March 2, 2001

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

SUBJECT: LASALLE COUNTY STATION - NRC INSPECTION REPORT 50-373/01-02(DRP); 50-374/01-02(DRP)

Dear Mr. Kingsley:

On February 10, 2001, the NRC completed an inspection at your LaSalle County Station. The enclosed report presents the results of that inspection. The results of this inspection were discussed on February 9, 2001, with Mr. C. Pardee and other members of your staff.

The inspection was an examination by the resident inspectors of activities conducted under your license as they relate to reactor safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the inspectors identified one issue of very low safety significance that was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it was entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this Non-Cited Violation, you should provide a response with a basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at LaSalle County Station.

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/

Bruce Burgess, Chief
Reactor Projects Branch 2

Docket Nos. 50-373; 50-374
License Nos. NPF-11; NPF-18

Enclosure: Inspection Report 50-373/01-02(DRP);
50-374/01-02(DRP)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373, 50-374
License Nos:	NPF-11, NPF-18
Report Nos:	50-373/01-02(DRP); 50-374/01-02(DRP)
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	January 1 through February 10, 2001
Inspectors:	E. Duncan, Senior Resident Inspector P. Krohn, Resident Inspector D. Schrum, Reactor Engineer, DRS J. Yesinowski, Illinois Department of Nuclear Safety
Approved by:	Bruce Burgess, Chief Reactor Projects Branch 2 Division of Reactor Projects

NRC's REVISED REACTOR OVERSIGHT PROCESS

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

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

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness	• Occupational • Public	• Physical Protection

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

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

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

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

SUMMARY OF FINDINGS

IR 05000373-01-02; IR 05000374-01-02; on 01/01-02/10/2001; Exelon; LaSalle County Station, Units 1 & 2; Temporary Modifications.

The inspection was conducted by the resident inspectors. The inspection identified one "Green" finding. The significance of an inspection finding is indicated by its color (Green, White, Yellow, Red) and was determined by using Inspection Manual Chapter 0609 "Significance Determination Process" (SDP).

A. <u>Inspector Identified Findings</u>

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

Green. During a review of a temporary modification associated with the reactor recirculation system, the inspectors identified that the licensee had failed to recognize that the temporary modification defeated a reactor recirculation pump control circuit safety interlock which prevented shifting a reactor recirculation pump from slow to fast speed. One Non-Cited Violation was identified. (Section 1R23).

B. <u>Licensee Identified Violations</u>

Violations of very low significance which were identified by the licensee have been reviewed by the inspector. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status: Both units operated at or near full power until January 31, 2001, when the Unit 1 reactor automatically shutdown due to a failure of the first “C” phase underslung and upright insulators on the Main Power Transformer output to the switchyard that initiated a generator trip. The problem was repaired and Unit 2 was restarted and synchronized to the grid on February 3. Both units operated at power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment - Partial Walkdown

a. Inspection Scope

During the inspection period, the inspectors performed a walkdown of accessible portions of the 2A Residual Heat Removal (RHR) system to verify system operability during maintenance on the 2B RHR system; the Unit 1 and Unit 2 Containment Combustible Gas Control System during several maintenance activities affecting Unit 1 and Unit 2 hydrogen recombiner unit crosstie capability, and the Unit 1 High Pressure Core Spray (HPCS) system to verify system operability during maintenance on the Reactor Core Isolation Cooling (RCIC) system. The inspectors reviewed documentation to determine correct system lineup. These documents included plant procedures, such as abnormal and emergency operating procedures, as well as plant drawings. Specifically, the inspectors verified the correct valve position of all critical valves in the primary system flowpath using the system piping and instrumentation drawings and the system mechanical checklist. The inspectors used the electrical system checklist to verify proper breaker alignment. The inspectors identified any discrepancies between the existing equipment lineup and the correct lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

The inspectors performed a routine walkdown of the following risk significant areas looking for any fire protection degradations:

- Fire Zone 3G: Unit 2 Reactor Building Elevation 710'
- Fire Zone 3H1: Unit 2 Reactor Building Elevation 694'
- Fire Zone 3I5: Unit 2 Reactor Building - 2A RHR Pump Cubicle

Emphasis was placed on the control of transient combustibles and ignition sources; the material condition, operational lineup, and operational effectiveness of the fire protection systems, equipment, and features; and the material condition and operational status of fire barriers used to prevent fire damage or fire propagation.

In particular, the inspectors verified that all observed transient combustibles were being controlled in accordance with the licensee's administrative control procedures. In addition, the inspectors observed the physical condition of fire detection devices, such as overhead sprinklers, and verified that any observed deficiencies did not impact the operational effectiveness of the system. The physical condition of portable fire fighting equipment, such as portable fire extinguishers, was also observed and verified to be located appropriately, and that access to the extinguishers was unobstructed. Fire hoses were verified to be installed at their designated locations and the physical condition of the hoses was verified to be satisfactory and access unobstructed. The physical condition of passive fire protection features such as fire doors, ventilation system fire dampers, fire barriers, fire zone penetration seals, and fire retardant structural steel coatings was inspected and verified to be properly installed and in good physical condition.

On January 16, 2001, the inspectors also performed a walkdown of the following areas looking for any fire protection degradations or issues when the Unit 1 turbine lube oil reservoir vapor extractor failed. The failure of the vapor extractor created a fine oil mist in the turbine lube oil reservoir bay that persisted for about 60 hours until the vapor extractor was returned to service.

- Fire Zone 5B3: Turbine Oil Package Room - Turbine Building, Elevation 735'
- Fire Zone 4D1: Unit 1 Cable Spreading Room - Auxiliary Building Elevation 731', West Wall Adjacent To Fire Zone 5B3
- Fire Zone 4D2: Unit 2 Cable Spreading Room - Auxiliary Building Elevation 731', West Wall Adjacent To Fire Zone 5B3

The inspectors searched for potential ignition sources, looked for deposition of lube oil film on equipment and electrical cable surfaces, verified no explosive atmospheric conditions existed, and inspected fire seal penetrations leading to adjacent safety-related areas.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

On January 24, 2001, the inspectors observed licensed operator requalification training simulator scenario SEG-01-05, "LaSalle Operating Surveillance (LOS) RI-Q3/Fire in 1A EDG [Emergency Diesel Generator] Room/Earthquake/Loss of 136Y-2/ RCIC [Reactor Core Isolation Cooling] Steam Leak With Failure to Isolate/Steam Leak in the Steam Tunnel/High Offsite Release," which involved a fire in the 1A EDG room, an earthquake which caused a steam leak which could not be immediately isolated, and a subsequent offsite release.

The inspectors verified crew performance in terms of clarity and formality of communication; the ability to take timely action in the safe direction; the prioritizing, interpreting, and verifying of alarms; the correct use and implementation of procedures, including alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; shift personnel oversight and direction provided by the shift manager, including the ability to identify and implement appropriate Technical Specification actions such as reporting and emergency plan actions and notifications; and group dynamics.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee implementation of the maintenance rule requirements, including a review of scoping, goal-setting, and performance monitoring, short-term and long-term corrective actions, and current equipment performance status. The systems selected for inspection were all classified as risk significant by the licensee's maintenance rule program. The systems evaluated were:

- Unit 1 and Unit 2 Auxiliary Power (AP)
- Unit 1 and Unit 2 Rod Position Indication System (RPIS)
- Unit 1 and Unit 2 Primary Containment Vent and Purge System (VQ)
- Unit 1 and Unit 2 Automatic Depressurization System (ADS)
- Unit 1 and Unit 2 Core Standby Cooling System (CSCS) Equipment Cooling (VY)

The Rod Position Indication System was selected based upon performance problems and classification as a maintenance rule (a)(1) system. The Auxiliary Power, Primary Containment Vent and Purge, Automatic Depressurization, and CSCS Equipment Cooling systems were classified as maintenance rule (a)(2) systems and were chosen based upon their relatively high risk significance.

The inspectors independently verified the licensee's implementation of maintenance rule requirements by verifying that these systems were properly scoped within the maintenance rule; that all failed structures, systems, or components (SSCs) were properly categorized and classified as (a)(1) or (a)(2); that performance criteria for SSCs classified as (a)(2) were appropriate; and that the goals and corrective actions for SSCs classified as (a)(1) were appropriate. The inspectors also verified that issues were identified at an appropriate threshold and entered in the corrective action program. The inspectors verified that the availability and functional failure data was accurate through a review of operator log entries, out-of-services, and work request documentation.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of plant risk, scheduling, configuration control, and performance of maintenance associated with planned and emergent work activities and verified that scheduled and emergent work activities were adequately managed. In particular, the inspectors reviewed the licensee's program for conducting maintenance risk safety assessments and verified that the licensee's planning, risk management tools, and the assessment and management of online risk was adequate. The inspectors also verified that licensee actions to address increased online risk during these periods, such as establishing compensatory actions, minimizing the duration of the activity, obtaining appropriate management approval, and informing appropriate plant staff, were accomplished when online risk was increased due to maintenance on risk-significant SSCs. The following specific activities were reviewed:

- The inspectors reviewed the risk impact associated with the unexcepted closures of Unit 2 reactor building ventilation system (VR) excess flow check damper 2VR091Y that occurred on January 8 and 10, 2001. On each occasion when 2VR091Y closed, ventilation flow through the main steam tunnel was lost. The loss of main steam tunnel ventilation caused a loss of high energy line break detection capability. The inspectors evaluated the impact of continued operation with main steam tunnel leak detection capability bypassed.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of January 21, 2001. This included work associated with the 2A Residual Heat Removal (RHR) system and the Unit 2 Motor-Driven Reactor Feedwater Pump.

- The inspectors reviewed the maintenance risk assessment for work planned for the week of January 28, 2001. This included work associated with the Unit 1 Reactor Core Isolation Cooling System and an 18-month preventive maintenance overhaul of the 2A Emergency Diesel Generator.

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

.1 Unit 1 Startup From Forced Outage L1F38

a. Inspection Scope

On January 31, 2001, Unit 1 automatically shutdown after experiencing an electrical transient which removed the main generator from service. The inspectors verified that the licensee's root cause evaluation identified and corrected the problem prior to plant startup. The inspectors observed portions of the Unit 1 restart activities, including approach to criticality, synchronization of the main generator to the grid, and power ascension.

b. Findings

No findings of significance were identified.

.2 Licensee Response to Unexpected Loss of Reactor Building Ventilation (VR)

a. Inspection Scope

On January 8 and 10, 2001, the inspectors observed operator and plant response to an unexpected loss of reactor building ventilation. In each case, reactor building exhaust ventilation excess flow check damper 2VR091Y unexpectedly closed causing a pressurization of the Unit 2 reactor building and an automatic trip of the reactor building ventilation supply fans. Loss of the reactor building ventilation system rendered the main steam tunnel high energy line break detection capability inoperable and caused entry into LaSalle Emergency Operating Procedure (LGA) 002, "Secondary Containment Control," alarm response procedure 2PM06J-A408, "Reactor Building Ventilation Exhaust Fans 2VR01CA/B/C No Air Flow," and LaSalle Abnormal Operating Procedure (LOA) VR-201, "Unit 2 Recovery From a Group 4 Isolation or Spurious Trip of Reactor Building Ventilation." The inspectors observed crew performance following each trip of the VR system and verified that the correct Technical Specification, emergency response procedure, alarm response procedure, and abnormal operating procedure actions were taken. The inspectors paid particular attention to bypassing main steam tunnel high differential temperature trip logic during the event response.

b. Findings

No findings of significance were identified.

.3 (Closed) Licensee Event Report (LER) 50-374/00-06: Unit 2 Scram on Turbine Control Valve Closure Due to High Reactor Water Level.

As discussed in Inspection Report 50-373/20000019; 50-374/20000019, on December 1, 2000, Unit 2 automatically shutdown during power ascension activities following the completion of LaSalle Refueling Outage L2R08. This event was entered into the licensee's corrective action program as Action Tracking Matrix (ATM) item 39916. No new issues were identified following a review of this LER. This item is closed.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations of degraded and non-conforming conditions affecting mitigating systems and barrier integrity to ensure that operability was properly justified and the component or system remained available, such that no unrecognized increase in risk had occurred. The following operability evaluations were reviewed:

- OE 95024 2B Residual Heat Removal (RHR) System Potential Waterhammer

This operability evaluation reviewed a potential waterhammer associated with the 2B RHR system which occurred on March 10, 1995 when the 2B RHR pump was started to flush the system. In the operability evaluation, the licensee documented that a walkdown was conducted and concluded that the system was operable. The inspectors reviewed OE 95024 and conducted an independent walkdown of piping associated with the system to verify that no damage to piping or piping supports was present.

- OE96006 1A RHR Room Cooler Is Not Available Which Affects the Operability of Shutdown Cooling

This operability evaluation reviewed the affects of unavailability of the 1A RHR room cooler on 1A RHR pump operability in the shutdown cooling mode of operation. The room cooler was considered essential support equipment for the 1A RHR pump since the cooler was needed to maintain the pump room temperatures below the environmental qualification limits. In the operability evaluation, the licensee concluded that although not operable for the technical specification shutdown cooling mode of RHR, the 1A RHR pump remained operable as an alternate method of decay heat removal. The inspectors reviewed the adequacy of the compensatory measures described in the operability evaluation for 1A RHR pump operation without the room cooler.

- OE 01-003 Unit 1 TCV [Turbine Control Valve] FAS [Fast Acting Solenoid] Response

This operability evaluation reviewed the impact of the failure of surveillance activity LaSalle Instrument Surveillance (LIS) RP-105AA, “Unit 1 Turbine Control Valve Fast Closure Trip Oil Pressure RPS [Reactor Protection System] A1 and B1 and EOC-RPT [End-of-Cycle Recirculation Pump Trip] System ‘A’ Relay Response Time Test (Run),” Revision 3. This surveillance was a response time test which verified that when a main turbine TCV closed, it energized a 500 pounds per square inch gauge (psig) emergency trip supply pressure switch. When the pressure switch closed, a trip signal was sent to the RPS within 80 milliseconds. The surveillance was initially considered inadequate since the response time was measured at 1100 milliseconds. A review of the testing methodology, however, revealed that the surveillance was excessively conservative and had inappropriately included the Fast Acting Solenoid (FAS) actuation time. With the FAS actuation time removed, the surveillance was repeated with acceptable results. The inspectors reviewed OE 01-003 and verified that the removal of the FAS actuation time from the overall testing time was appropriate to meet technical specification requirements.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed and observed the following post-maintenance testing activities involving risk significant equipment:

- Post Maintenance Testing for the Unit 2 Hydrogen Recombiner Accomplished In Accordance With Work Requests (WRs) 980093123, 980086515, 980086514, and 960045815.

- Post Maintenance Testing for the 2A Residual Heat Removal Pump Accomplished In Accordance With WRs 990120975 and 990237606.

- Post Maintenance Testing for the 2A Emergency Diesel Generator Accomplished in Accordance With WR 990059244.

During post-maintenance testing observations, the inspectors verified that the test was adequate for the scope of the maintenance work which had been performed, and that the testing acceptance criteria was clear and demonstrated operational readiness consistent with the design and licensing basis documents. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written and all testing prerequisites were satisfied; and that the test data was complete, appropriately verified, and met the requirements of the testing procedure. Following the completion of the test, the inspectors verified that the test equipment was removed, and that the equipment was returned to a condition in which it could perform its safety function.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors observed surveillance testing on risk-significant equipment and verified that the systems selected were capable of performing their intended safety function and that the surveillance tests satisfied the requirements contained in Technical Specifications, the Updated Final Safety Analysis Report (UFSAR), and licensee procedures. During surveillance testing observations, the inspectors verified that the test was adequate to demonstrate operational readiness consistent with the design and licensing basis documents, and that the testing acceptance criteria was clear. The inspectors also verified that the impact of the testing had been properly characterized during the pre-job briefing; the test was performed as written and all testing prerequisites were satisfied; the test data was complete, appropriately verified, and met the requirements of the testing procedure; and that the test equipment range and accuracy was consistent with the application, and the calibration was current. Following the completion of the test, the inspectors verified that the test equipment was removed and the equipment returned to a condition in which it could perform its safety function. When applicable, the inspectors also verified that test acceptance criteria specified in Technical Specifications was consistent with pump curves developed during pre-operational testing and was within the design analysis.

The following surveillance testing activities were observed:

- LaSalle Operating Surveillance (LOS) RH-01, Attachment 2C, "Unit 2 'C' RHR System Operability and Inservice Test," Revision 47

- LOS-RH-01, Attachment 1C, "Unit 1 'C' RHR System Operability and Inservice Test," Revision 47

- LaSalle Technical Surveillance (LTS) 200-230, "Division 1 RHR Service Water Balance Test," Revision 1

- LOS-DG-M3, Attachment 1B, "1B DG Operability Test - Idle Start," Revision 45

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification 9900444, "Defeat the Reactor Recirculation Pump Down Shift Circuitry," Revision 0, dated January 5, 2001. The purpose of this temporary modification was to electrically defeat the reactor recirculation (RR) pump downshift circuitry for a low reactor water level signal at Level 3 on both the 1A and 1B RR pump control circuits during Unit 1 power operation. The inspectors reviewed several electrical drawings to verify that reactor protection system design and control functions were not affected by the leads lifted in the temporary modification and to ensure that there was no unintended adverse impact on Unit 1. The inspectors reviewed the associated safety evaluation, L01-0009, completed in accordance with 10 CFR 50.59 to ensure that the licensee understood the full impact of lifting the leads on RR pump operations. The inspectors also referenced vendor report GE-NE-B3500504-01, "LaSalle 1 & 2 Jet Pump Cavitation Protection Logic," dated June 1996 to determine the potential effects of cavitation on RR system performance. Finally, the inspectors reviewed the UFSAR to determine the design bases of the RR pump circuitry.

b. Findings

On January 4, 2001, with Unit 1 at full power, a dynamic compensator card in the 1A and 1B turbine-driven reactor feedwater (TDRFP) pump level control logic failed causing high and low reactor vessel level alarms with no change in actual reactor vessel level. Operators subsequently reduced reactor power to 87 percent and placed both TDRFPs in manual control. With the Unit stabilized, temporary modification 9900444 was completed and installed on January 5 to avoid an unnecessary downshift of the RR pumps from fast to slow speed caused by a spurious Level 3 reactor vessel level indication. The dynamic compensator card was subsequently replaced on January 8, and the feedwater and reactor water level control system restored to a normal configuration.

During an independent review of the impact of the lifted leads on reactor recirculation operation conducted on January 12, the inspectors identified that the full impact of temporary modification 9900444 on plant operations had not been fully understood. Licensee personnel failed to recognize that the lifted leads defeated a reactor recirculation pump control interlock that prevented shifting a reactor recirculation pump from slow to fast speed when reactor vessel level was less than the Level 3 setpoint. Specifically, the licensee failed to recognize that lifting leads DD-42 and DD-46 on drawings 1E-1-4205AK, Revision H and 1-E-1-4205AV, Revision H, affected contacts 3 and 4 associated with relays K129A and K129B on drawings 1E-1-4205AF, Revision M and 1E-1-4205AR, Revision M. Preventing contacts 3 and 4 from changing state defeated the control interlock that blocked shifting reactor recirculation pumps from slow to fast speed with reactor vessel level below the Level 3 setpoint.

The inspectors reviewed the circumstances surrounding the preparation of the temporary modification and safety evaluation and determined that the licensee missed several opportunities to understand the effect of lifting the leads on reactor recirculation pump operation. The opportunities included, but were not limited to the following:

- The review of safety evaluation L01-0009 was inadequate. Although portions of UFSAR, Appendix G, Section 2, "Recirculation System Design Description," and Section 3, "Safety and Power Generation Evaluations," were specifically referenced in the safety evaluation, licensee personnel failed to identify that UFSAR Section 3.3.3.10.2.2, "Reactor Normal Power Level Low Speed To High Speed Transfer," stated that, among other parameters, reactor recirculation pump upshift sequences could only occur if reactor water level was normal. The safety evaluation was reviewed and approved by three different engineers, all of whom failed to identify this impact.

- The review of temporary modification package 9900444 was inadequate. The temporary modification package was required since the leads remained lifted beyond a 24-hour period. The temporary modification package was reviewed by four different engineers, all of whom failed to identify this impact.

- The work planning process was inadequate. Work Request (WR) 99024666 was prepared to execute the lifting of the leads in the field. The WR had an independent technical review that provided an additional opportunity to identify the error, but was missed.

- The operating shift review and authorization of the temporary modification was inadequate. This final review included at least one licensed operator review of the temporary modification installation. The inspectors noted that during the period that the reactor recirculation leads were lifted, no operations personnel were aware of or received special operating instructions concerning the control interlock functions of the reactor recirculation pumps which had been bypassed.

This finding did have a credible impact on safety and since the failure to understand the full impact of defeating interlocks could reasonably be viewed as a precursor to a more significant event and, if left uncorrected, the same issue could become a more significant safety concern. However, the inspectors considered the issue to be of very low safety significance for three reasons. First, since an automatic reactor scram occurs at the reactor vessel Level 3 setpoint, there would be no reactivity consequences associated with defeating the interlock. Second, an additional cavitation prevention interlock was unaffected and remained operable throughout the event. Finally, operating procedures LOP-RR-05, “Changing Reactor Recirculation Pump Speed From Slow to Fast Speed,” Revision 28, and LOP-RR-06, “Start of a Reactor Recirculation Pump In Fast Speed,” Revision 26, provided guidance which prevented an upshift of reactor recirculation pumps to fast speed with reactor vessel level less than the Level 3 setpoint. Because the issue did have a credible impact on safety, the inspectors applied the phase 1 significance determination process to the finding. However, because the finding did not contribute to an increase in initiating event frequency, the finding screened out as Green.

10 CFR 50, Appendix B, Criterion III, “Design Control,” requires that design changes, including field changes, be subject to design control measures commensurate with those applied to the original design and that measures be established to ensure that deviations from such standards are controlled. The failure to identify that reactor recirculation pump interlocks were bypassed for a design change specified in Temporary Modification 9900444 was an example where design control measures were not commensurate to those applied to the original design, and was a violation of 10 CFR Part 50, Appendix B, Criterion III (50-373/01-002-01(DRP)). However, because of its low safety significance and because it was entered in your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC’s Enforcement Policy. The issue was entered into the licensee’s corrective action program as Condition Report (CR) L2001-00283.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee conduct of drills and critique of performance through the observation of simulator training scenario ESG00C5-35, "Feedwater Malfunction/Loss of Bus 113/Stuck Open SRV [Safety Relief Valve]/Broken SRV Tailpipe/Drywell Spray Failure/ADS [Automatic Depressurization System] Valve Failure," on January 25, 2001; and simulator training scenario ESG00C5-02, "Narrow Range Level Instrument Failure/CW [Circulating Water] Pump Trip/ATWS [Anticipated Transient Without Scram]/LOCA [Loss-of-Coolant-Accident]/PCIS [Primary Containment Isolation Signal] Failure," on February 9, 2001. The inspectors reviewed the scenario to identify the timing and location of classification, notification, and protective action measure activities, and for licensee expectations and response. The inspectors verified that these actions were accomplished in a timely manner.

During the simulator training scenarios, simulated events which occurred were properly classified. One event concerning a simulated stuck open SRV with a broken SRV tailpipe was properly classified as an Alert due to the loss of two fission barriers. A second event concerning an ATWS with a LOCA was properly classified as a Site Area Emergency for the ATWS and then properly upgraded to a General Emergency when core cooling was threatened.

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES

Cornerstone: Mitigating Systems

4OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed reported 4th quarter data for the Unit 1 Residual Heat Removal (RHR) System Unavailability performance indicator. The inspectors utilized the performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Indicator Guideline," Revision 0.

The inspectors reviewed Licensee Event Reports, operator log entries, maintenance rule functional failures, and out-of-service logs for periods of RHR system unavailability. The inspectors verified that planned and unplanned unavailability hours were characterized correctly in determining performance indicator results. The inspectors also verified performance indicator data through independent calculations.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

January 31, 2001, Unit 1 Reactor Scram

a. Inspection Scope

On January 31, 2001, the Unit 1 reactor automatically shutdown from full power due to a failure of the first “C” phase underslung and upright insulators on the Main Power Transformer (MPT) output to the switchyard that initiated a generator trip. This generator trip then resulted in a turbine trip and a reactor scram. In response to the event, the inspectors observed plant parameters and status, including mitigating systems and fission product barriers; evaluated the performance of mitigating systems and licensee actions; and confirmed that the licensee properly reported the event as required by 10 CFR 50.72. The inspectors determined that all systems responded to the event as designed, the automatic shutdown was not complicated by material condition deficiencies associated with mitigation equipment, and that no human performance errors complicated the event response. Details of the event were communicated to the region-based risk analysts who determined that the event was of low risk-significance.

b. Findings

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

NCV Tracking Number	Requirements Licensee Failed to Meet
NCV 374/2001002-02	10 CFR 50, Appendix B, Criterion XVI, requires that conditions adverse to quality be promptly identified and corrected. Open feed breakers on the Unit 2 Division 2 Post Loss-of-Coolant-Accident (LOCA) monitoring system which rendered the system inoperable were not promptly identified. This issue was entered into the licensee’s corrective action program as CR L2000-07353.
NCV 374/2001002-03	10 CFR 50, Appendix B, Criterion V, requires that activities affecting quality be performed in accordance with prescribed procedures. During chemistry sampling activities on January 20, 2001, chemistry personnel failed to properly adhere to a sampling procedure and inadvertently isolated the Unit 2 offgas pre-treatment radiation monitor, which rendered the monitor inoperable for about 10 minutes. This issue was entered into the licensee’s corrective action program as CR L2001-00353.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. C. Pardee and other members of licensee management on February 9, 2001. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Exelon

D. Bost, Site Engineering Manager
D. Enright, Operations Manager
R. Gilbert, Work Control Manager
F. Gogliotti, Design Engineering Supervisor
G. Kaegi, Site Training Manager
C. Pardee, Site Vice President
J. Pollock, System Engineering Manager
W. Riffer, Regulatory Assurance Manager
M. Schiavoni, Station Manager
S. Taylor, Radiation Protection Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/2001002-01;50-374/2001002-01	NCV	Inadequate Temporary Modification
50-374/2001002-02	NCV	Unit 2 Division 2 Post-LOCA Monitoring
50-374/2001002-03	NCV	Unit 2 Offgas Radiation Monitor Isolated

Closed

50-373/2001002-01;50-374/2001002-01	NCV	Inadequate Temporary Modification
50-374/2001002-02	NCV	Unit 2 Division 2 Post-LOCA Monitoring
50-374/2001002-03	NCV	Unit 2 Offgas Radiation Monitor Isolated
50-374/00-06	LER	Unit 2 Scram Due to High Vessel Level

Discussed

None