braidwood maintenance rule unavailability data for the IA system;
- Braidwood Maintenance Procedure BwHP 4006-020, "Replacement of ASCO Solenoid Valves," Revision 6;
- Braidwood maintenance rule reliability data for the IA system; and
- Nuclear Station Procedure ER-AA-310, "Maintenance Rule," Revision 0.

The inspectors verified that the IA system performance criteria was commensurate with safety; verified that the licensee identified IA system problems, related to the maintenance rule, in their corrective action program; and verified that the improperly installed IA system valve was properly evaluated in accordance with the maintenance rule.

b. Findings

There were no findings identified.

.3 Maintenance Rule Implementation of Deficiencies Associated With The Unit 1B SI System

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the Maintenance Rule, 10 CFR Part 50.65, as it pertained to identified performance problems with the 1B SI system. Specifically, these problems had been documented in the following PIFs:

- PIF A1999-03034, "Valve 1SI8812B Failed Quarterly Stroke Time Requirement;"

- PIF A1999-03812, “Late Entry into Unit 1 Emergency Core Cooling System Limiting Condition for Operation;”

- PIF A2000-01217, “Leakage From SI Test Line Weld Identified During Operating Mode 3 Inservice Inspections;” and

- PIF A2000-02386, “1B SI Pump Vibration Levels Continue to Increase into Alert Range.”

The inspectors evaluated the licensee’s monitoring and trending of performance data, verified that performance criteria was established commensurate with safety, and selectively verified that equipment failures were appropriately evaluated in accordance with the maintenance rule. In addition, the inspectors observed selective components of the 1B SI system to identify potential maintenance issues. The inspectors interviewed the station’s maintenance rule coordinator and reviewed station procedure ER-AA-310, “Maintenance Rule,” Revision 0, and verified licensee compliance with that procedure.

b. Findings

There were no findings identified.

1R13 Maintenance Risk Assessments And Emergency Work Control

.1 Risk Assessment of Planned Maintenance Associated With the 2A SI Train Work Window

a. Inspection Scope

The inspectors reviewed the licensee’s assessment and management of plant risk for planned maintenance activities on the 2A SI train. The inspectors selected this maintenance activity because it involved a system that was risk significant in the licensee’s risk analysis.

During this inspection, the inspectors assessed the operability of redundant train equipment verifying key safety functions were maintained, verified the proper use of the on-line risk monitoring software by the licensee, and evaluated the licensee’s implementation of actions to minimize plant risk. The inspectors verified that the licensee’s maintenance activity planning minimized the duration that the plant was subject to the increased risk and verified that plant personnel were informed of the increased risk. The inspectors attended shift briefings and daily status meetings to verify that the licensee took actions to maintain a heightened level of awareness of the plant risk status among plant personnel. The inspectors reviewed Nuclear Station Procedure WC-AA-103, “On-Line Maintenance,” Revision 0, and verified licensee compliance with that procedure.

b. Findings

There were no findings identified.

.2 Risk Assessment of Planned Maintenance Associated With the 2A Diesel Generator Work Window

a. Inspection Scope

The inspectors reviewed the licensee's assessment and management of plant risk for planned maintenance activities on the 2A diesel generator. The inspectors selected this maintenance activity because it involved a system that was risk significant in the licensee's risk analysis.

During this inspection, the inspectors assessed the operability of redundant train equipment, verifying key safety functions were maintained; assessed the cumulative risk due to the concurrent unavailability of the 2A diesel generator, Unit 2 service air compressor, and the 2A SI pump; verified the proper use of the on-line risk monitoring software by the licensee; and evaluated the licensee's implementation of actions to minimize plant risk. The inspectors verified that the licensee performed Braidwood Operating Surveillance Procedure 2BwOSR 3.8.1.1, "Unit Two Offsite Alternating Current Power Availability Weekly Surveillance," Revision 0, every eight hours during the period when the 2A diesel generator was unavailable. The inspectors verified that the licensee's maintenance planning minimized the duration that the plant was subject to the increased risk and verified that plant personnel were informed of the increased risk. The inspectors attended shift briefings and daily status meetings to verify that the licensee took actions to maintain a heightened level of awareness of the plant risk status among plant personnel. The inspectors reviewed Nuclear Station Procedure WC-AA-103, "On-Line Maintenance," Revision 0, and verified licensee compliance with that procedure.

b. Findings

There were no findings identified.

.3 Risk Assessment of Planned Maintenance Associated With The 2B CC Pump Work Window

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk for planned maintenance activities on the 2B CC water pump. The inspectors selected this maintenance activity because it involved a system which was risk significant in the licensee's risk analysis. During this inspection, the inspectors assessed the operability of redundant train equipment and evaluated the licensee's implementation of planned contingency actions to minimize plant risk, where appropriate. The inspectors also interviewed operations and work control department personnel and reviewed Nuclear Station Procedure WC-AA-103, "On-Line Maintenance," Revision 0.

b. Findings

There were no findings identified.

.4 Risk Assessment of Planned Maintenance Associated With The 2A Steam Generator Power Operated Relief Valve Work Windows

a. Inspection Scope

The inspectors reviewed the licensee’s evaluation of plant risk for planned maintenance activities on the 2A steam generator power operated relief valve. The inspectors selected these maintenance activities because they involved a component which was risk significant in the licensee’s risk analysis. During this inspection, the inspectors assessed the operability of redundant train equipment and evaluated the licensee’s implementation of planned contingency actions to minimize plant risk, where appropriate. The inspectors also interviewed operations and work control department personnel and reviewed Nuclear Station Procedure WC-AA-103, “On-Line Maintenance,” Revision 0, and verified licensee compliance with that procedure.

b. Findings

There were no findings identified.

.5 Emergent Work Associated With The Failure of the 1A Motor Driven Feedwater Pump

a. Inspection Scope

The inspectors evaluated the licensee’s assessment of plant risk and equipment configuration control associated with the performance of emergent maintenance activities on the Unit 1 motor driven feedwater pump following an apparent failure of the discharge check valve during the shutdown of the feedwater pump on July 8, 2000. The inspectors interviewed the operations shift manager and reviewed the Unit 1 control room operating logs to assess the licensee’s awareness of plant configuration and current risk status. The inspectors reviewed the following procedures:

- WC-AA-103, “On-Line Maintenance,” Revision 0; and
- CR A2000-02848, “Plant Safety Analysis Evaluation Not Performed In A Timely Manner,” which was written because of observations made by the inspectors during this inspection.

In addition, the inspectors searched the licensee’s corrective actions data base for issues regarding the failure to properly classify risk for emergent work conditions documented in CRs and found none.

b. Findings

On July 8, 2000, the 1A Motor Driven Feed Pump discharge check valve failed. The licensee removed the 1A Motor Driven Feed Pump from service, performed a risk analysis of the emergent condition, and documented the results in the Shift Managers Log. On July 10, 2000, operations department personnel removed the 1B Essential Service Water Pump from service for planned maintenance. Work control personnel had qualitatively evaluated the risk of removing both the 1A Motor Drive Feed Pump and

the 1B Essential Service Water Pump at the same time and determined that the plant risk status would not change, but did not communicate the risk evaluation to operations department management. The inspectors identified that operations department management had removed the 1B Essential Service Water Pump from service without knowing the impact on risk status caused by having both pumps out-of-service at the same time. The inspectors determined that removing equipment from service without knowledge of how plant risk status was impacted could result in a more significant problem. The inspectors verified that the actual risk status of Unit 1 did not change significantly and using the significance determination process, determined the issue to be of very low safety significance.

.6 Emergent Work Associated With The Failure of the 2HD046A Heater Drain Pump Discharge Valve

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk and equipment configuration control associated with the performance of emergent maintenance activities on the 2HD046A heater drain pump discharge valve following an apparent failure of the valve controller. The inspectors interviewed the operations shift manager and reviewed the Unit 2 control room operating logs to assess the licensee's awareness of plant configuration and current risk status. The inspectors also reviewed the following documents:

- WC-AA-103, "On-Line Maintenance," Revision 0;
- OP-AA-101-402,"Operating Records," Revision 2;
- CR A2000-02830,"2HD046A Stuck At 70% Open;" and
- CR A2000-03111, "Procedure Discrepancies Associated With Logging Risk Re-Evaluation Details" which was written because of observations made by the inspectors during this inspection.

In addition, the inspectors searched the licensee's corrective actions data base for issues regarding the failure to properly classify risk for emergent work conditions documented in CRs and found none.

b. Findings

Procedure WC-AA-103, Step 4.5.1, requires an assessment of risk to be conducted for emergent work and that the results of that assessment be documented in the shift manager's log. On July 14, 2000, the air pressure regulator for the controller for the heater drain pump discharge flow control valve (2HD046A) was supplying air at too low a pressure. The valves 2HD046A and 2HD046B operate in parallel to control heater drain flow. The valve 2HD046A stuck open at 70 percent. This was allowing too much heater drain flow causing level in the heater drain tank to drop, which impacted net positive suction head on the heater drain pumps. Had the heater drain tank level dropped too far, the heater drain pumps would have tripped resulting in a plant transient

and possibly a reactor trip. The licensee identified the problem in time and valved out the 2HD046A.

A qualitative risk assessment was performed by work control personnel for removing valve 2HD046A from service, but the results of that assessment were not communicated to the operating shift and therefore the shift was not aware of the results and did not log the results in the shift managers log. Thus they were not aware of potential change to the plant risk for planned maintenance activities. The qualitative risk assessment determined that plant risk did not change significantly. The inspectors verified that the actual risk status of Unit 2 did not change significantly and using the Significance Determination Process determined the issue to be of very low safety significance (green).

1R14 Personnel Performance During Nonroutine Plant Evolutions And Events

.1 Personnel Performance During An Unanticipated Closure Of A Containment Isolation Valve

a. Inspection Scope

On June 30, 2000, reactor coolant letdown containment isolation valve 1CV8152 unexpectedly went shut due to a loss of instrument air (IA), when a radiation protection technician stepped on an instrument air isolation valve inadvertently shutting it. The inspectors verified that operations personnel took the appropriate procedural and Technical Specification actions. The following documents were reviewed during this inspection:

- CR A2000-02744, “Inadvertent Closure of IA to 1CV8152 Results in Loss of Unit 1 Letdown Flow;”
- CR A2000-02745, “1CV8160 Declared Inoperable;”
- Braidwood Annunciator Response Procedure 1BwAR 2-1PR11J, [Alarm Response Procedure For The 1PR11J, Containment Air Radiation Monitor] Revision 1E1; and
- 1BwOA PRI-1, “Excessive Primary Plant Leakage,” Revision 55.

In addition, the inspectors searched the licensee’s corrective actions database for risk significant issues regarding the inadvertent mis-positioning of equipment since January 1, 2000, documented in CRs and found none.

b. Findings

There were no findings identified.

.2 Licensee Event Report (LER) Review

a. Inspection Scope

The inspectors evaluated personnel performance in response to nonroutine plant evolutions and events. The inspectors reviewed the following:

- LER 50-456/2000-001-00, “Manual Actuation of the Main Control Room Ventilation (VC) System due to Conservative Decision Making Based on environmental conditions;”
- BwAR 2-0PR32J, “Main Control Room Out Air in OA,” Revision 4;
- BwAR 2-0PR33J, “Main Control Room Out Air in OB,” Revision 4;
- BwOP VC-5, “Placing the Control Room Heating, Ventilation, and Air Conditioning System Makeup Filter Train and Recirculation Charcoal Absorber in Operation,” Revision 10;
- Braidwood Shift Managers Log dated February 8, 2000;
- LER 50-456/2000-002-00, “Main Steam Safety Valves Tested in Excess of Required Setpoint Due to Bonding Between the Disc and Nozzle Seat;”
- Braidwood Maintenance Procedure BwMP 3305-107, “Main Steam Safety Valves Lift Point Verification using the Fermanite Trevitest System,” Revision 6;
- LER 50-456/2000-003-00, “Missed Technical Specification Surveillance on the 1B Diesel Generator due to Personnel Error due to Poor Supervisory Control;”
- LER 50-457/2000-002-00, “Automatic Reactor Trip on Power Range Neutron Flux High Negative Rate due to Stationary Gripper Fuse FU15 Failure for Control Rod P10 Causing the Rod to Drop into the Core;”
- LER 50-457/2000-003-00, “Manually Opened Reactor Trip Breakers due to Detector / Encoder Card Failure in Digital Rod Position Indication System;” and
- LER 50-457/2000-003-01, “Manually Opened Reactor Trip Breakers due to Detector / Encoder Card Failure in Digital Rod Position Indication System.”

The inspectors verified that personnel responded in accordance with station procedures and training, verified that the licensee had identified problems with personnel performance during nonroutine plant evolutions and events and had entered these problems into their corrective actions program, and verified that corrective actions taken by the licensee were appropriate.

b. Findings

(Closed) LER 50-456/2000-001-00: "Manual Actuation of the Main VC System due to Conservative Decision Making Based on environmental conditions." Operators manually re-aligned the main VC system because of an alert condition on the particulate channel of the 0PR32J outside air intake radiation monitor. The licensee later verified that the rising radiation levels were due to naturally occurring radon during a temperature inversion.

Since the manual actuation of an engineered safety feature was due to a naturally occurring event the issue was considered of very low safety significance. There were no findings during the review of this LER. This event did not constitute a violation of NRC requirements.

(Closed)LER 50-456/2000-002-00: "Main Steam Safety Valves Tested in Excess of Required Set Point Due to Bonding Between the Disc and Nozzle Seat." Results from testing revealed that 8 of 20 steam generator safety relief valves lifted in excess of their set points by greater than the three percent Technical Specification tolerance. An evaluation of the test data was performed by ComEd Nuclear Fuels Management. This evaluation concluded that the acceptance criteria for the applicable UFSAR accident scenario were not exceeded.

Since the remaining safety relief valves were capable of performing the required safety function this item was considered of very low safety significance. There were no findings during the review of this LER.

(Closed)LER 50-456/2000-003-00: "Missed Technical Specification Surveillance on the 1B Diesel Generator due to Personnel Error Due to Poor Supervisory Control." Operators failed to perform surveillance requirement 3.8.1.1, which verifies that the qualified sources of offsite power are available, within one hour in accordance with the Technical Specification surveillance requirement.

Since the required qualified sources of offsite power were available this item was considered of very low safety significance. The failure to perform the Technical Specification Surveillance Requirement within one hour constitutes a violation of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy.

(Closed)LER 50-457/2000-002-00: "Automatic Reactor Trip on Power Range Neutron Flux High Negative Rate due to Stationary Gripper Fuse FU15 Failure for Control Rod P10 Causing the Rod to Drop into the Core." There were no findings during the review of this LER. This event did not constitute a violation of NRC requirements.

(Closed)LER 50-457/2000-003-00 and LER 50-457/2000-003-01: "Manually Opened Reactor Trip Breakers due to Detector / Encoder Card Failure in Digital Rod Position Indication System." There were no findings during the review of this LER. This event did not constitute a violation of NRC requirements.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors evaluated the licensee's determination of operability as described in CRs pertaining to an essential service water system valve that had exceeded a surveillance test acceptance criteria for minimum opening time and the sporadic operation of the 2B Auxiliary Feedwater (AF) Pump Speed Increaser Auxiliary Oil Pump. The inspectors also reviewed an evaluation of operability for the Motor Driven Auxiliary Feed Pumps with only the Auxiliary Building Charcoal Booster Fans operating. In addition, the inspectors reviewed the following:

- CR A2000-02919, "2B AF Pump Speed Increaser Aux Oil Pump Cycling;"
- AF system vendor drawing 4113-901-046;
- CR A2000-02847, "1SX027B Exceeds Open Stroke Alert Time Limit;"
- BwOS 0.5.SX.1 Test, "Essential Service Water Valve Stroke Quarterly Surveillance," Revision 1 (completed 10/8/098);
- 1BwOSR 5.5.8.SX-1A, "Essential Service Water Train A Valve Stroke Quarterly Surveillance," Revision 1;
- NSP-CC-3001, "Operability Determination Process," Revision 0;
- Operability Evaluation 95-080, "Operability Concern With Low Non-Accessible Plenum Flows;" and
- Calculation VA-102, "Auxiliary Building Energy Lead Calcs For Elevations 330', 346', 364', 383', 401', and 426' In Abnormal Condition."

The inspectors verified that the licensee had entered these degraded components into their corrective actions program, verified that the licensee had properly evaluated these components for operability, verified that appropriate corrective actions had been taken for the identified degraded conditions, and verified that the licensee had appropriately considered the impact of the degraded components on Technical Specification and Technical Requirements Manual Limiting Conditions for Operations.

b. Findings

There were no findings identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed the following to determine if the functional capability of the Unit 1 containment radiation monitor (1PR11J) impacted human reliability in responding to an initiating event:

- CR A2000-02748, “1PR11J Alarm Following Filter Change;” and

- OP-AA-101-303, “Operator Work-Around Program,” Revision 0.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed the post maintenance testing on the following equipment:

- Unit 2 SI valves 2SI8807A, 2SI8814, and 2SI8821A;

- Unit 2A Diesel Generator;

- Unit 2B CC Pump; and the

- Unit 2A Steam Generator Power Operated Relief Valve.

The inspectors verified that control room and engineering personnel were aware of the effect that the testing would have on plant operation. The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post maintenance testing performed; observed portions of the work surveillance; reviewed test data; verified that the testing demonstrated the capability of the systems and support equipment to perform their intended safety functions; and conducted walkdowns of the system and support equipment shortly after the completion of maintenance. The inspectors observed or reviewed the performance of the following test procedures:

- Work Request 990189673-01, dated August 1, 2000, “Unit 2A Diesel Generator Monthly Operability Surveillance;”

- 2BwOSR 3.8.1.2-1, “Unit 2A Diesel Generator Operability Monthly and Semi-Annual Surveillance,” Revision 3;

- Braidwood Engineering Surveillance Procedure 2BwVSR 5.5.8.SI.1, “American Society of Mechanical Engineering Surveillance Requirements For The 2A SI Pump,” Revision 2;

- 2BwOSR 5.5.8.SI-2A, "Train A SI System Isolation Valve Indication 18-Month Surveillance," Revision 0;

- 2BwOSR 5.5.8.SI-1A, "Train A SI System Valve Stroke Quarterly Surveillance," Revision 1;

- NSP ER-AA-301, "Rising Stem Motor Operated Valve Votes Testing Procedure," Revision 1;

- Work Request 990055211-02, dated December 18, 1999, "Motor Operated Valve 2SI8814 Diagnostic Test;"

- Work Request 990172788-01, dated July 30, 2000, "Unit 2 Train A SI System Valve Stroke Quarterly Surveillance;"

- Work Request 990174536-01, dated August 1, 2000, "American Society of Mechanical Engineers Surveillance Requirements for the Unit 2 Train A SI Pumps;"

- Work Request 990046621-01, dated July 30, 2000, "Unit 2 Train A SI Isolation Valve 18 Month Indication Surveillance (Valves 8807A, 8814 and 8821A);"

- Work Request 960004513-01, 2MS019A "Inspection/Re-Lube of Gearbox/Valve Stem;" and

- Work Request 990139713-01, 2CC01PB "Replace AR3 Relays with Direct Current Contractors Per D20-2-96-282-013."

b. Findings

There were no findings identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors evaluated the surveillance testing activities listed below. The inspectors witnessed surveillance testing and reviewed the test data and verified that the associated structures, systems, and components met the Technical Specification requirements; met the Updated Final Safety Analysis Report requirements, and licensee procedural requirements. The inspectors verified that in-service testing methods and acceptance criteria were in accordance with American Society of Mechanical Engineering Code, Section XI, and were consistent with the station's design basis.

The inspectors reviewed the following:

- 1BwVSR 5.5.8.DO.1, "American Society of Mechanical Engineering Requirement for Testing the Diesel Oil Transfer System," Revision 2;

- Braidwood Instrumentation Surveillance Procedure BwISR 3.3.2.10-217, 19

- "Operational Test/Surveillance Calibration of AF Pump Suction Loop 2P-AF051," Revision 0E2;

- 1BwOSR 3.3.2.7-611B, " Unit One Engineered Safety Feature Actuation System Instrumentation Slave Relay Surveillance (Train B Automatic SI - K611)," Revision 1;

- 1BwVSR 5.5.8.CV.2, "American Society of Mechanical Engineering Surveillance Requirements For The 1B Centrifugal Charging Pump and Check Valve 1CV8480B Stroke Test," Revision 2;

- BwVS 4.5.2.f.1.b, "Surveillance Requirement for 1B Centrifugal Charging Pump Differential Pressure," Revision 3;

- Work Request 990177611-01, "American Society of Mechanical Engineers Surveillance Requirements for 1CV01PB;"

- Work Request 990177612-02, "Technical Specification Differential Pressure Check;"

- 1 BwVSR 5.5.8.SX.2, "American Society of Mechanical Engineers Surveillance Requirements For 1B Essential Service Water Pump," Revision 1; and

- 2BwOSR 3.5.2.2-2,"Unit Two Emergency Core Cooling System Venting and Valve Alignment Surveillance," Revision 3.

b. Findings

There were no findings identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluations

a. Inspection Scope

On Monday, July 17, 2000, the inspectors observed operations control room personnel participate in an Emergency Preparedness Drill from the Braidwood control room simulator. The purpose of the inspection was to evaluate licensee drill conduct and the adequacy of licensee critique of performance. The inspectors also verified that the opportunities and successes or failures for classification, reporting and protective action recommendations were accurately captured for performance indicator reporting.

The inspectors interviewed operations crew members, the station director, and the Emergency Preparedness Coordinator. The inspectors also reviewed the following documents:

- A memorandum written by the Emergency Preparedness Coordinator documenting the licensee's evaluation of the drill results;

- Braidwood Emergency Plan Implementation Procedure (BwZP) 200-1, "Braidwood Emergency Action Levels," Revision 8;

- BwZP 200-1A1,"Braidwood Station Emergency Action Levels," Revision 11E2; and

- Log entries by all members participating in the drill at the Technical Support Center.

b. Findings

There were no findings identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 Security Plan Changes

a. Inspection Scope

During a regional assist inspection at the Braidwood Station on June 26-28, 2000, the inspector conducted a review of the following revisions to the Station Security Plan:

- Revision 45 (submitted by letter dated July 19, 1999);

- Revision 46 (submitted by letter dated October 19, 1999);

- Revision 47 (submitted by letter dated March 10, 2000); and

- Revision 48 (submitted by letter dated April 11, 2000)

b. Findings

The documents were submitted in a timely manner and the changes did not appear to reduce the effectiveness of the previous plan. There was one Unresolved Item (URI) identified in Revision 47. In Section 9.6 the licensee added a new requirement allowing the transfer of searched material/equipment between protected areas at different sites. The unresolved issue is that the language of the plan change did not adequately describe the methodology relating to the transportation of secure (searched) materials being transported from a licensee site to another licensee site

(URI 50-456/457/2000011-02(DRS.)). This same issue was identified during a security plan review relating to the LaSalle Station Security Plan. During this inspection, the Braidwood Station Security Administrator agreed to resubmit the change to provide more details on the search process. The inspector will conduct further evaluation of this issue upon receipt of this revision.

4. OTHER ACTIVITIES

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. Tulon and other members of licensee management at the conclusion of the inspection on August 21, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Tulon	Site Vice President
J. Harvey	Nuclear Oversight
T. Simpkin	Regulatory Assurance Manager
C. Dunn	Operations Manager
F. Lentine	Design Engineering Manager
J. Madden	Assistant System Engineer Manager
M. Finney	Engineering Supervisor
J. Giuffre	Mechanical Maintenance Superintendent
M. Cassidy	Regulatory Assurance - NRC Coordinator

NRC

M. Jordan	Branch Chief, Division of Reactor Projects
C. Phillips	Senior Resident Inspector
N. Shah	Resident Inspector

Illinois Department of Nuclear Safety

J. Roman	Resident Engineer

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-456/2000011-01	NCV	failure to follow plant barrier impairment procedures
50-456/457/2000011-02	URI	failure to provide adequate procedural requirements

Closed

50-456/2000-001-00	LER	manual actuation of VC system
50-456/2000-002-00	LER	main steam safety valves tested in excess
50-456/2000-003-00	LER	missed technical specification surveillance
50-457/2000-002-00	LER	automatic reactor trip
50-457/2000-003-00	LER	manually operated trip breakers
50-457/2000-003-01	LER	manually operated trip breakers
50-456/2000011-02	NCV	failure to follow plant barrier impairment procedures

Discussed

None

LIST OF BASELINE INSPECTIONS PERFORMED

The following inspectable-area procedures were used to perform inspections during the report period. Documented findings are contained in the body of the report.

Inspection Procedure		Report
Number	Title	Section
71111-04	Equipment Alignment	1R04
71111-05	Fire Protection	1R05
71111-12	Maintenance Rule Implementation	1R12
71111-13	Maintenance Risk Assessments And Emergency Work Control	1R13
71111-14	Personnel Performance During Nonroutine Plant Evolutions And Events	1R14
71111-15	Operability Evaluations	1R15
71111-16	Operator Workarounds	1R16
71111-19	Post Maintenance Testing	1R19
71111-22	Surveillance Testing	1R22
71114-06	Drill Evaluations	1EP6
71130-04	Security Plan Changes	3PP4

LIST OF ACRONYMS AND INITIALISMS USED

AF	Auxiliary Feedwater
BwAR	Braidwood Annunciator Response Procedure
BwHP	Braidwood Maintenance Procedure
BwIS	Braidwood Instrumentation Surveillance Procedure
BwOP	Braidwood Operating Procedure
BwOS	Braidwood Operating Surveillance Procedure
BwVS	Braidwood Engineering Surveillance Procedure
BwZP	Braidwood Emergency Plan Implementation
CC	Component Cooling Water
CV	Chemical and Volume Control
CFR	Code of Federal Regulations
CR	Condition Report
HVAC	Heating, Ventilating, and Cooling
IA	Instrument Air
LER	Licensee Event Report
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations
PIF	Problem Identification Form
SI	Safety Injection
URI	Unresolved Item
VC	Control Room Ventilation