July 30, 2002

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

SUBJECT: DRESDEN NUCLEAR POWER STATION
NRC INSPECTION REPORT 50-237/02-08(DRP); 50-249/02-08(DRP)

Dear Mr. Skolds:

On June 30, 2002, the NRC completed an inspection at your Dresden Nuclear Power Station, Units 2 and 3. The enclosed report presents the inspection findings which were discussed with Mr. D. Bost and other members of your staff on June 25, 2002.

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. The inspection also included an inspection of the dry cask handling program.

Based on the results of this inspection, the inspectors identified three issues of very low safety significance (Green). Two of the issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as Non-Cited Violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these Non-Cited Violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspectors at the Dresden Nuclear Power Station.

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

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

Mark Ring, Chief
Branch 1
Division of Reactor Projects

Docket Nos. 50-237; 50-249; 72-00037
License Nos. DPR-19; DPR-25

Enclosure: Inspection Report 50-237/02-08(DRP);
50-249/02-08(DRP)

cc w/encl: Site Vice President - Dresden Nuclear Power Station
Dresden Nuclear Power Station Plant Manager
Regulatory Assurance Manager - Dresden
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional
Operating Group
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-237; 50-249; 72-00037
License Nos:	DPR-19; DPR-25
Report No:	50-237/02-08(DRP); 50-249/02-08(DRP)
Licensee:	Exelon Generation Company
Facility:	Dresden Nuclear Power Station, Units 2 and 3
Location:	6500 North Dresden Road Morris, IL 60450
Dates:	April 1, 2002 through June 30, 2002
Inspectors:	D. Smith, Senior Resident Inspector B. Dickson, Resident Inspector R. Lerch, Project Engineer P. Pelke, Reactor Engineer J. Gavula, Senior Reactor Inspector T. Ploski, Senior Emergency Preparedness Inspector T. J. Madeda, Physical Security Inspector W. Slawinski, Senior Radiation Specialist R. J. Leemon, Decommissioning Inspector, DNMS R. Zuffa, Illinois Department of Nuclear Safety C. Mathews, Illinois Department of Nuclear Safety D. Zemel, Illinois Department of Nuclear Safety
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000237-02-08(DRP), IR 05000249-02-08(DRP), Exelon Generation Company, on 6/30/2002. Dresden Nuclear Power Station, Units 2 and 3. Operability Evaluation, Surveillance, and Event Followup.

The inspection was conducted by resident and regional inspectors. The inspection identified three Green findings, of which two were considered Non-Cited Violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. 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.

A. Inspector Identified Findings

Cornerstone: Initiating Events

Green. A finding was identified involving deficient human performance during off-gas system testing, which resulted in operators manually initiating a scram of Unit 3 on May 4, 2000, due to degrading condenser vacuum conditions and increasing condensate inlet temperature.

This finding was more than minor because the event was potentially an initiating event. This event had minimal safety significance because the operator action of scramming the unit was consistent with plant procedures and pre-briefed in accordance with conservative decision making philosophy. (Section 4OA3.10)

Cornerstone: Mitigating Systems

- Green. A finding involving a Non-Cited Violation was identified for failure to comply with 10 CFR 50.62 due to the potential to lift standby liquid control system relief valves during an anticipated transient without scram on Unit 3.

  This finding was considered more than minor because the issue affected the function of a mitigating system. The risk significance of this issue was determined to be very low because the standby liquid control system could be recovered during an anticipated transient without scram event. Cycling of the relief valves would not prevent most of the borated solution from being injected into the reactor pressure vessel, and the licensee was able to demonstrate that the station remained within the acceptance criteria of their original anticipated transient without scram analyses during the relief valve lifts. (Section 1R15)

- Green. A finding involving a Non-Cited Violation was identified for failure to follow a surveillance procedure for calibrating isolation condenser differential pressure switches which resulted in a Group V isolation.

This finding was considered more than minor because the personnel error resulted in rendering the isolation condenser system inoperable. However, because the high pressure coolant injection system was operable, this finding was considered to be of very low safety significance. (Section 1R22)

B. Licensee Identified Findings

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 4OA7 of this report.

Report Details

Summary of Plant Status

Unit 2 began the inspection period at 912 MWe (95 percent thermal power and 100 percent of rated electrical capacity). On May 4, 2002, operators decreased power on Unit 2 from 912 to 500 MWe to perform control rod pattern adjustments. The unit returned to full power on May 5, 2002. On May 30, 2002, the 2A reactor feed pump developed a gasket leak and operations reduced power to 700 MWe. The unit returned to full power the following day.

Unit 3 began the inspection period at 822 MWe (100 percent thermal power). On May 18, 2002, operators decreased power to 700 MWe to swap reactor feed pumps. The unit returned to full power later that day. On May 25, operators decreased power to 550 MWe to perform deep/shallow rod swaps. The unit returned to full power on May 27, 2002. On May 30, 2002, a down power to 700 MWe was performed by operators to swap reactor feed pumps. The unit was returned to full power the same day. On June 15, 2002, operators reduced power to 495 MWe to perform deep/shallow rod swaps and swap reactor feed pumps. The unit was returned to full power operation later that night.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather (71111.01)

a. Inspection Scope

The inspectors assessed the licensee's implementation of the station's summer readiness process which included a review of tornado/severe wind and summer readiness procedures.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors selected a redundant or backup system to an out-of-service or degraded train, reviewed documents to determine correct system lineup, and verified critical portions of the system configuration. Instrumentation valve configurations and appropriate meter indications were also observed. The inspectors observed various support system parameters to determine the operational status. Control room switch positions for the systems were observed. Other conditions, such as adequacy of housekeeping, the absence of ignition sources, and proper labeling were also evaluated.

The inspectors performed equipment alignment walk-downs of the Unit 2 emergency diesel generator, Unit 2/3 emergency diesel generator, Unit 2 standby liquid control system, Unit 3 standby liquid control system, Unit 2 isolation condenser system, Unit 2/3 isolation condenser makeup pump fuel oil system, Unit 2/3 isolation condenser clean demineralization makeup system, and the 3A core spray pump system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured plant areas important to safety to assess the material condition, operating lineup, and operational effectiveness of the fire protection system and features. The review included control of transient combustibles and ignition sources, fire suppression systems, manual fire fighting equipment and capability, passive fire protection features, including fire doors, and compensatory measures. The following areas were walked down:

1) Unit 2 Reactor Building, Elevation 476'-6" West Low Pressure Coolant Injection Corner Room (Fire Zone 11.2.1)

2) Unit 3 Reactor Building, Elevation 476'-6" West Low Pressure Coolant Injection Corner Room (Fire Zone 11.1.1)

3) Unit 2 Reactor Building, Elevation 476'-6" East Low Pressure Coolant Injection Corner Room (Fire Zone 11.2.1)

4) Unit 3 Reactor Building, Elevation 476'-6" East Low Pressure Coolant Injection Corner Room (Fire Zone 11.1.2)

5) Unit 2 Reactor Building, Elevation 476'-6" High Pressure Coolant Injection Room (Fire Zone 11.2.3)

6) Unit 3 Reactor Building, Elevation 476'-6" High Pressure Coolant Injection Room (Fire Zone 11.1.3)

7) Unit 2 Reactor Building, Elevation 589' Stand-by Liquid Control Area (Fire Zone 1.1.2.5.D)

8) Unit 3 Reactor Building, Elevation 589' Stand-by Liquid Control Area (Fire Zone 1.1.1.5.D)

9) Isolation Condenser Pumphouse, Elevation 517', North Cubicle (Fire Zone 18.7.1)

10) Isolation Condenser Pumphouse, Elevation 517’, South Cubicle (Fire Zone 18.7.2)

11) Unit 2 Reactor Building, Elevation 613’ Refuel Floor (Fire Zone 1.1.2.6)

12) Unit 3 Reactor Building, Elevation 613’ Refuel Floor (Fire Zone 1.1.1.6)

13) Auxiliary Electric Equipment Room, (Fire Zone 6.2)

14) Unit 2 250Vdc Motor Control Centers 2A and 2B, Elevation 570’, (Fire Zone 1.1.2.4)

15) Unit 2 Isolation Condenser area, Elevation 589’, (Fire Zone 1.1.2.5.A)

16) Unit 3 Isolation Condenser area, Elevation 589’, (Fire Zone 1.1.1.5.A)

17) Unit 3 250Vdc Motor Control Centers 3A and 3B, Elevation 570’, (Fire Zone 1.1.4)

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed flooding mitigation and protection plans to ensure that areas highly susceptible to flooding were adequately addressed. The inspector reviewed procedures for coping with flooding in risk significant areas in the plant.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors assessed the licensee’s implementation of the maintenance rule by determining if systems were properly scoped within the maintenance rule. The inspectors also assessed the licensee’s characterization of failed structures, systems, and components, and determined whether goal setting and performance monitoring were adequate for the control room heating, ventilation, and air conditioning, dry well sampling, and intermediate range monitor systems.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated the effectiveness of the risk assessments performed before maintenance activities were conducted on structures, systems, and components and verified how the licensee managed the risk. The inspectors evaluated whether the licensee had taken the necessary steps to plan and control emergent work activities. The inspectors completed evaluations of maintenance activities on the Unit 3, Division I, low pressure coolant injection/containment cooling service water system, Unit 2 125Vdc battery charger, Unit 3 low pressure coolant injection/containment cooling service water system, Division II, Unit 3 battery charger, motor control center 29-2 emergency diesel generator vent fan power supply breaker, 2B standby liquid control pump and gearbox oil change out, and bus 34-1/bus24-1 tie breaker.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed operator logs, condition reports, and alarm printer outputs associated with one non-routine event. The event involved the control room heating, ventilation, and air conditioning (HVAC) system realignment from the normal mode of operation to the smoke purge mode due to a faulty smoke detection circuit.

On April 10, 2002, the control room HVAC system realigned from the normal mode of operation to the smoke purge mode due to a faulty smoke detection circuit. In the normal mode of operation, the control room HVAC provides 2,000 standard cubic feet per minute (SCFM) of outside air. Based on this flowrate, the operators were allotted 40 minutes to manually realign the system to its emergency lineup during a loss of coolant accident to meet 10 CFR Part 100 limits to the operators. However, with the unit in the smoke purge mode, approximately 22,000 SCFM of outside air would be provided. As a result, the operators would have significantly less time, which had not been previously analyzed, to place the unit in its emergency lineup (See Section 4OA3.6 of this report).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability evaluations to ensure that operability was properly justified and the component or system remained available, such that no unrecognized increase in risk occurred. The review included issues involving the operability of a crack in the scavenger oil pump strainer enclosure of the Unit 3 station blackout diesel, degraded Unit 2/3 control room emergency ventilation refrigeration cooling flow rate, degraded Unit 2 control rod drive hydraulic piping, 24 of the 96 tubes in the Unit 2 high pressure coolant injection room cooler having through wall leaks, various reactor protection system cable trays missing covers, and the inadvertent lifting of the Unit 3 standby liquid control system relief valve.

b. Findings

Lifting of Standby Liquid Control Relief Valves During Anticipated Transients Without Scram (ATWS)

One Green finding involving a Non-Cited Violation was identified for failure to comply with 10 CFR 50.62 due to the potential to lift the standby liquid control system relief valves during an anticipated transient without scram on Unit 3.

Background

The inspectors reviewed the operability evaluation performed for Condition Report (CR) 75599, “Potential to Lift Standby Liquid Control Pump Discharge Relief Valves During ATWS Transient,” to determine the impact that the premature lifting of relief valves had on system operability and compliance with 10 CFR 50.62.

The standby liquid control (SBLC) system was part of the original plant design and provided an independent and diverse method for shutting down the reactor if an insertion of the control rods did not occur. The SBLC system would shut down the reactor by pumping a neutron absorbing solution that would achieve and maintain subcriticality in the reactor vessel. Although the SBLC system contains two pumps, only one pump was needed to perform the initial design basis function. Quarterly in-service testing data showed that the actual expected flow from two SBLC pump was 83 gallons per minute (gpm).

In 1984, the NRC issued the ATWS rule (10 CFR 50.62). This rule implemented more stringent pump flow rate requirements for the SBLC pumps. Specifically, paragraph (c)(4) of 10 CFR 50.62 requires, in part, that each boiling water reactor must have a SBLC system with the capability of injecting into the reactor vessel a borated water solution at such a flow rate that the resulting reactivity control was at least equivalent to that resulting from the injection of 86 gpm of 13 weight percent sodium pentaborate decahydrate (boron) solution.

Compliance with the ATWS rule

To achieve compliance with the ATWS rule, licensee personnel used the methodology provided in General Electric Topical Report NEDE-31096-P-A to determine the required SBLC pump flow rate and boron concentration. The results of a calculation provided in the topical report showed that two pump operation was needed in order to provide 80 gpm of at least 14 weight percent boron solution to the reactor vessel. The pump flow rate and boron concentration were reviewed and approved by the NRC in Technical Specification safety evaluation reports dated on or before March 28, 1988. The licensee performed a calculation and determined that a SBLC system pump discharge pressure of 1355 pounds per square inch gauge (psig) was required to ensure that the boron solution was injected into the reactor vessel. Head losses were not included in this result. This calculation also assumed a reactor vessel dome pressure of 1135 psig which was consistent with General Electric's ATWS analyses NEDE-25026 and NEDE-24223 performed in the 1970s. Both NEDE documents assumed that reactor pressure had stabilized due to actuation of the safety relief valves at the time that the standby liquid control system was initiated. The NEDE documents also used simplified generic main steam relief and safety valve models rather than plant specific models.

During preparations for power uprate implementation, ATWS conditions were re-analyzed using the ODYN computer code approved by the NRC. The ODYN computer code used plant specific main steam relief and safety valve flow capacity and setpoint information. When the plant specific information was input into the ODYN code, the licensee determined that reactor vessel pressure could be as high as 1263 psig rather than the 1135 psig calculated in the original ATWS analyses. When the SBLC system head losses of 206 psig were added to the newly calculated reactor vessel pressure of 1263 psig, a SBLC pump discharge pressure of 1469 psig was achieved. This new pump discharge pressure was higher than the SBLC system relief valve setting (1500 psig ± 45 psig) and could have resulted in the relief valves lifting during system operation. The lifting of the relief valves would cause SBLC system flow to be recirculated to the system storage tank rather than injected into the reactor vessel. Due to the inability to provide continuous SBLC system design flow into the reactor vessel as stated by the ATWS rule, the licensee failed to comply with the rule.

Review of Technical Specification Operability

Technical Specification Surveillance Requirement 3.1.7.7 required the licensee to demonstrate that each SBLC pump was capable of pumping at a rate of at least 40 gpm with a discharge pressure of greater than or equal to 1275 psig. The inspectors reviewed additional information on the relief valves and determined that due to differences in system head losses during one and two pump system operation, the licensee could perform the testing specified in Technical Specification Surveillance Requirement 3.1.7.7 without lifting the relief valves because only one pump was tested at a time. Based upon the continued ability to satisfy Technical Specification Surveillance Requirement 3.1.7.7, the licensee determined the standby liquid control system remained operable even though the licensee was unable to continuously inject 80 gpm of sodium pentaborate solution as required to meet 10 CFR 50.62. This decision was based on the following:

- The original design basis for the standby liquid control system did not specify a required flow rate or sodium pentaborate decahydrate concentration.

- For Dresden Unit 3, the analysis credited a SBLC flowrate of 80 gpm up to a reactor pressure of 1235 psig. For reactor pressure greater than 1235, no SBLC flow was credited. The licensee stated that this was conservative because a flow of 80 gpm could be delivered to the reactor without lifting the relief valves up to a reactor pressure of 1264.7 psig (1455-190.3).

- Additionally, the licensee's calculations determined that the intermittent periods where the SBLC relief valve would open was on the order of 40 seconds total. SBLC was expected to inject intermittently after the relief valve closed. The overall impact of intermittent SBLC relief valve opening was only an increase of one degree Fahrenheit in peak suppression pool temperature.

- The requirements of 10 CFR 50.62 were beyond the design basis of the plant. The inspectors disagreed with this statement because after 10 CFR 50.62 was published, the requirement became a part of the plant design basis.

The similar issue existed at the Quad Cities and Susquehanna plants which also involved conflicting information regarding the relationship between Technical Specifications and 10 CFR 50.62. The Susquehanna issue was the subject of a Region I Task Interface Agreement which was completed by the Office of Nuclear Reactor Regulation. Based on the Office of Nuclear Reactor Regulation response to the Task Interface Agreement, the inspectors concluded that although the original Dresden design basis did not include ATWS events, these types of events must now be included as part of the design basis. The inspectors used the above information to determine that the potential to lift the standby liquid control relief valve resulted in the licensee being outside of their design basis and in noncompliance with the ATWS rule because the system would be unable to meet the required injection flow rate and boron concentration during the time the relief valves were lifting.

Part 50.62 to 10 CFR requires, in part, that each boiling water reactor must have a SBLC system with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate that the resulting reactivity control was at least equivalent to that resulting from the injection of 86 gpm of 13 weight percent sodium pentaborate decahydrate (boron) solution. The failure, since 1984, to have a standby liquid control system with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate that the resulting reactivity control was at least equivalent to that resulting from the injection of 86 gpm of 13 weight percent sodium pentaborate decahydrate (boron) solution was considered a **Non-Cited Violation (50-249/02-08-01)** of 10 CFR Part 50.62. This issue was entered into the licensee's corrective action program as CR# 75599.

The inspectors reviewed the risk significance of this issue and determined that the inability of the standby liquid control system to meet the requirements of the ATWS rule was more than minor because the issue affected the function of a mitigating system.

The inspectors screened the issue using the Significance Determination Process and determined the risk significance of this issue to be very low (Green) because the standby liquid control system could be recovered during an ATWS event. Cycling of the relief valves would not prevent most of the borated solution from being injected into the reactor pressure vessel, and the licensee was able to demonstrate that the station remained within the acceptance criteria of their original ATWS analyses even if no boron solution was injected into the reactor pressure vessel during the relief valve lifts.

The Office of Nuclear Reactor Regulation also determined that although the licensee was not in compliance with the ATWS rule, the standby liquid control system remained operable as required by Technical Specification 3.1.7. This determination was based upon information contained in NUREG-1433, "Standard Technical Specifications General Electric Plants," which states that Technical Specification 3.1.7 does not require meeting the requirements of 10 CFR 50.62 to meet the associated Technical Specification Limiting Condition for Operation. Based upon this information, no violation of Technical Specifications occurred. The licensee planned to modify the standby liquid control systems during the upcoming refueling outages to eliminate the lifting of the relief valves during two pump operation.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed post-maintenance test results to confirm that the tests were adequate for the scope of the maintenance completed and that the test data met the acceptance criteria. The inspectors also reviewed the tests to determine if the systems were restored to the operational readiness status consistent with the design and licensing basis documents. The inspectors reviewed post-maintenance testing activities associated with the Unit 3 relay replacement in the division I low pressure coolant injection system loop, control room emergency heating, ventilation, and air conditioning valve replacement, repair of the Unit 2 service water radiation monitor, maintenance of the 3B core spray pump, and limit switch replacement on the Unit 2 low pressure coolant injection system mini-flow valve.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and reviewed test results. The inspectors assessed whether the selected plant equipment could perform its intended safety function and satisfy the requirements contained in Technical Specifications. Following the completion of the test, the inspectors determined that the test equipment was removed and the equipment returned to a condition in which it could perform its intended safety function. The review included surveillance testing activities associated with the Unit 3 emergency diesel generator, Unit 3 containment cooling service water, Unit 2 standby liquid control, Unit 2 reactor vessel, local power range monitoring, average power range monitoring, isolation condenser, Unit 3 drywell radiation monitor, Unit 2 reactor building ventilation, Unit 2 reactor vessel high pressure scram pressure switch, and reactor recirculation systems.

b. Findings

One Green finding involving a Non-Cited Violation was identified for failure to follow a surveillance procedure resulting in a Group V isolation.

On June 6, 2002, instrument mechanics [IMs] were performing DIS 1300-02, “Isolation Condenser Steam/Condensate Line High Flow Calibration,” Revision 22 on Unit 2. The IMs had just completed the calibration of one of four condensate/steam differential pressure switches. The IMs had completed procedural step I.14.g. The following step, I.14.h directed the IM to notify the nuclear station operator that previously installed test switches for bypassing the trip inputs from DPIS 2-1349-B and DPIS 2-1350-B, that would cause a Group V isolation, would be manipulated to prevent them from performing their intended function. As a result the isolation condenser would be inoperable for a short period of time due to all of its trip signals being unable to cause a Group V isolation. The IM skipped this step and proceeded to step I.14.I, which directed the IM to slowly open DPIS 2-1349-A low side instrument isolation valve.

This valve shared a common sensing line with DPIS 2-1349-B, and when the valve was opened a spike was experienced on DPIS 2-1349-B which caused a Group V isolation because its inputs had not been bypassed. The IM realized the error and reported to the Operations Shift Manager. The onshift crew entered the applicable technical specification for the isolation condenser being inoperable and verified that the high pressure coolant injection system was operable. Administrative procedure HU-AA-104-101, “Procedure Use and Adherence,” Revision 0, Step 4.3.2 requires that all numbered steps in Category 1 procedures be performed in sequence. DIS 1300-02 is a Category 1 procedure.

The IMs were directed to rebrief the procedure to restore the switches to normal and back out of the procedure. This would allow the operators to restore the isolation condenser system to a standby lineup. This event was determined not to be reportable because the isolation signal was not due to a valid plant condition requiring isolation of the isolation condenser.

Dresden Technical Specification 5.4.1, states that procedures shall be established, implemented, and maintained covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A of Regulatory Guide 1.33, states, in part, that surveillance and calibration tests are typical safety-related activities that should be covered by procedures. Procedural Step 4.3.2 of HU-AA-104-101, requires that all numbered steps in Category 1 procedures (DIS 1300-02 is a Category 1 procedure) be performed in sequence. Contrary to the above, on June 6, 2002, an instrument mechanic’s failure to perform procedural steps of DIS 1300-02 in sequence as required by HU-AA-104-101 was a violation of Technical Specification 5.4.1. This violation is being treated as a Non-Cited Violation, consistent with Section VI.A.1, of the NRC Enforcement Policy **(NCV 50-237/02-08-02(DRP))**. This issue is in the licensee's corrective action program as CR #00110970.

This finding was considered more than minor because the IM's error resulted in rendering the isolation condenser system inoperable. However, because the high pressure coolant injection system was operable, this finding was considered to be of very low safety significance.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs (71122.03)

.1 Reviews of Radiological Environmental Monitoring Reports and Data

a. Inspection Scope

The inspector reviewed the Annual Radiological Environmental Operating Reports for calendar years 2000 and 2001, and the results of monthly radiological environmental monitoring analyses for 2002 thru April 2002. The inspector also reviewed the results of the last two land use censuses, changes made to the Offsite Dose Calculation Manual (ODCM) relative to the radiological environmental monitoring program and the results of the vendor laboratory inter-laboratory comparison program for 2000 and 2001. These reviews were conducted to verify that the radiological environmental monitoring program (REMP) was implemented as required by Technical Specifications and the ODCM, and to verify that any changes to the program did not affect the licensee's ability to monitor the impacts of radioactive effluents on the environment. Additionally, the inspector evaluated the current locations of the environmental monitoring stations and the types of samples collected from each location to determine if they were consistent with the ODCM and NRC guidance in Regulatory Guides 1.21 (Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Cooled Nuclear Power Plants"), 4.8 ("Environmental Technical Specifications for Nuclear Power Plants") and an associated NRC Branch Technical Position.

b. Findings

No findings of significance were identified.

.2 Walkdowns of the Radiological Environmental Monitoring Stations and Meteorological Tower

a. Inspection Scope

The inspector walked-down all eight near and far field environmental air sample indicator monitoring stations to determine whether they were located as described in the ODCM, to assess equipment material condition and operability, and to verify that monitoring station orientation, vegetation growth control, and equipment configuration allowed for the collection of representative samples. The meteorological tower was also walked-down by the inspector to verify that the tower was sited adequately and that instrumentation was installed consistent with Regulatory Guide 1.23 ("Onsite Meteorological Programs"). Meteorological data readouts and recording instruments located at the tower and as provided by the plant process computer were viewed and verified to be operable and were compared to determine if there were any line loss differences.

b. Findings

No findings of significance were identified.

.3 Reviews of Radiological Environmental Monitoring Equipment Maintenance and Testing

a. Inspection Scope

The inspector reviewed the REMP contractor's pump maintenance procedure and environmental air sample pump and meteorological tower equipment calibration and maintenance records for 2001 thru May 2002, to verify that the testing and maintenance programs for this equipment were implemented consistent with procedural requirements and industry standards. Calibration records for 2001 thru April 2002 for those rotameters used by the REMP technician to field check air sample pump flow and the most recent calibration of the rotameter standard used to calibrate the field rotameters were reviewed to verify that instrument certifications met industry standards and had traceability to the National Institute of Standards and Technology. The inspector discussed air sample pump maintenance practices with the contractor REMP technician to assess the adequacy of the preventive maintenance program for this equipment and to evaluate the technician's knowledge of the program and procedures.

b. Findings

No findings of significance were identified.

.4 Reviews of REMP Sample Collection and Laboratory Analyses

a. Inspection Scope

The inspector accompanied the contractor REMP technician and observed the individual collect a Des Plaines River surface water sample and exchange air particulate filters at five environmental air sampling stations. The observations were made to determine whether samples were collected in accordance with the contractor's sampling procedure and to determine if appropriate practices were used to ensure sample integrity. Additionally, the inspector observed the technician complete pump air flow, vacuum and sampling train leak checks to verify that they were accomplished adequately, consistent with the vendor's procedure. The inspector assessed the analytical detection capabilities of the contract laboratory used by the licensee to analyze its environmental samples, and discussed with radiation protection management its plans to revise the ODCM to better reflect the current inter-laboratory comparison program. The assessment was conducted to determine if the radiological environmental sample analysis and inter-laboratory comparison programs were implemented consistent with the ODCM and industry standards, and to verify that the vendor was capable of performing adequate radiological measurements.

b. Findings

No findings of significance were identified.

.5 Unrestricted Release of Material From Radiologically Controlled Areas (RCAs)

a. Inspection Scope

The inspector evaluated the licensee's procedures and practices for the unrestricted release of material from RCAs, for the survey of personnel leaving the RCA and the site, and for responding to personnel contamination monitor alarms. Specifically, the inspector reviewed the licensee's personnel survey and unconditional release program to verify that: (1) radiation monitoring instrumentation used to perform surveys of personnel and for unrestricted release of materials and equipment were appropriate; (2) instrument sensitivities were consistent with NRC guidance contained in Inspection and Enforcement Circular 81-07 ("Control of Radioactively Contaminated Material") and Health Physics Positions in NUREG/CR-5569 ("Health Physics Positions Database") for both surface contaminated material and material in volumetric form; (3) criteria for survey and unconditional release conformed to NRC requirements; and (4) licensee procedures were technically sound and provided appropriate guidance for survey techniques. The inspector reviewed the licensee's most recent 10 CFR Part 61 analyses and its assessment of the plant's radionuclide mix to determine if the potential impact of difficult to detect contaminants (such as those that decay by electron capture) was adequately evaluated and factored into the unrestricted release survey program.

b. Findings

No findings of significance were identified.

.6 Identification and Resolution of Problems

a. Inspection Scope

The inspector reviewed Nuclear Oversight field observations performed since 2001, and condition reports (CRs) generated in 2001 thru June 21, 2002 relative to the REMP and radioactive material control programs. In addition, the inspector reviewed the results of a REMP/radioactive material control program self-assessment completed in June 2002, including the corrective actions taken for the deficiencies identified during the 2000 self-assessment. These reviews were conducted to determine if the licensee adequately identified individual problems and trends, evaluated contributing causes and extent of condition, and developed corrective actions to prevent recurrence. Additionally, several radioactive material control incidents that involved the release of contaminated items outside the RCA and which occurred during the 12 months preceding the inspection were reviewed to assess their significance, causes and the adequacy of the licensee's corrective actions.

b. Findings

No findings of significance were identified.

3. REACTOR SAFETY

Cornerstone: Emergency Preparedness

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

a. Inspection Scope

The inspector reviewed Revisions 13 and 14 of the Dresden Station Annex to the Exelon Nuclear Standardized Radiological Emergency Plan to determine whether changes identified in these revisions reduced the effectiveness of the licensee's emergency planning, pending onsite inspection of the implementation of these changes.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed station personnel during a licensee only participation emergency preparedness drill exercise to determine the effectiveness of drill participants and the adequacy of the licensee's critique in identifying weaknesses and failures. The drill scenario involved the loss of 125Vdc bus 2A-2, loss of annunciator, failure of 2B recirculation motor generator set speed feedback signal, and an isolation condenser steam line break.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP4 Security Plan Changes (71130.04)

a. Inspection Scope

The inspector reviewed Revision 67 to the Dresden Nuclear Power Station Security Plan to verify that the changes did not decrease the effectiveness of the security plan. The referenced revision was submitted in accordance with regulatory requirements by licensee letter dated April 11, 2002.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator (PI) Verification (71151)

Occupational Exposure Control Effectiveness and Radiological Effluent Technical Specification (RETS)/ODCM Radiological Effluent Occurrence PIs

a. Inspection Scope

The inspector reviewed data associated with the Occupational Exposure Control Effectiveness PI and the RETS/ODCM PI to determine if these indicators were accurately assessed and reported since last reviewed in September 2001. Specifically, the inspector reviewed the licensee's CR database and selected CRs generated between October 2001 and June 21, 2002, to identify any potential occurrences that were not recognized by the licensee. For the occupational radiation safety PI, the inspector selectively reviewed electronic dosimetry dose alarm investigation reports and radiation exposure investigation logs and reports to determine if any potential unintended dose occurrences took place. For the public radiation safety PI, the inspector selectively reviewed gaseous and liquid effluent release data and the results of associated offsite dose calculations. The inspector also reviewed monthly PI verification records to assess compliance with station procedures LS-AA-2140 and 2150, "Monthly PI Data Elements for Occupational Exposure Control Effectiveness" and "Monthly PI Data Elements for RETS/ODCM Radiological Effluent Occurrences", respectively. Additionally, PI data collection and analyses were discussed with involved radiation protection staff to determine if the program and processes were implemented consistent with industry guidance in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline."

b. Findings

No findings of significance were identified.

4OA3 Event Follow-up (71153)

a. Inspection Scope

The inspectors reviewed licensee event reports (LERs) to ensure that issues documented in these reports were adequately addressed in the licensee's corrective action program. The inspectors also interviewed plant personnel and reviewed operating and maintenance procedures to ensure that generic issues were captured appropriately.

The inspectors reviewed operator logs, the Updated Final Safety Analysis Report, and other documents to verify the statements contained in the Licensee Event Reports. Also, the inspectors reviewed Unresolved Items to determine if the licensee was in violation of any regulatory requirement.

b. Findings

.1 (Open) Unresolved Item 50-249/01-21-01: Past Operability of the HPCI System With a Degraded Support was Indeterminate Because There was No Consideration for Transient Load That Had Damaged the Support

In response to this Unresolved Item, the licensee provided several calculations to demonstrate that the system would have been able to perform its safety function between July 5, 2001, when a pipe support was apparently damaged during an automatic initiation of the system, and September 30, 2001, when a significant amount of air was vented from the system's discharge piping and when the damaged pipe support was repaired. These calculations showed that although one additional support would have failed during an automatic initiation, the system would have still been operable.

The NRC inspectors reviewed the calculations, but could not conclude that the system would have been able to perform its safety function due to several relevant questions. These questions were provided to the licensee informally on May 8, 2002, and in a letter to Mr. J. Skolds of Exelon from Mr. J. Jacobson of Region III on June 26, 2002. Pending a review of the licensee's responses to these questions, and the impact on the current calculations, this item will remain open.

.2 (Closed) Unresolved Item 50-237/01-13-03: Inadequate Surveillance Acceptance Criteria

This Unresolved Item involved the performance of Dresden Operating Surveillance 2300-03, "High Pressure Coolant Injection Operability Verification," Revision 67, on May 25, 2001. During the surveillance the high pressure coolant injection system pump failed to achieve 1218-1280 psig discharge pressure at a flow rate of greater than 5000 gpm within 25 seconds as specified by the procedure. The licensee contended that only the flow requirements needed to be met within the 25 second time period. A licensee investigation determined that this was caused by a high pressure coolant injection system controller dead band issue. The operators were eventually able to achieve 5000 gpm at greater than 1218 psig.

During a subsequent review of this Unresolved Item by the Office of Nuclear Reactor Regulation, it was determined that no specific design basis accident assumptions required the high pressure coolant injection system to be able to inject into the reactor vessel within this specified time period. Therefore, this item is closed.

.3 (Closed) Unresolved Item 50-237/249/02-04-01: Corrective Action Program Change not Factored into Maintenance Rule Program

In August 2001 the station made a change to the corrective action program which allowed a work request to be generated instead of a condition report for low level equipment problems. The change did not take into account how the system engineer would evaluate these equipment problems for maintenance rule functional failure determination because the system engineers were not required to review work requests. Subsequently, the system engineers were tasked with reviewing 1,700 work requests to determine if any of the identified equipment deficiencies and failures resulted in maintenance rule functional failures.

The licensee identified a maintenance rule functional failure associated with a main steam line drain valve due to intermittent operation of the motor and the motor running hot. The valve is used in a Dresden emergency operating procedure for emergency depressurization. The procedure also had other equipment that could perform this same depressurization action. This was the only failure for the valve over a 24 month rolling average. The performance monitoring criteria are three failures over a 24 month period. The valve motor was replaced and the station could not determine the cause of the motor running hot. The licensee did not exceed any performance monitoring criteria due to this failure. A violation of regulatory requirements did not occur. This issue is documented in CRs #00098406 and #00103812 and this issue is closed.

.4 (Closed) 50-249/02-04-02 URI: Potential Violation of Technical Specification Limiting Condition for Operation 3.5.1 Due to Improper Alignment of High Pressure Coolant Injection One Non-Cited Violation was identified by the licensee involving the onshift operating crew's failure to have the high pressure coolant injection system in a operable condition prior to making reactor steam dome pressure reaching 150 psig. See Section 4OA7 of this report. This URI is closed.

.5 (Closed) LER 50-249/02-01-00: High Pressure Coolant Injection not in Standby Operation When Required by the Technical Specifications. One licensee identified violation is discussed in section 4OA7 of this report.

On March 23, 2002, during startup from a maintenance outage on Unit 3, the onshift crew determined that the high pressure coolant injection system (HPCI) was not required to be aligned in its operational standby readiness mode prior to reactor steam dome pressure reaching 150. As a result the crew maintained the HPCI steam support valves closed (3-2301-4 and 5). The onshift crew incorrectly believed that HPCI was inoperable, because maintenance had been performed on the system, and the system required the performance of a 24 month technical specification surveillance test.

Therefore, when steam dome pressure reached 150 psig, the on-shift crew entered technical specification limiting condition for operation action (LCO) statement 3.5.1.F for HPCI being inoperable which required immediate verification that the isolation condenser was operable and restoration of HPCI within 14 days. These actions were incorrect and resulted in the licensee violating Technical Specification 3.0.4.

The licensee's investigation revealed that a knowledge deficiency associated with high pressure coolant injection LCO applicability and the misinterpretation of a footnote for 24 month TS testing requirements on HPCI. Also the licensee identified that unit start-ups had occurred with the high pressure coolant injection turbine uncoupled from the pump during D2R14 (June 3, 1995 to February 26, 1996) and D3R14 (March 29, 1997 to June 3, 1997).

An initiation signal would have realigned the high pressure coolant injection steam supply valves to an open condition and high pressure coolant injection would have automatically started and performed its design function. Also, the core spray, low pressure coolant injection systems, isolation condenser, and automatic depressurization system were operable. However, during the two occasions when the HPCI turbine was uncoupled from the pump, HPCI could not perform its intended function. Therefore, the issue was evaluated in the phase 2 significant determination process and was determined to be of very low significance (Green) because all the other mitigating systems were available. This LER is closed.

.6 <u>(Closed) LER 50-237/02-01-00</u>: Unit 2 Isolation Condenser Time Delay Relay Surveillance Failures Due to Setpoint Tolerance Specified with No Margin

This issue involved the licensee installation and testing of new relays. Instrument mechanics failed to follow the appropriate procedure when calibrating the relays which resulted in a Green finding as documented in inspection report 50-237/50-249/02-03. This issue was entered into the licensee's corrective action program as CR #00091401. This LER is closed.

.7 <u>(Closed) LER 50-237/02-02-00</u>: Smoke Purge Mode Operation Prevents the Fulfillment of the Safety Function of the Control Room Emergency Ventilation System

On April 10, 2002, following smoke detector alarms, the nonsafety-related control room heating, ventilation, and air conditioning (HVAC) system automatically aligned to the smoke purge mode. This prevented the safety-related Control Room Emergency Zone Train "B" Ventilation System (CREVS) from fulfilling its safety function. In the purge mode, Train "A" provides approximately 20,000 scfm of outside air which renders the control room emergency air filtration unit incapable of mitigating the control room dose expected during an accident. The licensee determined that this design issue was not recognized during the development and safety evaluation of modification M12-0-82-001 for installation of CREVS in 1982. Temporary Modification No. 336451 was implemented on April 12, 2002, to disable the automatic smoke purge mode of Train "A" pending implementation of a permanent modification to allow only manual actuation. This issue had minimal safety significance because the probability of an automatic initiation of the smoke purge mode concurrent with an accident requiring the placement of CREVS in service is low. This issue was entered into the licensee's corrective action program as CRs #103270 and #103286. This LER is closed.

.8 (Closed) LER 50-237/01-01-00: Primary Containment Isolation System Valve Adjusted During Valve Operation Test and Evaluation System (VOTES) Test with No Local Leak Rate Test (LLRT) Performed

On December 19, 2000, VOTES testing was performed on Core Spray Motor Operated Valve (MOV) 2-1402-25B. During the performance of this test the closing torque switch was adjusted resulting in an increase in thrust of about 20.4 percent. During the post job review of the work package the MOV engineer recognized that the valve was a primary containment isolation valve, and As-Found and As-Left LLRTs had not been performed for a thrust change greater than 5 percent, in accordance with the 10 CFR 50, Appendix J Program. The valve was declared inoperable because the change in the torque switch setting could affect the leakage rate of the valve. The valve was retested and returned to its previously established seating force.

Technical Specification 3.7.D.1 required that with one or more primary containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either: a) restore the inoperable valves to OPERABLE status, or b) isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position, or c) isolate each affected penetration by use of at least one closed manual valve or blind flange. The Limiting Condition for Operation, of 4 hours, was exceeded placing the plant in a condition prohibited by Technical Specifications. The violation was minor because the increase in thrust did not affect the valve's ability to open if the core spray system was needed for injection. Additionally, because the other MOV in series with this valve remained operable, a complete loss of containment integrity did not occur. The inspector reviewed the licensee's corrective actions including recently developed Exelon Procedure ER-MW-301, "Rising Stem Motor-Operated Valve Diagnostic Testing," Revision 0, which specified precautions, notes, and signoffs to prevent a recurrence of this event.

Although this issue was entered into the licensee's corrective action program as CR# D2000-06892 and corrected, it constitutes a violation of minor significance that is not subject to enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

.9 (Closed) LER 50-237/01-04-00: Unit 2 Torus High Water Level Switches Failed Calibration Surveillance due to Historical Poor Post Modification Testing and Overly Conservative Technical Specification Allowable Value

The licensee revised the setpoint and relocated the switches to provide the appropriate actuation level. This LER is closed.

.10 (Closed) LER 50-249/00-03-01: Manual Scram on Loss of Vacuum from Air Binding of Condenser Tubes due to Off Gas Recombiner Train Failure

One Green finding was identified involving deficient human performance which resulted in operators manually scramming Unit 3 on May 4, 2002, due to degrading condenser vacuum conditions and condensate inlet temperature.

On May 4, 2000, following a forced outage, operators manually scrammed Unit 3 due to degrading condenser vacuum conditions and condensate inlet temperature. The licensee conducted a root cause investigation for this scram and determined that there were four contributors to this event: 1) failure of the 3A offgas system due to the installation of an incorrectly sized orifice; 2) failure by the system engineer to follow procedures when developing appropriate acceptance criteria for post modification testing of the system; 3) failure of the 3A offgas system flow indication caused by inappropriate application of the installed flow element; and 4) failure of the seal steam bypass valve yoke which was due to a procedural deficiency which allowed inappropriate use of the valve at reactor pressures higher than design. This issue was entered into the licensee's corrective action program as CR #D2000-02614.

This finding **(FIN 50-249/02-08-03(DRP))** is more than minor because the event was potentially an initiating event. This event had minimal safety significance (Green) as the operator action in scramming the unit was consistent with plant procedures and pre-briefed in accordance with conservative decision making philosophy. This LER is closed.

4OA5 <u>Other Activities</u>

<u>Operation of an Independent Spent Fuel Storage Installation (60855)</u>

.1 <u>Loading of the HI Storm Overpack</u>

a. <u>Inspection Scope</u>

The inspectors observed various portions of the loading of the Hi Storm overpack on May 22, 2002 to verify compliance with the applicable sections of the loading procedures.

b. <u>Observations and Findings</u>

Pre-job briefings were held and attended by all the workers involved with the evolution. Overall, the briefings went well with open exchanges of questions, clarifications and identification of each worker's roles and responsibilities.

Workers completed all the procedural tasks correctly during the loading. The team reinforced safety concerns. Radiation protection activities and controls were good. The inspectors noted good communication between workers and health physics personnel. Workers exhibited good radiation worker practices.

c. <u>Conclusions</u>

During the loading of the Hi Storm, the cask team demonstrated a thorough understanding of the procedures and activities. All activities observed by the inspectors were performed well.

.2 Handling Dry Cask With Unit 2 and 3 Reactor Building Crane

a. Inspection Scope

The inspectors reviewed a safety calculation and the crane operating procedure and interviewed licensee staff to evaluate dry cask handling safety using the Unit 2 and 3 crane.

b. Observations and Findings

After determining the limitations and restrictions for lifting and transporting a dry cask with the Unit 2 and 3 crane from the safety calculation, the inspectors observed that the dry cask transfer safe path yellow line was painted on the Unit 2 and 3 reactor building floor. The inspectors determined the transfer path was within the limits of the safety calculation. During discussions with a fuel handler, the inspectors determined that the crane had electrical interlocks to ensure the crane traveled on the safe load path. The inspectors also determined the fuel handlers had a good working knowledge of the crane operating procedure which the inspectors had reviewed. The procedure was not required to be in the operator's hand while operating the crane.

c. Conclusions

Effective implementation of appropriate controls contributed to safe handling of a dry cask with the Unit 2 and 3 reactor building crane.

4OA6 Exit Meetings

The results of the Safeguards inspection were presented to Ms. V. Gengler at the conclusion of the inspection on June 17, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

The senior radiation specialist presented his inspection results to Mr. D. Bost and other members of licensee management at the conclusion of the inspection of June 28, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

The resident inspectors presented their inspection results to Mr. D. Bost and other members of licensee management at the conclusion of the inspection on June 25, 2002. The licensee acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Violation

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

<u>NCV Tracking Number</u>	<u>Requirement Licensee Failed to Meet</u>
NCV 50-249/02-08-04	Technical Specification 3.0.4 requires when an LCO is not met, entry into a MODE or other specified condition in the applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the applicability for an unlimited period of time. Technical Specification 3.5.1 requires HPCI to be operable in Modes 1, 2, and 3 when reactor steam dome pressure is equal to or greater than 150 psig. On March 23, 2002, during startup from a maintenance outage on Unit 3, the operations crew erroneously determined that the high pressure coolant injection system (HPCI) was not required to be aligned in its operational standby readiness mode prior to reactor steam dome pressure reaching 150. Subsequently, the steam dome pressure exceeded 150 psig. The issue was evaluated in the phase 2 significance determination process and was determined to be of very low significance (Green) because all the other mitigating systems were available.

KEY POINTS OF CONTACT

Licensee
R. Bauman, ISI Coordinator
S. Bell, Health Physicist
D. Bost, Station Director
K. Bowman, Operations Director
H. Bush, Lead Radiation Protection Supervisor
V. Castle, Training Operations Manager
J. DeYoung, Corporate EP Specialist
J. Ellis, Performance Monitoring Group Lead
T. Fisk, Chemistry Manager
M. Friedman, Emergency Preparedness Coordinator
J. Ferguson, ALARA Analyst
V. Gengler, Dresden Site Security Director
R. Geier, RV/ISI NDE Coordinator
K. Hall, NDE Level III
S. Hunsader, Corporate Maintenance Rule Owner
T. Luke, Director, Engineering
R. May, NDE Level III
C. Melgoza, ALARA Analyst
D. Nestle, Radiation Protection
L. Oshier, Radiation Protection Technical Support Supervisor
M. Overstreet, Radiation Protection Shift Supervisor
M. Phelan, Assistant Radiation Protection Manager
R. Ruffin, Regulatory Assurance - NRC Coordinator
R. Rybak, Acting Regulatory Assurance Manager
J. Sipek, Nuclear Oversight Director
N. Spooner, Site Maintenance Rule Coordinator
W. Stoffels, Maintenance Director
B. Hovey, Site Vice President
S. Taylor, Radiation Protection Director
D. VanAken, Corporate EP Specialist

NRC
M. Ring, Chief, Division of Reactor Projects, Branch 1
D. Smith, Dresden Senior Resident Inspector
B. Dickson, Dresden Resident Inspector
R. Lerch, Project Engineer
P. Pelke, Reactor Engineer

IDNS
R. Zuffa, Illinois Department of Nuclear Safety
C. Mathews, Illinois Department of Nuclear Safety
D. Semel, Illinois Department of Nuclear Safety

Contractor
A. Lewis, REMP Technician, Environmental Incorporated - Midwest Laboratory

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-249/02-08-01	NCV	Potential to Lift Standby Liquid Control Pump Discharge Relief Valves During ATWS (Anticipated Transients Without Scram) Transient
50-237/02-08-02	NCV	Instrument Mechanic Failed to Follow Procedure During Isolation Condenser System Testing
50-249/02-08-03	FIN	Deficient Human Performance Associated with Offgas System Testing Contributed to a Manual Scram of Unit 3
50-237/249/02-08-04	NCV	Violation of Technical Specification Limiting Condition for Operation 3.0.4 With High Pressure Coolant Injection Inoperable with Reactor Steam Dome Pressure 150 psig

Closed

50-249/02-08-01	NCV	Potential to Lift Standby Liquid Control Pump Discharge Relief Valves During ATWS (Anticipated Transients Without Scram) Transient
50-237/02-08-02	NCV	Instrument Mechanic Failed to Follow Procedure During Isolation Condenser System Testing
50-249/02-08-03	FIN	Deficient Human Performance Associated with Offgas System Testing Contributed to a Manual Scram of Unit 3
50-237/249/02-08-04	NCV	Violation of Technical Specification Limiting Condition for Operation 3.0.4 With High Pressure Coolant Injection Inoperable with Reactor Steam Dome Pressure 150 psig
50-237/01-13-03	URI	Inadequate Surveillance Acceptance Criteria
50-237/249/02-04-01	URI	Corrective Action Program Change not Factored into MR Program
50-249/02-04-02	URI	Potential Violation of Technical Specification Limiting Condition for Operation 3.5.1 Due to Improper Alignment of High Pressure Coolant Injection
50-249/2002-001-00	LER	High Pressure Coolant Injection not in Standby Operation When Required by the Technical Specifications

50-237/2002-001-00	LER	Unit 2 Isolation Condenser Time Delay Relay Surveillance Failures Due to Setpoint Tolerance Specified with No Margin
50-237/2002-002-00	LER	Smoke Purge Mode Operation Prevents the Fulfillment of the Safety Function of the Control Room Emergency Ventilation System
50-237/2001-001-00	LER	Primary Containment Isolation System Valve Adjusted During Valve Operation Test and Evaluation System (VOTES) Test with No Local Leak Rate Test (LLRT) Performed
50-237/2001-004-00	LER	Unit 2 Torus High Water Level Switches Failed Calibration Surveillance due to Historical Poor Post Modification Testing and Overly Conservative Technical Specification Allowable Value
50-249/2000-003-01	LER	Manual Scram on Loss of Vacuum from Air Binding of Condenser Tubes due to Off Gas Recombiner Train Failure

Discussed

50-249/01-21-01	URI	Past operability of the HPCI system with a degraded support was indeterminate because there was no consideration for transient load that had damaged the support.

LIST OF ACRONYMS USED

ALARA	As Low As Is Reasonably Achievable
ATWS	Anticipated Transients Without Scram
CFR	Code of Federal Regulations
CR	Condition Report
CREVS	Control Room Emergency Ventilation System
DES	Dresden Electrical Surveillance
DIS	Dresden Instrument Surveillance
DOS	Dresden Operating Surveillance
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
gpm	gallons per minute
HPCI	High Pressure Coolant Injection
HVAC	Heating Ventilation and Air Condition
IDNS	Illinois Department of Nuclear Safety
IM	Instrument Mechanic
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MWe	megawatts electrical
MOV	Motor Operated Valve
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
OA	Other Activities
ODCM	Offsite Dose Calculation Manual
PI	Performance Indicator
psig	pounds per square inch gauge
RCA	Radiologically Controlled Area
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specifications
RP	Radiation Protection
SBLC	Standby Liquid Control system
SCFM	Standard Cubic Feet per Minute
SDP	Significance Determination Process
URI	Unresolved Item
VOTES	Valve Operation Test and Evaluation System
WO	Work Order

LIST OF DOCUMENTS REVIEWED

1R01 Adverse Weather Protection

	Dresden Summer Readiness 2002 Checklist	
Exelon Procedure OP-AA-108-109	Seasonal Readiness	Revision 0
DOA 0010-02	Tornado Warning / Severe Winds	Revision 4
DOS 0010-27	Securing from Cold Weather Operations for Unit 3	Revision 4
DOS 0010-24	Securing from Cold Weather Operations for Unit 2	Revision 5

1R04 Equipment Alignment

CR 00110925	Small oil leak (<1 drop per minute) 2/3A isolation condenser make up pump day tank	June 6, 2002
CR 00108935	NRC concerns on unit 2 isolation condenser	May 21, 2002
CR 00106557	Buzzing sound coming from unit 2 high pressure coolant injection undervoltage relay. Device # 27.	May 3, 2002
CR 00107352	Pipe hangers not supporting unit 2/3 torus pump-down line	May 9, 2002
CR 00105604	Portable scaffold against low pressure coolant injection system valve actuator	April 30, 2002
DOP 1300-M1/E1	Isolation Condenser System	Revision 13
DOP 1300-M2	Isolation Condenser Makeup Pump Fuel Oil System	Revision 03
DOP 1100-E1	Standby Liquid Control Electrical	Revision 03
DOP 1100-M1	Standby Liquid Control System	Revision 12
DOP 1100-M1/E1	Unit 3 Standby Liquid Control (SBLC) System Checklist	Revision 11
DOP 6600-M2	Unit 2/3 Standby Diesel Generator	Revision 20
DOP 6600-M1	Unit 2 Standby Diesel Generator	Revision 23
DOP 1400-M1/E1	Unit 3 Core Spray System	Revision 17

DOP 1400- M2	Emergency Core Cooling System	Revision 8

1R05 Fire Protection

CR 00112939	Unexpected entry into technical requirements manual	June 23, 2002
CR 00112316	NRC concern regarding fire suppression in isolation condenser pump house	June 19, 2002
CR 00096359	NRC identifies lack of access to bus 31 area for firefighting	February 26,

1R06 Flood Protection

CR 00113075	Updated Final Safety Analysis Report update not per requirements of 10 CFR 50.71(e)	June 28, 2002
CR 00111126	Generating stations emergency plan isolation condenser make up pump not in a position to effectively use in a generating stations emergency plan	June 8, 2002
CR 00111005	NRC identifies weakness in external flood procedure	June 5, 2002

1R12 Maintenance Rule Implementation

CR 00102380	Unit 3 east low pressure coolant injection submersible sump pump (A,B) maintenance rule functional failure on 9/17/00	April 4, 2002

1R13 Maintenance Risk Assessments and Emergent Work Control

WO 0039034601	(DOS 1100-04) Oil sample for 2B standby liquid control gearbox and pump	
CR 00111319	Debris disposal from open systems	June 10, 2002
CR 00110857	Maintenance rule database data inaccurate	June 5, 2002
CR 00110467	Replacement of TT 2/3-5731-7A	June 3, 2002
CR 00107314	Unnecessary high risk evolution found during walkdown	May 10, 2002
CR 00107223	SSR tables do not identify low pressure coolant injection system valve as required.	May 7, 2002

WO 9906551101	34-1-4160KV to bus 24-1 tie breaker	
WO 0036557303	Unit 2 250Vdc battery charger replacement	
WO 9906668901	Unit 2, Division I low pressure coolant injection and containment cooling service water system	
WO 99200909	Unit 3 125Vdc battery charger 4-hour load test	

<u>1R14</u> <u>Nonroutine Evolutions</u>

CR 00106822	Multiple intermediate range monitor 15 spurious alarms in the main control room	May 6, 2002
CR 00106578	Enter DEOP 300-01 & Technical Specification 3.6.4.1 for loss of reactor building differential pressure	May 6, 2002
CR 00105754	3B reactor recirculation pump speed oscillation, scoop tube locked	April 28, 2002

<u>1R15</u> <u>Operability Evaluations</u>

CR 00113397	DIS 250-3 data sheet expanded tolerance exceeds technical specifications value	June 28, 2002
CR 00113216	Engineering operability determination focus area self-assessment identifies operability determination corrective actions without ATIs	June 28, 2002
CR 00113230	Engineering operability determination focus area self-assessment identifies 5 occurrences of superceded procedure usage	June 28, 2002
CR 00112351	Significance Determination Process DIS 500-10 LS 3-302-82M found out of tolerance	June 18, 2002
CR 00108145	Condition reports not generated for missed operability determination actions	May 20, 2002
CR 00108128	Reactor protection system cable trays missing covers in various areas of the plant	May 17, 2002
CR 00104809	Scram discharge volume level switch calculation not completed on time per operability determination	April 22, 2002

CR 00103133	Historical operability determination administrative closure deficiencies	April 22, 2002
CR 00105114	2 technical support center penetrations made without approval	April 23, 2002
CR 00104834	Focus area self-assessment identifies operability determination 99043 closure deficiency	April 22, 2002

1R19 Post Maintenance Testing

CR 00106214	Missed post maintenance test due to clearance order still being in place	May 14, 2002
CR 00105057	Alternating current feed breaker for 3-83125-3 battery charger tripped	April 23, 2002
CR 00103037	Reactor recirculation motor generator 2A1 oil pump failed post maintenance test	April 9, 2002
WO 00442050	DOS 1600-04, unit 2/3 quarterly valve timing	Revision 17
WO 00442754-01	Repairs to control room HVAC refrigeration condenser unit service water flow control valve 2/3-3999-332	
WO 00442924	Repairs to control room HVAC refrigeration condenser unit service water inlet valve, 2/3-3999-334	
WO 433273-01	Repair to control room HVAC refrigeration condenser unit service water inlet check valve #1	
DOS 1400-05	Unit 3 Core Spray System Pump Test with Torus Available	Revision 26

1R22 Surveillance Test

CR 00113412	Calibration of electromagnetic relief valve pressure switches being performed monthly	June 28, 2002
CR 00112493	PS 32-305-130-38-35 out of tolerance during DIS 300-02	June 27, 2002
DOS 1600-21	Drywell high radiation monitor channel functional test	Revision 7
DIS 2300-08	Unit 2 Torus Level Switches Functional Test	Revision 20
DIS 2300-16	Torus Level Switches Channel Calibration	Revision 01

WO 430471-01	DIS 0500-01 Reactor Vessel High Pressure Scram Pressure Switch Calibration	Revision 13
CR 00111818	Seating thrust on 3-1402-38B found exceeded seismic limit	June 12, 2002
CR 00111999	During DIS 250-02 setpoint change found 2-261-30B out of tolerance	June 14, 2002
CR 00111654	Diesel oil storage tank level switches out of tolerance	June 12, 2002
CR 00108864	Instrument maintenance department unsatisfactory relay operation during DIS 1300-03 logic	May 22, 2002
CR 00108853	Performance of DIS 1300-03 PM isolation condenser initiation logic	May 21, 2002
CR 00108598	GE monitor 2 failed source check	May 20, 2002
CR 00107478	Unplanned technical specification entry for unit 3 125 volt batteries	May 10, 2002
CR 00104596	Containment cooling service water flow transmitter found out of tolerance 2-1556-A	April 19, 2002
CR 00102958	Unable to perform seat leakage test of low pressure coolant injection pump suction motor operated valve	April 8, 2002
CR 00102901	Found 3 out of 16 switches out of tolerance, non technical specification	April 8, 2002
CR 00102063	Analog trip system trouble alarm with maximum torus cooling mode	April 2, 2002
CR 00102451	During DIS 250-02 instrument maintenance department found 2-261-30B & D switch out of technical specification	April 4, 2002
CR 00108864	Instrument maintenance department unsatisfactory relay operation during DIS 1300-03 logic	May 22, 2002
DOP 700-06	Traversing incore probe (TIP) system operations	Revision 16
WO 427745	DTS 8236 Whole Core LPRM Calibration	Revision 15
WO 378502-01	Jumper installed to enable ball check valve permissive	

DOS 0202-02	Jet pump operability and degradation	Revision 23
DIS 1300-03	Isolation condenser initiation and isolation logic system functional test	Revision 16
WO 99193539	DOS 1600-28, Air operated valve fail safe and accumulator integrity test	Revision 2
WO 99177201-01	DIS 1300-03, Isolation condenser initiation and isolation logic system functional test	Revision 16
WO 00391166-01	DOS 6600-0, Diesel Generator Surveillance Tests	Revision 75
WO 00421881-01	DOS 1500-02 Containment Cooling Service Water Pump Test and Inservice Test	Revision 42
WO 0042538301	DOS 1100-04 Quarterly Standby Liquid	Revision 23
WO 0042538401	Control system pump test for the Inservice testing program	

2PS3 Radiological Environmental Monitoring and Radioactive Material Control Programs

	2000 Annual Radiological Environmental Operating Report	April 2001
	2001 Annual Radiological Environmental Operating Report	April 2002
	Murray and Trettal, Inc Monthly Reports on the Meteorological Monitoring Program at the Dresden Nuclear Station	January 2001 thru April 2002
EIML-SPM-1-16	Sampling Procedures Manual Environmental Incorporated - Midwest Laboratory	Revision 6
RP-AA-651	REMP Program Management	Revision 2
Offsite Dose Calculation Manual, Dresden Annex, Chapters 11 and 12.5	Radiological Environmental Monitoring Program	Revisions 1.5 (Chapter 11) and 1.13 (Chapter 12)
	10 CFR Part 61 Analyses - Difficult to Measure Nuclide Evaluation Effective Cobalt-60 Dose Consequence	April 2001
RP-AA-350	Assessment of Radiologically Contaminated Personnel	Revision 0

RP-AA-304	Unconditional Release Surveys	Revision 4
CR # 00108656	Purple Socket and Yellow Velcro Strap Found Outside the RCA	May 17, 2002
CR # D2001-03457	NRC Identifies Uncontrolled Exit from RPA	June 29, 2001
CR # 00109755	RAM Movement with no RP Escort	May 29, 2002
CR # D2001-03445	Revision Needed to Vendor's REMP Sampling Procedures Manual	June 29, 2001
CR # 00080032	RAM Discovered in Trash Container - Vendor Break Area	October 23, 2001
CR # 00080245	Purple Painted Tool Found Outside RPA at U3 West Access	October 24, 2001
CR # 00112799	2001 Annual Radiological Environmental Report Errors	June 21, 2002
CR # 00112803	Focus Area Self-Assessment Identifies Procedure Issue	June 21, 2002
	Radiation Protection Program Related CR Database	January 2001 - June 21, 2002
Focus Area Self-Assessment Report	REMP and Rad Material Control	June 10 - 14, 2002
Environmental Inc. Midwest Lab	Flowmeter Calibration Accuracy Certificates	Various Certificates for January 2001 - May 2002
Root Cause Investigation Report	Analysis of Radiation Protection Programmatic Health	April 17, 2002

<u>1EP4</u> <u>Emergency Action Level and Emergency Plan Changes</u>

	Dresden Station Annex to the Exelon Nuclear Standardized Radiological Emergency Plan	Revision 13
	Dresden Station Annex to the Exelon Nuclear Standardized Radiological Emergency Plan	Revision 14

<u>3PP4</u> <u>Security Plan Changes</u>

	Dresden Nuclear Power Station Security Plan	Revision 67 dated March 2002

4OA1 Performance Indicator Verification

LS-AA-2140, Attachment 1	Monthly Performance Indicator Data Elements for Occupational Exposure Control Effectiveness	October 2001 - May 2002
LS-AA-2150, Attachment 1	Monthly Performance Indicator Data Elements for RETS/ODCM Radiological Effluent Occurrences	October 2001 - May 2002
	Electronic Dosimetry Dose Alarm Investigation Reports	Selected Reports for October 2001 - May 2002
	Radiation Exposure Investigation Reports	Selected Reports for October 2001 - May 2002
	Gaseous and Liquid Effluent Release Data	Selected Data for October 2001 - May 2002

4OA3 Event Followup

DRE01-0072	HPCI pipe support historical operability analysis for transient loads	Revision 0
DRE01-0074	Dresden unit 3 HPCI historical operability analysis due to failed support M-1187D-80	Revision 0
DRE01-0076	Analysis of HPCI injection piping dynamic loads	Revision 0
DRE01-0078	HPCI pipe support operability analysis for steam transient loads	Revision 0
DRE01-0079	HPCI piping operability analysis for steam transient loads	Revision 0

4OA5 Other Activities

Calculation No. DRE98-0020	Reactor Building Crane Load Capacity	Revision 1
DFP 0800-72	HI Storm Processing at the Cask Transfer Facility	Revision 13
DFP 0800-20	Operation of the 2 and 3 Reactor Building 125 and 9 Ton Crane	Revision 17