December 12, 2001

EA-01-311

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2
NRC INSPECTION REPORT 50-456/01-11(DRP); 50-457/01-11DRP)
AND NOTICE OF VIOLATION

Dear Mr. Kingsley:

On November 19, 2001, the NRC completed an inspection at your Braidwood Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on November 19, 2001, 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. Specifically, this inspection focused on resident and regional specialist inspection activities.

Based on the results of this inspection, the inspectors identified an issue of very low safety significance (Green). The inspectors determined that instrument uncertainties associated with the ultimate heat sink average temperature were not assumed in design analyses and were not accounted for in the Technical Specification limit or associated testing acceptance criteria. The inspectors has also determined that a violation of NRC requirements is associated with this issue. This violation was evaluated in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), NUREG -1600. [The current Enforcement Policy is included on the NRC's website at www.nrc.gov/OE.] The violation is cited in the enclosed Notice of Violation (Notice) and the circumstances surrounding it are described in detail in the subject inspection report. The violation is being cited in the Notice because your staff disagreed with conclusions drawn by the inspectors as to the necessity to include instrument uncertainty when developing Technical Specification Surveillance Requirement acceptance criteria and has not placed this issue into the corrective action program.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Based on the results of this inspection, the NRC has also identified two additional issues that were evaluated under the risk significance determination process as having very low safety significance (green). The NRC has also determined that violations are associated with these issues. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section VI.A of the Enforcement Policy. If you contest a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, Region III, Resident Inspector and the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

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,

Original signed by
Geoffrey E. Grant

Geoffrey E. Grant, Director
Division of Reactor Projects

Enclosures: 1. Notice of Violation
2. Inspection Report 50-456/01-11(DRP);
50-457/01-11(DRP)

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

See Attached Distribution

DOCUMENT NAME: G:\brai\bra 2001-11drp.wpd

To receive a copy of this document, indicate in the box:"C" = Copy without enclosure "**E"=** Copy with enclosure"**N"=** No copy

OFFICE	RIII		RIII		RIII			
NAME	AMStone:dtp		BClayton		GGrant			
DATE	12/12/01		12/12/01		12/12/01			

OFFICIAL RECORD COPY

cc w/encls: J. Skolds, Chief Operating Officer
W. Bohlke, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President - Mid-West Regional Operating Group
J. Cotton, Senior Vice President - Operations Support
J. Benjamin, Vice President - Licensing and Regulatory Affairs
R. Hovey, Operations Vice President
K. Ainger, Director - Licensing
R. Helfrich, Senior Counsel, Nuclear
DCD - Licensing
J. von Suskil, Site Vice President
K. Schwartz, Plant Manager
A. Ferko, Regulatory Assurance Manager
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

cc w/encls: J. Skolds, Chief Operating Officer
W. Bohlke, Senior Vice President, Nuclear Services
C. Crane, Senior Vice President - Mid-West Regional Operating Group
J. Cotton, Senior Vice President - Operations Support
J. Benjamin, Vice President - Licensing and Regulatory Affairs
R. Hovey, Operations Vice President
K. Ainger, Director - Licensing
R. Helfrich, Senior Counsel, Nuclear
DCD - Licensing
J. von Suskil, Site Vice President
K. Schwartz, Plant Manager
A. Ferko, Regulatory Assurance Manager
M. Aguilar, Assistant Attorney General
Illinois Department of Nuclear Safety
State Liaison Officer
Chairman, Illinois Commerce Commission

ADAMS Distribution:
AJM
DFT
MLC
RidsNrrDipmIipb
GEG
HBC
CJP3
C. Ariano (hard copy)
DRPIII
DRSIII
PLB1
JRK1

NOTICE OF VIOLATION

Exelon Generation Company, LLC
Braidwood Station

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

During an NRC inspection conducted on October 1 through November 19, 2001, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

10 CFR 50, Appendix B, Criterion XI, states, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

The maximum analyzed design limit for essential service water temperature was 100 degrees Fahrenheit as referenced below. Instrument uncertainty of +/- 2.6 degrees Fahrenheit for 1TI-SX015A,B (main control board 1A, 1B SX pump discharge analog temperature gauges) was not accounted for in these analyses.

- The Updated Final Safety Analysis Section 9.2.2.1 stated, "The component cooling (CC) system design is based on the design-basis service water supply maximum temperature of 100 [degrees Fahrenheit]." The CC water system provided cooling water to the residual heat removal system and the spent fuel pool cooling system.

- In addition, the Updated Final Safety Analysis Section 6.2.1.1.3, listed the maximum temperature limit analyzed for essential service water inlet temperature for the containment heat removal system (reactor containment fan cooler heat exchanger) as 100 [degrees Fahrenheit].

- Finally, the Updated Final Safety Analysis Section 9.5.5.2 stated that the maximum essential service water inlet temperature to the emergency diesel generator jacket water cooling heat exchanger was 100 [degrees Fahrenheit].

Technical Specification Surveillance Requirement 3.7.9.2. required verification that the average water temperature of the ultimate heat sink (source of the essential service water system) was less than or equal to 100 [degrees Fahrenheit] every 24 hours.

Procedure 1(2)BwOSR 0.1-1,2,3, "Unit One - Modes 1, 2, and 3 Shiftly and Daily Operating Surveillance Data Sheet," Revision 4, was the implementing procedure for Surveillance Requirement 3.7.9.2. Surveillance Requirement 3.7.9.2 acceptance criteria as less than or equal to 100 degrees Fahrenheit.

Contrary to the above, since July 11, 2000, Operating Surveillance Procedure 1(2)BwOSR 0.1-1,2,3 was inadequate, in that, previously identified measurement instrument tolerance band of +/- 2.6 degrees Fahrenheit for 1TI-SX015A,B was not accounted for in the Surveillance Requirement 3.7.9.2 acceptance criteria. Therefore, the test program to assure the satisfactory performance of several safety related systems would have allowed the actual temperature of the essential service water system to exceed acceptance limits contained in applicable design documents.

This violation is associated with a green SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Exelon is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator, Region III, and a copy to the NRC Resident Inspector at the Braidwood Station, within 30 days of the date of the letter transmitting this Notice of Violation. This reply should be clearly marked as a "Reply to a Notice of Violation" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response will be placed in the NRC Public Document Room (PDR), to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be placed in the PDR without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.790(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

In accordance with 10 CFR 19.11, you may be required to post this Notice within two working days.

Dated at Lisle, Illinois
this 12th day of December 2001

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-456; 50-457
License Nos:	NPF-72; NPF-77
Report Nos:	50-456/01-11(DRP); 50-457/01-11(DRP)
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	35100 S. Route 53 Suite 84 Braceville, IL 60407-9617
Dates:	October 1 through November 19, 2001
Inspectors:	C. Phillips, Senior Resident Inspector N. Shah, Resident Inspector D. Chyu, Reactor Inspector R. Daley, Reactor Inspector M. Mitchell, Radiation Specialist D. Nelson, Radiation Specialist K. O'Brien, Reactor Engineer D. Schrum, Reactor Engineer S. Sheldon, Reactor Engineer J. Roman, Illinois Department of Nuclear Safety
Approved by:	Ann Marie Stone, Chief Branch 3 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000456-01-11(DRP), 05000457-01-11(DRP); on 10/01-11/19/01, Exelon Generation Company; Braidwood Station; Units 1 & 2. Refueling and outage activities, surveillance testing, and access control to radiologically significant areas.

This report covers a 6-week routine inspection, a baseline radiation monitoring instrumentation inspection, and a baseline maintenance rule inspection. The inspection was conducted by resident and regional specialists. Three Green findings were identified. One of the findings involved a Cited Violation and two of the findings involved Non-Cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process" (SDP). The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violations.

A. Inspector Identified Findings

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

- Green. The 1A reactor coolant pump first stage seal failed due to operators failing to follow procedural guidance during pump startup.

  This finding was determined to be of very low safety significance because the seal failure did not result in an actual loss of reactor coolant. A Non-Cited Violation of Technical Specification 5.4.1.a. was identified. (Section 1R20).

- Green. The licensee did not consider instrument inaccuracies when establishing the acceptance criteria for Technical Specification Surveillance Requirement 3.7.9.2, ultimate heat sink average temperature. This instrument tolerance band was not accounted for in design analyses.

  This finding was determined to be of very low safety significance because with the most conservative instrument inaccuracies applied to the actual maximum ultimate heat sink temperature recorded, the Technical Specification limit was not exceeded. The inspectors determined this failure to properly control test procedures was a violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control." The licensee disagreed with the inspectors' conclusions and did not place this issue into the corrective action program. Therefore, a Notice of Violation was issued. (Section 1R22)

Cornerstone: Occupational Radiation Safety

- Green. The licensee failed to barricade, conspicuously post, and install a flashing light activated as a warning device to control access to a high radiation area (greater than 1000 mrem/hour) located in the 1B Reactor Containment Fan Coolers plenum.

This finding was determined to be of very low safety significance because unauthorized entry into the inadequately controlled high radiation areas did not appear to occur and a substantial potential for an overexposure did not exist. A Non-Cited Violation of Technical Specification 5.7.2(d) was identified. (Section 20S1).

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 entered the inspection period in refueling outage A1R09, was restarted on October 11, 2001, and was synchronized to the grid at 2:05 p.m. on October 12. Unit 1 reached full power on October 16. As part of the restart, Unit 1 raised reactor power level to 3586.6 megawatts (thermal), completing the implementation of its full power uprate.

Unit 2 entered the period at full power, but gradually reduced power because of turbine generator end turn vibration concerns to about 63 percent between October 15 and October 17. Unit 2 stayed at about 63 percent power until October 21 when reactor power was gradually raised to about 83 percent. Reactor power stayed steady between October 21 and November 7 when Unit 2 was shut down to repair the generator. Unit 2 was made critical at 2:35 p.m. and was synchronized to the grid at 10:59 p.m. on November 16. Unit 2 reached full power at 12:51 p.m. on November 17.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Intergrity

1R04 Equipment Alignment (71111-04)

b. Inspection Scope

The inspectors reviewed selected issues that the licensee entered into its corrective action program to verify that identified problems were being entered into the program with the appropriate characterization and significance and that corrective actions were being completed in a timely manner.

c. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111-11)

a. Inspection Scope

The inspectors reviewed the implementation of the licensee's Licensed Operator Requalification Program by observing simulator training conducted on October 24, 2001. Specifically, the inspectors observed operator response to a simulated event involving a design basis steam generator tube rupture as described in licensee scenario 0161, dated August 24, 2001, Revision 0.

The inspectors observed whether the training was monitored by the licensee's staff and that deficiencies were identified and remediated. The inspectors also observed that operators effectively responded to alarms, communicated plant conditions, and made emergency declarations. The inspectors also selectively compared the simulator equipment to actual control room equipment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111-12)

Periodic Evaluation

a. Inspection Scope

The region-based inspectors reviewed the licensee's implementation and conformance with the maintenance rule. Specifically, the inspectors:

- verified that the periodic evaluation was completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 2 years).

- ensured that the licensee reviewed its goals, monitored Structures, Systems, and Components (SSCs) performance, reviewed industry operating experience, and made appropriate adjustments to the maintenance rule program as a result of the above activities;

- verified that the licensee balanced reliability and unavailability during the previous refueling cycle, including a review of safety significant SSCs;

- verified that (a)(1) goals were met, that corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and that (a)(1) activities and related goals were adjusted as needed; and

- verified that the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, and reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

The region-based inspectors examined the periodic evaluation report completed for the time period of January 2000 - April 2001. To evaluate the effectiveness of (a)(1) and (a)(2) activities, the inspectors examined a number of Braidwood Condition Reports (CR) (contained in the list of documents at the end of this report). In addition, the CRs were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the maintenance rule program documents were reviewed.

In addition, the resident inspectors reviewed the licensee's implementation of the maintenance rule, 10 CFR 50.65, as it pertained to identified performance problems with the following systems:

- Pressurizer system;
- Condensate/condensate booster; and
- Main feedwater.

The resident inspectors evaluated the licensee's monitoring and trending of performance data and the appropriateness of a(1) goals and corrective actions. Specifically, the inspectors determined whether performance criteria were established commensurate with safety and whether equipment problems were appropriately evaluated in accordance with the maintenance rule. The inspectors interviewed the stations maintenance rule coordinator and reviewed selective CRs to determine whether identified problems were being entered into the corrective action program with the appropriate characterization and significance.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments And Emergency Work Control (71111-13)

a. Inspection Scope

The inspectors reviewed the licensee's assessment and management of plant risk for planned maintenance and/or surveillance activities on the following system:

- Unit 1D power operated relief valve;
- Unit 2 A train flux rate trip removal; and
- Unit 2 forced outage.

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, and evaluated the availability of redundant train equipment. In particular, the inspectors observed whether licensee's operations and engineering staff were aware of the licensee's revised probabilistic risk assessment model which was issued on June 28, 2000. The inspectors also reviewed Nuclear Station Procedure WC-AA-103, "On-Line Maintenance," Revision 3, and evaluated licensee compliance with that procedure. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111-15)

a. Inspection Scope

The inspectors reviewed and evaluated the following operability evaluations:

- Nuclear Oversight identified a weld deficiency with valve 1CV8392A; Reactor head vent hose connection in containment;
- Potential non-conservatism in spent fuel pit analysis; and
- Wall thinning identified on service water suction piping (0SX01CF) in lake screen house.

The inspectors also reviewed the technical adequacy of the evaluations against the Technical Specification (TS), Updated Final Safety Analysis Report (UFSAR), and other design information; determined whether compensatory measures, if needed, were taken; and determined whether the evaluations were consistent with the requirements of RS-AA-105, "Operability Determination Process," Revision 0.

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

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111-16)

a. Inspection Scope

The inspectors interviewed operations and engineering staff involved with the operator workaround program. The inspectors also reviewed the following administrative and operations procedures listed at the end of this report to ensure that current design and operation of plant systems, which could negatively impact the operators' ability to control the plant and respond to transients, did not include operator workarounds. One CR was generated due to the inspectors' observation during this inspection.

The inspectors also reviewed the current operator workaround and challenge lists and some recently developed CRs to verify that identified problems had been appropriately characterized and that the proposed corrective actions were adequate and completed in a timely manner.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111-19)

a. Inspection Scope

The inspectors reviewed the post-maintenance testing associated with the following components:

- Unit 1B diesel driven auxiliary feedwater pump;
- Unit 1 main steam safety valves;
- Unit 1 power uprate implementation and
- Unit 1 full power uprate.

For each activity, the inspectors reviewed the applicable sections of the TS and UFSAR, and observed portions of the maintenance work. The inspectors also evaluated the adequacy of work controls (including foreign material exclusion [FME] controls), reviewed post-maintenance test data, and conducted walkdowns to verify system restoration after the testing was completed. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

For the Unit 1 full power uprate, the inspectors observed the licensee's implementation of the special test procedure governing the uprate, and observed the primary and secondary plant response, as power was raised to the revised, normal operating power level of 3586.6 megawatts (thermal).

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

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111-20)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the Unit 1 ninth refueling outage (A1RO9) conducted from September 9 to October 12, 2001.

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 documents listed at the end of this report were also used by the inspectors to evaluate this area.

The inspectors performed the following during this inspection:

- Observed reactivity control, inventory control and decay heat removal during plant cooldown (including starting the residual heat removal system) and portions of plant heatup;
- Observed licensee inventory management activities during periods when reactor water level was at or below the reactor vessel flange;
- Performed walkdowns of the residual heat removal system during periods when the alternate train was out-of-service for planned maintenance;
- Observed reactivity control and alignment of the fuel pool cooling and building ventilation systems during fuel movement;
- Observed reactor core unloading and reloading;
- Observed the following equipment out-of-service activities:
  - licensee administrative checklists for entering/exiting operational modes 5 and 6;
  - isolation/restoration of the common header to the Unit 1 charging pumps;
  - isolation/restoration of the Unit 1B diesel driven auxiliary feedwater pump and associated valves;
  - isolation/restoration of the Unit 1 reactor vessel flange leak detection manual isolation valves; and
  - isolation/restoration of the Unit 1A and B residual heat removal systems
- Observed the operability of reactor coolant system (RCS) instrumentation and compared channels and trains against one another;
- Verified proper electrical alignment during station switchyard work and observed the planned, electrical cross-tie of DC buses 111/211 and 112/212; and
- Performed periodic walkdowns of containment to observe the alignment of selected containment integrity devices (including temporary penetrations), the condition of the emergency core cooling system sumps, and the overall containment material condition following the licensee's containment closeout inspection.

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 1.

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

b. Findings

A finding of very low safety significance (Green) was identified when the 1A reactor coolant pump (RCP) first stage seal failed due to operator error. The inspectors determined that the failure to properly follow procedure was a Non-Cited Violation of TS 5.4.1.a.

On October 5, 2001, the operators started the 1A RCP with conflicting indications of first stage seal leak-off flow. The high range flow indicator read greater than 0.2 gallons per minute (gpm) while the low range flow indicator read less than 0.2 gpm. Operating

Procedure BwOP RC-1, “Startup Of A Reactor Coolant Pump,” Revision 12, Step D.6.b, stated, in part, that an RCP must not be started unless there is greater than 0.2 gpm #1 seal leak-off flow. Station management was convinced that there was a problem with the low range flow indicator, did not believe the existing indication, and directed the operators to start the pump. In addition, during the pump startup, the high range flow indication dropped below the normal operating range; however, the operators did not trip the RCP as instructed by Step E.3 of BwOP RC-1. The pump was not secured until about seven hours later after consultation with the Westinghouse representative. About 13 hours later, the operators restarted 1A RCP once sufficient seal flow was verified. About 6 hours after this pump start, the first stage seal water outlet temperatures and the lower seal water bearing temperatures began to slowly increase. The operators tripped the 1A RCP about 14 hours later, due to the rising first stage seal water outlet and the lower seal water bearing temperatures indicating that the first stage seal had failed.

The licensee determined that the root cause of this event was inherent outage schedule pressure coupled with a lack of a formal protocol for the outage control center management personnel to deal with rising technical issues.

This finding was considered more than minor, as the failure to follow a procedure resulted in the failure of an RCP seal and could be reasonably viewed as a precursor to a significant event. The inspectors entered the significance determination process using Manual Chapter 0609, Appendix G, “Shutdown Operations - Pressurized Water Reactor Cold Shutdown Operation Reactor Coolant System And Steam Generators Available For Decay Heat Removal. However, since there was no actual loss of RCS inventory due to the failed RCP seal, the issue screened out as Green.

Technical Specification 5.4.1, states, “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. Paragraph 3.a. of this Regulatory Guide states, in part, that procedures for startup of the reactor coolant system shall be prepared. The licensee established BwOP RC-1, “Reactor Coolant Pump Seal Failure,” Revision 12, as an implementing procedure for startup of the reactor coolant system. Contrary to the above, on October 5, 2001, licensee personnel failed to follow Step D.6.b of operating procedure BwOP RC-1, “Reactor Coolant Pump Seal Failure,” Revision 12, when they started the 1A RCP without having at least 0.2 gpm #1 seal leak off flow and Step E.3 when they failed to trip the RCPs immediately when the #1 seal leak off decreased during normal RCP startup to less than normal operating range per Attachment A. However, because this violation was of very low risk significance, was non-repetitive, and was captured in the licensee’s corrective action program, it is considered a Non-Cited Violation consistent with Section VI.A of the NRC enforcement policy
(NCV 50-456/457-01-11-01(DRP)).

1R22 Surveillance Testing (71111-22)

a. Inspection Scope

The inspectors reviewed the following surveillance activities:

- Unit 1 main steam isolation valve testing;
- Unit 2 B safety injection pump testing;
- Unit 1 reactor coolant system flow measurement testing;
- Unit 1 B solid state protection system bimonthly testing;
- Unit 1 B auxiliary feedwater pump suction loop calibration (1P-AF055); and
- Unit 2 B main steam isolation valve testing.

For each activity, the inspectors witnessed portions of the testing or reviewed the test data and determined if the associated structures, systems, and components met the American Society of Mechanical Engineers (ASME) operating criteria, TS and UFSAR technical and design requirements. For selected activities, the inspectors also reviewed past test results to evaluate any adverse trends and to determine whether past testing was performed using consistent protocols.

In addition, 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.

b. Findings

A finding of very low safety significance (Green) was identified when the inspectors observed that the licensee did not consider instrument inaccuracies when establishing the acceptance criteria for TS Surveillance Requirement 3.7.9.2. This instrument inaccuracy had not been included in design analyses. The inspectors determined this failure to properly control test procedures was a Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control."

The inspectors determined that the maximum analyzed design limit for essential service water temperature was 100 degrees Fahrenheit as described below. These analyses did not account for instrument tolerance band of +/- 2.6 degrees Fahrenheit for 1TI-SX015A,B (main control board 1A, 1B SX pump discharge analog temperature gauges).

- The UFSAR Section 9.2.2.1 stated, "the CC system design is based on the design-basis service water supply maximum temperature of 100 [degrees Fahrenheit]." The CC water system provided cooling water to the residual heat removal system and the spent fuel pool cooling system.

- In addition, the UFSAR Section 6.2.1.1.3, listed the maximum temperature limit analyzed for essential service water inlet temperature for the containment heat removal system (reactor containment fan cooler [RCFC] heat exchanger) as 100 [degrees Fahrenheit].

- Finally, the UFSAR Section 9.5.5.2 stated that the maximum essential service water inlet temperature to the emergency diesel generator jacket water cooling heat exchanger was 100 [degrees Fahrenheit].

Technical Specification Surveillance Requirement 3.7.9.2. required verification that the average water temperature of the ultimate heat sink (source of the essential service water system) was less than or equal to 100 [degrees Fahrenheit] every 24 hours. Procedure 1BwOSR 0.1-1,2,3, "Unit One - Modes 1, 2, and 3 Shiftly and Daily Operating Surveillance Data Sheet," Revision 4, listed Surveillance Requirement 3.7.9.2 acceptance criteria as less than or equal to 100 degrees Fahrenheit.

The inspectors also observed that the licensee did not consider instrument inaccuracies when establishing the acceptance criteria for the 1A, 1B, 2A, and 2B SX pump discharge temperature indicators. For example, in work request package 990154317-01, the "as-left" calibration acceptance criteria for 1TI-SX015A,B (main control board 1A, 1B SX pump discharge analog temperature gauges) was +/-2.6 degrees Fahrenheit. As allowed by procedure, an instrument maintenance technician could leave the instrumentation in a condition such that the indicated gauge temperature could be at the TS limit of 100 degrees Fahrenheit while the actual temperature could be as high as 102.6 degrees. Licensee management personnel stated that this was an acceptable practice as identified in URI 50-456/457/01-09-01.

The inspectors consulted with NRR personnel and determined that the conclusions drawn on two previous Task Interface Agreements (TIA) for Millstone and Susquehanna regarding instrument uncertainties in surveillance testing acceptance criteria were applicable to the Braidwood issue. Specifically, because the ultimate heat sink limiting condition for operation provides operability determination criteria and confirmation that a design limit is met, the NRC concluded that associated testing must establish limits and/or acceptance criteria which include instrument inaccuracies. Because measured values involve uncertainty, this uncertainty must be accounted for when performing surveillance tests either by including the uncertainty in the limiting value that is established for the surveillance or by adjusting the measured value to include the uncertainty. In the Braidwood case, the analysis value, the TS limit, and the TS surveillance requirement acceptance criteria are the same. The licensee has not demonstrated that any of the heat load analyses included margin for measurement uncertainty or that the actual measurement uncertainty did not exceed analyses limits. (The TIAs are located in ADAMS as ascension numbers ML013460185 and ML013460191, respectively.)

Therefore, it is possible for actual essential service water temperature to be above the design basis limit and still pass the surveillance requirement. Based on this, the inspectors concluded that the issue had a credible impact on safety and that this issue could credibly affect the operability, availability, reliability, or function of a system or train in a mitigating system. The inspectors also concluded that this issue could affect the integrity of the reactor containment during a design-basis accident. However, since the inspectors could not identify a time when the ultimate heat sink temperature actually exceeded 100 degrees Fahrenheit, even with the worst case instrument uncertainty applied, there was no actual loss of safety function and this issue screened out as green. The failure to incorporate the requirements and acceptance limits contained in applicable design documents in testing procedures was a violation of 10 CFR 50, Appendix B, Criterion XI, (VIO 50-456/457-01-11-02(DRP)). Unresolved item URI 50-456/457/01-09-01 is closed to this violation (Section 40A5.3).

The root cause of this violation was the licensee's belief that enough margin was implicitly available in existing calculation assumptions to account for temperature measurement instrument uncertainty. The licensee disagreed with the inspectors' conclusion that they were not in compliance with 10 CFR 50, Appendix B, Criterion XI, and did not enter this issue into the corrective action program. The licensee had not presented information to the inspectors which satisfactorily demonstrated that the assumed margins in existing calculations would bound the existing temperature measurement instrument uncertainty.

1R23 Temporary Plant Modifications (71111-23)

a. Inspection Scope

The inspectors evaluated the licensee's installation of the following temporary modification:

- Essential service water room sump pump control circuitry

Specifically, the inspectors reviewed the UFSAR Report to determine whether the licensee adequately addressed system operability, design requirements, configuration control, risk significance, and post-installation testing.

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 and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The regional radiation specialist 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 radiation areas) in the Unit 1 containment building and auxiliary buildings, 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

TSs. The inspectors also reviewed the radiological conditions within those work areas walked down, to assess the radiological housekeeping and contamination controls.

The inspectors reviewed a selection of RWPs used to access radiologically significant work areas (radiation areas and high radiation areas (HRAs) during the Unit 1 refueling outage, A1R09. Work activities in those areas included reactor head work, seal table work, reactor cavity decontamination, RCP A and D seal and motor work, and fuel moves during refueling. The inspectors reviewed the RWPs to verify that they contained adequate work control instructions. In the case of HRA access, the inspectors reviewed the RWP controls to verify that the licensee complied with the specific requirements contained in the TSs. The inspectors also reviewed electronic dosimeter alarm setpoints and compared them to area radiation levels and expected personnel exposures to verify that the alarm setpoints were adequately determined. The inspectors also evaluated established work controls to determine if worker exposures were maintained As-Low-As-Reasonably-Achievable (ALARA).

b. Findings

No findings of significance were identified.

.2 High Dose Rate HRA and Very HRA Controls

a. Inspection Scope

The regional radiation specialist inspectors reviewed the licensee's controls for high dose rate HRAs and very HRAs. In particular, the inspectors reviewed the licensee's procedures for posting and controlling HRAs to verify the licensee's compliance with 10 CFR Part 20 and its TSs. The inspectors also reviewed licensee records of HRA boundary and posting surveillances for calendar year 2001 and performed walkdowns to verify the adequacy of boundaries, controls, and postings. In addition, the inspectors reviewed the licensee's controls for highly irradiated material that was stored in spent fuel storage pools to verify that the licensee implemented adequate measures to prevent inadvertent personnel exposures from these materials.

b. Findings

A finding of very low safety significance (Green) was identified on October 2, 2001, when the licensee found a high dose rate trash bag not controlled as a Locked (greater than 1000 mrem/hr at 12 inches) High Radiation Area controls (LHRA). The bag of waste was found improperly stored in the Danger HRA inside the Unit 1B RCFC plenum. The Unit 1B RCFC plenum is located in Unit 1 containment. The dose rates on the bag were 10,000 mrem/hr on contact and 1500 mrem/hr at 12 inches. This issue was dispositioned as a Non-Cited Violation (NCV) of TS 5.7.2 for the failure to properly control access to LHRA.

The inspectors identified that the failure to barricade, conspicuously post, and not activate a flashing light as a warning device on October 2, 2001, to control access to Unit 1B RCFC plenum did not meet the LHRA access control requirements of TS 5.7.2(d). This finding, if uncorrected, would become a more significant safety issue because the required controls provide an important radiological barrier to obstruct inadvertent entry into a LHRA and prevent unintended radiation exposure. Based on worker electronic dosimetry alarm data generated and reviewed by the licensee, it does not appear that unauthorized personnel entered the LHRA that existed while the bag of waste was stored in the plenum. The inspectors evaluated the risk significance of this issue using the Occupational Radiation Safety Significance Determination Process (Appendix C to NRC Manual Chapter 0609, "Significance Determination Process"), and determined that there was not a substantial potential for an overexposure, nor would the licensee's ability to assess worker dose be compromised should an individual have climbed up the ladder onto one of the platforms. Therefore, the issue was determined to be of very low safety significance (Green).

Technical Specification 5.7.2(d) requires that for HRAs accessible to personnel with radiation levels of greater than 1000 mrem/hour at 30 cm (12 inches) that are located within larger areas such as reactor containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, the individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device. The failure to barricade, conspicuously post, and install a flashing light as a warning device inside the Unit 1B RCFC plenum was a violation of TS 5.7.2(d). However, because upon discovery the licensee immediately took action to control the Unit 1B RCFC plenum as a LHRA and subsequently placed this issue into its corrective action program (Braidwood Action Request AR00077476), this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC enforcement policy (NCV 50-456/01-011-03; NCV 50-475/01-011-03).

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

.1 Job Site Inspections and ALARA Control

a. Inspection Scope

The regional radiation specialist inspectors selected a number of A1RO9 refueling outage high exposure or HRA work activities to evaluate the licensee's use of ALARA controls for each activity.

The inspectors reviewed ALARA plans for each activity and observed work associated with each activity. The inspectors evaluated the licensee's use of engineering controls to achieve dose reductions. The inspectors also determined if workers were utilizing the low dose waiting areas for each activity and whether the first-line supervisor for each job ensured that the jobs were conducted in a dose efficient manner. The inspectors also reviewed individual exposures of selected work groups to determine if there were any significant exposure variations which may exist among workers.

b. Findings

No findings of significance were identified.

.2 Source Term Reduction and Control

a. Inspection Scope

The regional radiation specialist inspectors evaluated the licensee's source term reduction program in order to verify that the licensee had an effective program in place, and was knowledgeable of plant source term and techniques for its reduction. Areas of review included:

- The installation of permanent shielding and scaffolding;
- The hot spot reduction program;
- Additional system flushes;
- Research Zinc injection; and
- Increased letdown during hydrogen peroxide addition.

b. Findings

No findings of significance were identified.

.3 Radiological Work Planning

a. Inspection Scope

The regional radiation specialist inspectors selected high collective dose A1RO9 refueling outage job activities to assess the adequacy of the radiological controls and work planning. For each job activity, the inspectors reviewed ALARA evaluations including initial reviews, in-progress reviews, and associated dose mitigation techniques and evaluated the licensee's exposure estimates and performance. The inspectors also assessed the integration of ALARA requirements into work packages to evaluate the licensee's communication of radiological work controls.

The inspectors reviewed the exposure results for the selected activities to evaluate the accuracy of exposure estimates in the ALARA plan. 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, if additional engineering/dose controls had been established and if required corrective documents had been generated.

b. Findings

No findings of significance were identified.

.4 Verification of Exposure Goals and Exposure Tracking System

a. Inspection Scope

The regional radiation specialist inspectors reviewed the methodology and assumptions used for A1RO9 refueling outage exposure estimates and exposure goals and compared job dose rate and man-hour estimates for accuracy. The inspectors examined job dose history files and dose reductions anticipated through lessons learned to verify that the licensee appropriately forecasted outage doses. The inspectors also reviewed the licensee's exposure tracking system to determine if the level of exposure tracking detail, exposure report timeliness and exposure report distribution was sufficient to support control of collective exposures.

a. Findings

No findings of significance were identified.

.5 Declared Pregnant Workers

a. Inspection Scope

The regional radiation specialist inspectors reviewed the controls implemented by the licensee for controlling declared pregnant worker dose. Specifically, the inspectors reviewed the licensee's adherence to the requirements contained in 10 CFR 20.1208 and its procedures, and reviewed the licensee's evaluation of the dose to the individual's embryos/fetus to verify that appropriate limitations were implemented to control dose from both external and internal sources.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The regional radiation specialist inspectors reviewed the licensee's self-assessments and audits, since the last outage, as well as selected outage generated action requests, which focused on ALARA planning and Access Controls. The inspectors evaluated the effectiveness of the licensees self-assessment process to identify, characterize, and prioritize problems. The inspectors reviewed the licensee's ability to identify repetitive problems, contributing causes, the extent of conditions, and corrective actions which would achieve lasting results.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Shipping Records

a. Inspection Scope

The regional radiation specialist inspectors reviewed one non-excepted package shipment manifest completed on May 16, 2001, to verify compliance with NRC and Department of Transportation requirements (i.e., 10 CFR Parts 20 and 71 and 49 CFR Parts 172 and 173).

b. Findings

No findings of significance were identified.

1. OTHER ACTIVITIES

40A1 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors reviewed whether the licensee was accurately reporting data for the following performance indicators:

- Emergency diesel generator unavailability.

The inspectors reviewed system operating logs and licensee monthly operating reports submitted to the NRC, and interviewed licensee engineering and operations staff to determine whether the performance indicator data was being collected and reported consistent with the guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 1.

b. Findings

No findings of significance were identified.

4OA5 Other

.1 (Open) Unresolved Item (URI) 50-456/00-06-02; 50-457/00-06-02: Licensing requirements reduced for two auxiliary building fire zones. The licensee provided additional information concerning this item in a letter dated February 7, 2001, as requested in Inspection Report 50-456/00-06; 50-457/00-06. The NRC will continue to review Braidwood's licensing basis as discussed in Amendments 3 and 7 to the Fire Protection Report, Byron Safety Evaluation Report and subsequent supplements (NUREG 876) and Braidwood Safety Evaluation Report and subsequent supplements (NUREG 1002) to determine the applicability of the requirement for an area-wide suppression system in these fire zones.

While reviewing this item, the inspectors identified a change to the fire protection program which appeared to reduce the fire protection program effectiveness and adversely affect the licensee's safe shutdown capabilities in the event of a fire. The licensee's approved fire protection program required the licensee to develop and maintain an administrative procedure that established requirements for implementing fire watch activities in response to identified inoperable fire suppression systems. The inspectors reviewed administrative procedure, BwAP 1110-1, "Fire Protection System Requirements," Revision 0, dated January 20, 1987. This procedure, considered as a part of the licensee's fire protection program, was submitted to the NRC for review and approval. At that time, the procedure required the following when a water suppression system was identified as being inoperable:

- Within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; and
- For other areas, establish an hourly fire watch patrol.

The administrative procedure in effect during this inspection period was BwAP 1110-1, Revision 15, which contained requirements less stringent than Revision 0 and required the following if the water suppression systems were inoperable:

- Establish a continuous fire watch for Fire Zone 11.3-0, auxiliary building elevation 364' CC pumps area;
- Establish an hourly fire watch with operable automatic fire detection instrumentation for auxiliary building 364' containment pipe penetration areas (Fire Zones 11.3-1 and 11.3-2) and for auxiliary building, general area center stairway (Fire Zones 11.2-0, 11.3-0, 11.4-0 11.5-0 and 11.6-0); and
- For all other areas, no fire watch is required if automatic detection instrumentation is verified available.

This change in fire watch requirements appeared to reduce the effectiveness of the plant's fire protection program because a fire area that originally required a continuous fire watch under the original approved fire protection program now only required an hourly fire watch or no fire watch at all. Furthermore, this reduction in fire protection program effectiveness may have an adverse impact on the licensee's ability to achieve and maintain safe shutdown conditions in the event of a fire when a continuous fire watch was not used to compensate for the lack of suppression systems in fire areas where redundant equipment could be damaged. Fire damage caused by a localized fire could spread, affecting redundant safe shutdown equipment, thereby imposing more challenges on plant operators tasked with implementing time critical safe shutdown activities.

The Braidwood Station's Facility Operating Licenses, NPF-72 and NPF-77, Section 2.E, required that the licensee implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR. The licensee could make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the licensee's ability to achieve and maintain safe shutdown conditions in the event of a fire.

Byron/Braidwood UFSAR, Section 9.5.1, "Fire Protection Systems," stated, in part, that the design bases, system description, safety evaluation, inspection and testing requirements, personnel qualification, and training are described in Byron/Braidwood Station Fire Protection Report in Response to Appendix A of Branch Technical Position APCSB 9.5-1 (also known as the Fire Protection Report).

Section II.A, "Fire Protection Program," in Appendix A5.7 of the Fire Protection Report, stated, in part, that administrative procedures provide fire watch in areas where detection or suppression systems are inoperable.

The licensee changed the administrative procedure, which established fire watch requirements, to reduce or eliminate fire watch requirements in areas where detectors or suppression systems were inoperable. The inspectors determined that the licensee did not seek NRC approval prior to implementing this change which appeared to reduce the effectiveness of the fire protection program and adversely affect the licensee's ability to achieve and maintain safe shutdown conditions in the event of a fire. This item will be treated as part of the URI pending further review by the NRC of regulatory requirement for the lack of suppression system in Fire Zones 11.5-0 and 11.6-0.

.2 (Closed) URI 50-456/457/00-06-04 and URI 50-456/457/00-06-05: Alternative shutdown capability was not independent of Fire Zones 11.5-0 and 11.6-0 and did not ensure integrity of the primary coolant boundary for Fire Zone 11.5-0. The items involved a condition in which a spurious operation of the volume control tank outlet valves could result in the loss of suction to the charging pumps. Furthermore, the reactor water storage tank outlet valves were susceptible to mechanical damage such that they could not manually be aligned to supply a suction source to the charging pumps. Also, the URIs postulated the spurious closing of the CC water supply valves to the RCP thermal barrier heat exchangers with a concurrent loss of RCP seal injection flow due to spurious closure of the seal injection flow path valves. These concurrent spurious operations had the potential to overheat the RCP seals causing a seal rupture.

The inspectors determined that in the case of the volume control tank outlet valves, the licensee had prescriptive manual actions in place in their operations procedures to mitigate the spurious closing of the valves. The prescriptive action ensured that the alternate suction source would be available by manually opening the reactor water storage tank outlet valves. Additionally, the licensee maintained procedures that would ensure tripping the RCPs prior to seal temperature reaching 235 degrees Fahrenheit. However, the temperature instrumentation for the RCP seals had not been analyzed for use in a fire scenario creating the possibility that during a fire the operators would not know when to trip the RCPs. While the temperature indication was not analyzed, the licensee had procedures in place which ensured through manual actions that the seal injection flow path valves would be open. Because these prescriptive actions were in place at the time of the original triennial fire protection inspection, the inspectors determined that safe shutdown could have been achieved. No violations of NRC requirements were identified and these items are considered closed.

.3 (Closed)URI 50-456/457/01-09-01: Temperature measurement instrument uncertainty was not applied to TS Surveillance Requirement 3.7.9.2 acceptance criteria. This issue is discussed in detail in paragraph 1R22 of this report and resulted in a Notice of Violation of 10 CFR 50, Appendix B, Criterion XI, "Test Control." This item is closed.

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 November 19, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

Interim Exit Meetings

The maintenance rule inspectors presented the inspection results to Mr. J. von Suskil and other members of licensee management at the conclusion of the inspection on November 8, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

The radiation specialist inspectors presented the inspection results to Mr. J. von Suskil and other members of licensee management at the conclusion of the inspection on September 28, 2001 and October 4, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

The reactor inspectors presented the inspection results to Mr. J. von Suskil and other members of licensee management at the conclusion of the inspection on October 18, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following finding of very low significance (Green) was identified by the licensee and is a violation of NRC requirements which met the criteria of Section VI of the NRC Enforcement Manual, NUREG-1600 for being dispositioned as Non-Cited Violation (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV456/01-11-04	Technical Specification 5.4.1.c. requires written procedures shall be established, implemented, and maintained for Fire Protection Program Implementation. The Fire Protection Program was implemented, in part, by procedure OP-AA-201-004, "Fire Prevention for Hot Work." Condition Report 00079302 cited 13 examples of the failure to follow OP-AA-201-004 during the licensee's Unit 1 Spring 2001 refueling outage.

KEY POINTS OF CONTACT

Licensee

J. Bailey, Regulatory Assurance - NRC Coordinator
G. Baker, Security Manager
S. Chingo, Cantera, Exelon
C. Chovan, Work Management Director
G. Dudek, Operations Manager
C. Dunn, Engineering Director
A. Ferko, Regulatory Assurance Manager
D. Goldsmith, Radiation Protection Manager
L. Guthrie, Maintenance Director
F. Lentine, Design Engineering Manager
R. Linthicum, Engineering Programs - PRA
G. O'Donnell, Fire Protection Engineer
A. Ronstadt, Site Maintenance Rule Coordinator
K. Schwartz, Plant Manager
J. von Suskil, Site Vice President

Nuclear Regulatory Commission

M. Chawla, Project Manager, NRR
A. Stone, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-456/457/01-11-01	NCV	failure to follow procedure (Section 1R20)
50-456/457/01-11-02	VIO	failure to maintain an adequate test control program (Section 1R22)
50-456/457/01-11-03	NCV	failure to follow TS 5.7.2(d) (Section 2OS1.2)
50-456/01-11-04	NCV	failure to follow procedure (Section 4OA7)

Closed

50-456/457/01-11-01	NCV	failure to failure to follow procedure (Section 1R20)
50-456/457/01-11-03	NCV	failure to follow TS 5.7.2(d) (Section 2OS1.2)
50-456/01-11-04	NCV	failure to follow procedure (Section 4OA7)
50-456/457/00-06-04	URI	Alternative shutdown capability was not independent of Fire Zone 11.5-0 and did not ensure integrity of the primary coolant boundary (Section 4OA5.2)
50-456/457/00-06-05	URI	Alternative shutdown capability was not independent of Fire Zone 11.6-0 (Section 4OA5.2)
50-456/457/01-09-01	URI	Instrument uncertainty was not applied to TS acceptance criteria (Section 4OA5.3)

Discussed

50-456/457/00-06-02	URI	License requirements reduced for two auxiliary building fire zones (Section 4OA5.1)

LIST OF ACRONYMS AND INITIALISMS USED

A1R09	Unit 1 2001 Refueling Outage
ADAMS	Agencywide Documents Access and Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BwAP	Braidwood Administrative Procedure
BwAR	Braidwood Annunciator Response Procedure
BwEP	Braidwood Emergency Procedure
BwGP	Braidwood General Procedure
BwMP	Braidwood Maintenance Procedure
BwOA	Braidwood Abnormal Operating Procedure
BwOP	Braidwood Operating Procedure
BwOSR	Braidwood Operability Surveillance Requirement
BwVS	Braidwood Engineering Surveillance
CC	Component Cooling Water
CFR	Code of Federal Regulations
CR	Condition Report
DG	Diesel Generator
dpm	disintegrations per minute
DRP	Division of Reactor Projects
ESF	Engineered Safety Features
FME	Foreign Material Exclusion
gpm	Gallons per Minute
HRA	High Radiation Area
LHRA	Locked High Radiation Area
LCO	Limiting Condition for Operation
MS	Main Steam
MSIV	Main Steam Isolation Valve
NCV	Non-Cited Violation
NES	Nuclear Engineering Standards
NOA	Nuclear Oversight Assessment
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulations
OOS	Out-of-Service
PARS	Publicly Available Records
PC	Primary Containment
PIF	Problem Identification Form
PRA	Probabilistic Risk Assessment
RCFC	Reactor Containment Fan Coolers
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWP	Radiation Work Permit
SDP	Significant Determination Process
SI	Safety Injection
SSC	Structures, Systems, and Components
SX	Essential Service Water
TPC	Temporary Procedure Change

UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VA	Auxiliary Building Ventilation System
VD	Ventilation - Diesel
VIO	Violation
WR	Work Request

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

WO 99230485	U-0 Freezing Temp Equip Protection Annual	September 28, 2001

1R04 Equipment Alignment

CR A2000-04707	Number of Out-of-Service Errors is Increasing (PI&R)	December 28, 2000
CR A2001-00813	Improper OOS for 1FSV-SX178 (PI&R)	March 19, 2001

1R11 Licensed Operator Requalification Program

1BwOP-0	Reactor Trip or Safety Injection Unit 1	Revision 1A
1BwEP-3	Steam Generator Tube Rupture Unit 1	Revision 100
EP-AA-111	Emergency Classification and Protective Action Recommendations	Revision 1
	Radiation Emergency Plan Annex For Braidwood Station	Revision 7

1R12 Maintenance Rule Implementation

	Maintenance Rule - Evaluation History: CD/CB	May 10, 2001
	Maintenance Rule - Evaluation History: RC	April 10, 2001
	Maintenance Rule (a)(I) Action Items	No Date
	Maintenance Rule - Performance Monitoring (Reliability Graph) User Parameters - RY	October 15, 2001
	Maintenance Rule - Performance Monitoring (Availability Graph) User Parameters - RY	October 15, 2001
	Maintenance Rule Expert Pane Scoping Determination - RY	October 15, 2001
	High Safety Significant Status of In-Scope Function (User Parameters)	October 15, 2001
	Maintenance Rule - Performance Criteria (User Parameters) - RY	October 15, 2001
	Maintenance Rule - Evaluation History (User Parameters) - RY	October 15, 2001

	Expert Panel Meeting Notes for October 9, 2000 and August 20, 2001	October 29, 2001
CR A2001-00332	Maintenance Rule Peer Group Containment Closure Industry Event Review (PI&R)	November 15, 2001
CR A2001-00607	2B D/G Fails Surveillance Prerequisites (PI&R)	November 14, 2001
CR A2001-00512	2FW540 Oscillations in Auto	February 18, 2001
CR A2001-00802	2A FW Pump Trip During Post Maintenance Run	March 19, 2001
CR A2001-00852	Trip of the 1FW01PC-B While Performing 1BwOP FW-10	March 21, 2001
CR A2001-00854	1FW01PC-B Tripped After Running for Approx. 50 Minutes	March 22, 2001
CR A2001-00962	2A FW Pump Trip 1 Minute After Start	March 30, 2001
CR A2001-00967	2A FW Pump Main Oil Pump Failure	March 31, 2001
CR A2001-01436	FW Pump Steam Leaks	May 14, 2001
CR A2001-01738	1FW009D Hydraulic Pump Cycling	June 10, 2001
CR 00073470	MR Criteria Exceeded and Not Identified in a Timely Manner (PI&R)	November 14, 2001
CR 00079964	Pipe Supports Found with Painted/Broken Grease Fittings	December 4, 2001
CR 00080661	Maint Rule Time Extended on Startup FW Pump	October 26, 2001
	Maintenance Rule Periodic Assessment #2 January 1998 - December 1999	February 28, 2000
	Maintenance Rule Periodic Assessment #3 January 2000 - April 2001	October 30, 2001
	List of Functional Failures (October 1999 - October 2001)	October 22, 2001
	List of Systems Returned to (a)(2) During the Assessment Period (01/01/2000 -04/30/2001)	November 2001
	Braidwood Station: Maintenance Rule Systems in (a)(1) Status: Goals/Monitoring	November 2001
	SSCs Removed/Added to the Maintenance Rule Program	November 2001

	Performance Criteria Changes During the Assessment Period	November 2001
	Maintenance Rule Reviews for the Auxiliary Feedwater, Diesel Generator (DG), and Residual Heat Removal Systems	2000 - 2001
	Focus Area Self-Assessment - Braidwood Station Maintenance Rule Implementation	April 28, 2000
NOA-20-ES05	Braidwood Station Assessment Report Nuclear Oversight Assessment - Maintenance Rule	March 27, 2000
BB PRA-017.03	Braidwood PRA Application Notebook: Maintenance Rule Performance Criteria	Revision 0
	Unavailability Data Used During the (a)(3) Periodic Assessment	April 2001
	Systems Classified Maintenance Rule (a)(1) During the Assessment Period (01/01/2000-04/30/2001)	April 2001
	Expert Panel Meeting Minutes	January 14, 2000
	Expert Panel Meeting Minutes	February 15, 2000
	Expert Panel Meeting Minutes	July 10, 2000
	Expert Panel Meeting Minutes	July 24, 2000
	Expert Panel Meeting Minutes	September 18,2000
	List of Performance Criteria Changes Made During the Assessment Period	November 2001
	Maintenance Rule - Performance Criteria	November 1, 2001
PIF [problem idenfication form] A1999-00351	Reliance Motors Need to be Reviewed for Applicability to Part 21	February 8, 1999
PIF A1999-03947	2B Diesel Generator Room Overcooled Due to Damper Hydramotor Failure	December 20, 1999
PIF A1999-03031	1VD01YA Damper Failed Full Open	October 12, 1999
PIF A1999-04041`	Availability and Reliability Criteria Has Been Exceeded for Function MS2, MSIVs	December 27, 1999

CR A2000-01254	Maintenance Rule Unavailability Not Being Adequately Captured in the Maintenance Rule Database	March 19, 2000
CR A2000-01935	Maintenance Rule Criterion PC4 Did Not Return to (a)(2) After A1R08	April 18, 2000
CR A2001-00201	Possible Revision Needed to Performance Criteria for the VD System	January 23, 2001
CR A2001-00695	2B Diesel Generator Room Low Temperature	March 7, 2001
CR A2001-01179	1A DG Vent Damper Failed Open	April 21, 2001
CR A2001-01668	Potential Rework: 2TZ-VD002AA Hydramotor Does Not Stroke in Closed Position	July 5, 2001
ER-AA-310	Maintenance Rule	Revision 0
NES-G-15.01	Maintenance Rule: Scoping Standard	Revision 0
NES-G-15.02	Maintenance Rule: Risk Significance Determination Standard	Revision 0
NES-G-15.03	Maintenance Rule: Performance Criteria Determination Standard	Revision 0
NES-G-15.04	Maintenance Rule: System Monitoring Standard	Revision 1
NES-G-15.05	Maintenance Rule: Goal Setting Standard	Revision 0
NES-G-15.06	Maintenance Rule: Periodic Assessment Standard	Revision 0

1R13 Maintenance Risk Assessments And Emergency Work Control

AR 00079250	Paint/Thinner Unattended in Safety-Related Area (NRC Identified)	October 17, 2001
WO 98123889 09	Whisping Past Seat - Very, Very Minor Contingency - Replace Valve Body	October 17, 2001
WO 99236830 01	Slight Packing Leak - < 1 DPM - Adjust Packing	October 15, 2001
OU-AA-103	Shutdown Safety Management Program - Attachment 1	November 2, 2001
CR 00076213	U1 Outage Activity Risk Impact on U2 Not Identified (PI&R)	November 15, 2001
CR A2001-01000	No Basis for Operator Response Times Used in the On-Line Risk Management Program (PI&R)	November 15, 2001

1R15 Operability Evaluations

WO 99159753	Repair Valve 1CV8392A	September 27 2001
AR00076349	1RY085A/B and 1RY086A/B Failed During Performance of 1BwOSR 3.4.11.3 (PI&R)	November 15, 2001
AR 00077031	NOS Identified Weld Deficiency 1CV8392A (Atlantic Group)	September 30, 2001
AR 00078486	Reactor Head Vent Hose	October 10 2001
AR 00079191	Potential Non-Conservatism in the SF Critical Analysis	October 17, 2001
CR 00076349	During Performance of 1BwOSR 3.4.11.3 Check Valves Failed (PI&R)	September 24, 2001
CR A2001-02001	Extent of Condition Review of Safety-Related Motors Based on 1D CD/CB Pump Failure Analysis (PI&R)	November 16, 2001

1R16 Operator Workarounds

AR 00078651	Operator Workaround Procedure Has Confusing Examples (NRC Identified)	October 11, 2001
OP-AA-102-103	Operator Work Around and Operator Challenges	Revision 0
BwOP RH-5	Residual Heat System Startup for Recirculation	Revision 12
BwOP RH6	Placing Residual Heat System in Shutdown Cooling	Revision 23
BwGP 100-4	Power Reduction	Revision 17
BwOA PRI-6	Component Cooling Water System Malfunction	Revision 100
BwOP CC-1	Component Cooling Water System Startup	Revision 8E3
	Operator Workaround Meeting Minutes	April 5 and August 22, 2001
A2001-02208	Operator Workaround: Unit 1-Feedwater Pump Speed Controller in Manual	July 29, 2001
A2001-02099	Possible Failure of Startup Feedwater Pump Oil Pressure Regulator	July 18, 2001
A2001-01796	OP AID 01-011 Out of Range for Current MWD/MTU	June 16, 2001
OWA 197	Heater Relief Valves Lift and Fail During Reactor Trip on Unit 1	

OC 5	Boron Dilution Prevention System is not Functional on Unit 1
OC 191	HD Pump Casing Pumpdown Affecting Unit 1
OC 193	1RH610/611 Controller Switch
OC 194	1 FW016 Positioner and I/P Enhancements on Unit 1
OC 195	Unit 1 MPT Disconnect (0C Phase) Has Jumper Installed
OC 198	Unit 2 Boron Dilution Prevention System is not Functional
OC 199	2 FW016 Positioner and I/P Enhancements
OC 200	1 HD046A Binding Problems
OC 201	Unit 1 RM-11 Requires Excessive Attention
OC 202	Unit 2 RM-11 Requires Excessive Attention
OC 203	2 RH610/611 Control Switch
OC 204	CW Blowdown Valves Require Manual Throttling When Starting System Flow

1R19 Post Maintenance Testing

AR 00082252	Unexpected Increase in 2B RH Suction Pressure	November 7, 2001
AR 00082563	2RH01PB - FME Issue Shavings in Pump Housing	November 7, 2001
AR 00082594	Metal Shaving Found Inside of Pump Bowl - FME Issue	November 12, 2001
AR 00082631	2B RH Pump Missing Anti-Rotation Pin on Diffuser Ring	November 12, 2001
AR 00082711	Disposition of 1B RH Pump Running Clearances	November 12, 2001
WO 99146116	Test Main Steam Safety Valve After Completion	October 5, 2001
BwMP 3305-107	Main Steam Safety Valves Lift Point Verification Using the Furmanite Trevitest system	Revision 7
WO 00373399	ASME Surv Requirements for RHR Pump	November 14, 2001
WO 99161427	Equip Response Time Test of Auxiliary Feedwater Pumps	October 2, 2001

WO 99242253	Replace Fuel Injectors 1B Aux Feed Diesel Pump	September 24, 2001
WO 99283704	Engineered Safety Features Response Time Compilation	October 6, 2001
WO 99284382	NIS Power Range Flux Rate Trip Elimination	October 18, 2001
CR A2001-00570	WRs Not Closed Out in a Timely Manner After Work was Completed (PI&R)	February 23, 2001
CR A2001-00583	MOV Diagnostic Test Scheduled Without Required As-found LLRT Scheduled (PI&R)	February 26, 2001
CR A2001-00614	Rework - Incorrect Orientation for 2C CD/CB Oil Cooler Reversing Heads (PI&R)	January 20, 2001
CR A2001-00802	2A FW Pump Trip During Post Maintenance Run (PI&R)	March 17, 2001
CR A2001-00922	Rework - 0AB04PA Pump Seized During Startup Retest Following Maintenance (PI&R)	March 26, 2001
SPP 01-003	Braidwood Unit 1 Power Uprate Project Full Power Ascension Procedure	August 9, 2001
ReMa Form	Initial Power Ascension Following A1R09	October 4, 2001
ESBU-TB-96-03-R0	RHR Pump Operating Recommendations	June 20, 1996

1R20 Refueling and Outage Activities

BwAP 370-3	Administrative Control During Refueling	Revision 26
BwAP 370-3A12	Fuel Handling Guidance for Fuel Movement from the Reactor Core to the Spent Fuel Pool	Revision 3
BwAP 370-3A13	Fuel Handling Guidance for Fuel Movement from the Spent Fuel Pool to the Reactor Core	Revision 4
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
BwAP 2364-3T2	Nuclear Component Transfer List Package - A1R09 Component Shuffles	September 24, 2001
BwAP 2364-3T2	Nuclear Component Transfer List Package - U1C10 Core Onload	September 24, 2001

BwAP 2364-3T2	Nuclear Component Transfer List Package - U1C9 Core Offload	September 24, 2001
BwCB-1Figure 28	Reactor Coolant System Cooldown Limitations	Revision 4
1BwGP 100-5	Plant Shutdown and Cooldown	Revision 24
1BwGP 100-5A1	Admin Out-of-Services for 1BwGP 100-5	Revision 3E1
1BwGP 100-6	Refueling Outage	Revision 15
1BwGP 100-6A1	Admin Clearance Orders for 1BwGP 100-6	Revision 2
BwMP 3100-092	Installation and Removal of Temporary Containment Penetration Covers	Revision 1
0BwOA Refuel-3	Loss of Spent Fuel Pit Cooling Unit 0	Revision 0
BwOP VA-E2	Electrical Lineup - Unit 0	Revision 3
BwOP AP-60T1	Bus 142 Outage Checkoffs	Revision 2
BwOP AP-60	Bus 142 Outage While in Mode 6 or Defueled	Revision 3
BwOP DC-7-111	125V DC ESF Bus 111 Cross-Tie/Restoration	Revision 3E2
BwOP DC-7-112	125V DC ESF Bus 112 Cross-Tie/Restoration	Revision 3E2
BwOP FC - E1	Electrical Lineup - Unit 1	Revision 1
BwOP FC-M1	Operating Mechanical Lineup Unit 1	Revision 6
BwOP FC-1	Fuel Pool Cooling System Start-Up	Revision 13
BwOP FC-15	Start-Up and Shutdown of the U1 Fuel Pool Cooling Purification Loop on the U1 Fuel Pool Cooling System	Revision 3
BwOP RC-4	Reactor Coolant System Drain	Revision 22
BwOP RH-6	Placing the RH System in Shutdown Cooling	Revision 23
BwOP RH-8	Filling the Reactor Cavity for Refueling	Revision 13
BwOP RH-11	Securing the RH System from Shutdown Cooling	Revision 16E1
BwOP RH-13	Lowering Reactor Cavity Level While Defueled	Revision 2
BwOP RH-14	Filling the Reactor Cavity While Defueled	Revision 3
1BwOSR 3.6.3.3	Primary Containment Integrity Verification of Isolation Devices Outside Containment	Revision 2
1BwOSR 3.6.3.4	Primary Containment Integrity Verification of Isolation Devices Inside Containment	Revision 1

1BwOSR 3.8.4.5-1	125V DC Bus 111 Crosstie to DC Bus 211 Non-Routine Surveillance	Revision 1
BwVSR 3.5.2.8	Visual Surveillance of Containment Recirculation Sumps	Revision 2
OOS 00002430	Accum/SI Test Line Isolation Valve - Unit 1	No Date
OOS 99031108	Bus 142 Bus Outage	No Date
OOS 99027688	Pump, 1B RHR Assembly	No Date
CR A2000-00268	Potential Trend - Increased Frequency of Human Performance Errors in Fuel Handling (PI&R)	January 18, 2000
CR A2000-00834	Poor Outage Planning for Known Problem Causes Unplanned Emergent/Non-Outage Work (PI&R)	February 24, 2000
CR A2000-01203	Inadvertent Containment Isolation Signal (PI&R)	March 17, 2000
CR A2000-01675	Safety Issues Raised by the Craft During A1R08 (PI&R)	March 26, 2000
CR A2000-03234	Poor Planning on 2B SI Train Relief Valve Work Causes Extra Work and Dose for Operations (PI&R)	August 14, 2000
CR A2000-03971	Material Not Pulled Prior to the Start of A2R08 (PI&R)	October 22, 2000
CR A2000-04329	A2R08 Outage Reactivity Management (PI&R)	November 9, 2000
CR A2001-00006	Engineering Self Assessment Identifies NRC Commitment Not Met	January 2, 2001
WO 98097709	LLRT 1PS228A/229A P-45 1A H2 MON SUP 1R20 Refuel Outage	37151
WO 99163960	Visual Inspection of Containment Sumps	October 2, 2001
WO 99168752	Provide Temporary Power for Fuel Cool Spent Fuel Pit Pump	September 25, 2001
AR 00076572	Hydro Lacing SX Cooling in 1B AF Water Room, Water Spray (NRC Identified)	September 26, 2001
ITR 01-058	Braidwood Independent Technical Review Report Revise Technical Requirements Manual TLCO 3.9.a, “Decay Time” and TS Bases B3.9.4, “Containment Penetrations,” and B3.9.7, “Refueling Cavity Water Level”	September 11, 2001

MA-AA-AD-6-03008	Foreign Material Exclusion	Revision 0
NF-AA-440	Fuel Conditioning	Revision 2
OP-AA-10	Equipment Clearance Process Description	Revision 0
OP-AA-101-201	Station Equipment Clearance and Tagging	Revision 4
OP-AA-108-108	Unit Restart Review	Revision 0
ReMa Form Attach 1	Coast Down Guidance	August 24, 2001
ReMa Form Attach 1	Shutdown Unit 1 for A1R09 on 9/21/01	September 10, 2001
ReMa Form Attach 1	Initial Power Ascension Following A1R09	October 4, 2001
	Control Room Log - Unit 1	September 22, 2001
	A1R09 Containment Closure Contingency Plan for Spare Penetration Restoration (October 2, 2001)	September 14, 2001
99027687	Pump, 1A Residual Heat Removal Assembly	No Date
99028998	RX Vessel Flange Leak Detect Main Isolation Valve	No Date
99029156	Auxiliary Feedwater Pump 1B SX Suction Valve Assembly	No Date
99029194	Pump, 1A Centrifugal charging Assembly	No Date
99031147	Charging to RC Loop 1B Isolation Valve (C/S @ 1PM05J) Assembly	September 26, 2001

<u>1R22</u> <u>Surveillance Testing</u>

1BwOSR 3.7.2.1	Main Steam Isolation Valve Full Stroke Quarterly Surveillance	Revision 3
2BwOSR 3.7.2.1	Main Steam Isolation Valve Full Stroke Quarterly Surveillance	Revision 2
WO 00342025	ASME Surveillance Requirements for 2B SAF Injection Pump	October 10, 2001
WO 00367688 01	Unit 1 Train B Solid State Protection System, Reactor Trip Breaker and Reactor Trip Bypass Breaker Bi-Monthly Surveillance	October 29, 2001

WO 99165136 01	1BwVSR 3.4.1.4, RX Coolant System Flow Measurement	October 12, 2001
NES-MS-08.1	Inservice Testing Bases Document Format and Content	Revision 3
IST-BWD-BDOC-V-25	Braidwood IST Program Bases Document, Volume 25 of 27	June 27, 2000
WO 00369151-01	1PSL-AF055 Functional Check of 1B AF Pump Suction Pressure Switch	November 1, 2001
CR A2000-04703	1B DG JW & LO Temp Outside of Acceptance Range for SR 3.8.1.7 (PI&R)	November 14, 2001
CR A2001-01782	U1 Component Cooling High Temperatures/Low RCP Flow During the Performance of SPP 01-005 (PI&R)	November 14, 2001
	Safety Evaluation to Operating License No. NPF-72 and NPF-77	Amendment 107
Regulatory Guide 1.27	Ultimate Heat Sink for Nuclear Power Plants	Revision 2
237/249/94016 (DRS)	Dresden Nuclear Power Station - Units 2 and 3 Inspection Report	September 16, 1994
NES-EIC-20.04	Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy	Revision 3
BwAR 1-2-B2	SX Pump DSCH HDR Temp High Low	Revision 8
1BwOSR 0.1-1,2,3	U1 - Modes 1, 2, and 3 Shiftly and Daily Operating Surveillance Data Package Cover Sheet	July 11, 2000
	Braidwood Design Engineering “White Paper” on Instrument Uncertainty	No Date
50-382/00-01	Waterford Steam Electric Station, Unit 3 Inspection Report	March 30, 2000
TAC Nos. MA3776 and MA3777	Task Interface Agreement Response: Accounting For Instrument Uncertainties In Surveillance Testing Acceptance Criteria	November 13, 1998
TAC No. M95177	Task Interface Agreement Evaluation Regarding Instrument Accuracy Affecting Millstone Unit 2	July 22, 1996

1R23 Temporary Plant Modifications

CR 00077799	Possible Unapproved Temporary Modification (NRC identified)	October 4, 2001
CC-AA-112	Temporary Configuration Changes	Revision 4

2OS1 Access Control to Radiologically Significant Areas

BwAP 370-3	Administrative Control During Refueling	Revision 26
BwFP FH-5	Fuel Movement in Containment	Revision 7
BwOP RC-10	Draining an Isolated Reactor Coolant Loop	Revision 19
BwRP 5010-1	Radiological Posting and Labeling Requirements	Revision 14
BwRP 6210-9	Requirements for Maintaining Emergency Hatch Access During Outage Conditions	Revision 3
RP-AA-403	Administration of the Radiation Work Permit Program	Revision 0
RP-AA-462	Controls for Radiography Activities	Revision 1
RP-AA-460	Controls for High and Very High Radiation Areas	Revision 1
RWP 5928	Reactor Coolant Pump Work	Revision 0
RWP 5929	Remove and Reinstall Reactor Head and Upper Internals	Revision 0
RWP 5942	Radiography	Revision 0
RWP10000367	Seal Table Work	Revision 0
RWP 5941	Fuel Moves During A1RO9	Revision 1
RWP 5921	Reactor Head Work	Revision 1
AR00075153	Perceived Procedure Violation (Security) Sy-AA-101-123	September 13, 2001
AR00075167	Perceived Violation of Security Procedure SY-AA-101-123	September 13, 2001
AR00076090	High Radiation Area Control Violation	September 22, 2001
AR00076116	Inadequate Doffing of PCs	September 22, 2001
AR00076117	Contaminated Area Boundaries Not Controlled	September 22, 2001
AR00076126	PCE-100kdpm Hot Particle From Sandbox Cover Work	September 23, 2001

AR00076138	Unit 1 Containment Airborne Contamination	September 23, 2001
AR00076246	1 Gallon RCS Spill During and Isolate RCS Loop Drain	September 24, 2001
AR00076335	Rad Practices - Personnel Exiting CNMT, Not Proceeding to PM8's	September 25, 2001
AR00076396	PCE-5 million DPM on Both Shoes of NRC Inspector	September 25, 2001
AR00077476	High Level Trash Bag Improperly Controlled in HRA	October 2, 2001
AR00076490	Personnel Climbing Above Six Feet Without Permission of RP	September 26, 2001

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

ALARA Plan 5908	Snubber Inspection and Testing	Revision 1
ALARA In Progress Review for 5908	Snubber Inspection and Testing	Revision 1
ALARA Plan 5921	Reactor Head Work	Revision 0
ALARA Plan 5918	Cavity Decon	Revision 0
ALARA Plan Amendment for 5918	Cavity Decon	Revision 1
ALARA Plan 5928	RCP Pump Seal and Motor Work	Revision 0
ALARA Plan Amendment for 5928	RCP Pump Seal and Motor	Revision 0
ALARA Plan 5935	Scaffold Build and Tear Down	Revision 0
ALARA Plan 5941	Fuel Moves During A1RO9	Revision 0

ALARA In Progress Review for 5941	Fuel Moves During A1RO9	Revision 1
ALARA Plan 5943	Install and Remove Temporary Shielding	Revision 0
ALARA In Progress Review for 5943	Install and Remove Temporary Shielding	Revision 1
	Braidwood A1RO9 Dose Performance	October 1, 2, 3 and 4, 2001
AR00073701	Exceeded Dose for Scheduled Work Orders	August 28, 2001
RP-AA-270	Prenatal and Post Natal Programs	Revision 1
RP-AA-400	ALARA Program	Revision 1
RP-AA-401	Operational ALARA Planning and Controls	Revision 1
	Station ALARA Committee Meeting	September 25, 2001
Focus Area Self-Assessment Plan #2001-026	Outage Readiness and Preparation	September 14, 2001
	Radiation Protection Self-Assessment Report July 2001	August 23, 2001
	Radiation Protection Self-Assessment Report 2nd Quarter 2001	July 31, 2001
	Braidwood 5 Year Exposure Reduction Plan	

2PS2 Radioactive Material Processing and Transportation

Shipping Documents	LSA II, Class B Waste (Dewatered Mixed Resin)	May 16, 2001

4OA1 Performance Indicator Verification

	Braidwood 1 - 2Q/2001 Performance Summary	October 22, 2001
	Braidwood 2 - 2Q/2001 Performance Summary	October 22, 2001
	Safety System Unavailability (HPSI/HPCI, RHR, AFW/RCIC, EDG)	No Date

40A5 Other

AR 00079520	NRC Notice of Apparent Violation Due to Lack of Sprinklers	October 18, 2001
BwAP 1110-1	Fire Protection Program System Requirements	Revision 0, 14, 15
BwAR 0-37-1A	Alarm No. 0-37-A4, Unit 1 Area Fire	Revision 8
BwAR 0-39-A4	Alarm No. 0-39-A4, Unit 2 Area Fire	Revision 8
1BwOA PRI-5	Control Room Inaccessibility, Unit 1	Revision 57C
Dwg. 20E-0-3663	Cable Pans Routing Auxiliary Building Plan El. 401'-0"	Revision AU
Dwg. 20E-0-3667	Cable Pans Routing Auxiliary Building Plan El. 426'-0"	Revision BB
Dwg. 20E-0-3664	Cable Pans Routing Auxiliary Building Plan El. 401'-0"	Revision AK
Dwg. 20E-0-3668	Cable Pans Routing Auxiliary Building Plan El. 426'-0"	Revision AD
Dwg. M61, Sht 1A	Diagram of Safety Injection Unit 1	
	Byron Station Units 1 and 2 Application for Amendment to Facility Operating License, NPF 37, Appendix A, TS	August 29, 1986
	Braidwood Station Units 1 and 2 Fire Protection	January 30, 1987