September 7, 2000

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

SUBJECT: LASALLE COUNTY STATION - INSPECTION REPORT 50-373/00-11(DRP); 50-374/00-11(DRP)

Dear Mr. Kingsley:

On August 11, 2000, the NRC completed an inspection at your LaSalle County Station. The enclosed report presents the results of this inspection. The results of this inspection were discussed on August 10, 2000, 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, verification of performance indicators, event followup, and to compliance with the Commissions 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 NRC identified one issue that was evaluated under the significance determination process and was determined to be of very low safety significance (Green). This issue involved a violation of NRC requirements. However, the violation was not cited due to the very low safety significance and because it was entered into your corrective action program. This Non-Cited Violation (NCV) is described in the subject inspection report. If you contest this NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-001; and the NRC Resident Inspector at the LaSalle facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available **electronically** for public inspection in the NRC Public Document Room **or** from the *Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from* the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Christine Lipa, Acting Chief
Reactor Projects Branch 2

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

Enclosure: Inspection Report 50-373/00-11(DRP);
50-374/00-11(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:	50-373/00-11(DRP); 50-374/00-11(DRP)
Licensee:	Commonwealth Edison Company
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	July 1 - August 11, 2000
Inspectors:	E. Duncan, Senior Resident Inspector P. Krohn, Resident Inspector
Approved by:	Christine Lipa, Acting 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

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

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

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

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

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

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

SUMMARY OF FINDINGS

IR05000373-00-11, IR05000374-00-11; on 07/01-08/11/2000; Commonwealth Edison Company; LaSalle County Station; Units 1 & 2; Operability Evaluations.

The inspection was conducted by the resident inspectors. The inspection was conducted of the following baseline activities: Equipment Alignment, Fire Protection, Maintenance Rule Implementation, Maintenance Risk/Emergent Work, Operability Evaluations, Operator Workarounds, Post Maintenance Testing, Surveillance Testing, Temporary Plant Modifications, Emergency Preparedness Drill Evaluation, and Performance Indicator Verification. The inspection identified one green issue, which was a Non-Cited Violation. The significance of the issue is indicated by the color (green, white, yellow, red) and was determined by the Significance Determination Process.

Cornerstone: Mitigating Systems

- Green. A Non-Cited Violation was identified because the design basis for the correction of suppression pool temperature for the effects of thermal stratification was not adequately translated into operating procedures.

  The issue was of very low safety significance because, after further review by the licensee, the correction factor was determined to be appropriate. (Section 1R15)

Report Details

Summary of Plant Status: Both units operated at or near full power for the entire inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment

Partial System Walkdowns

a. Inspection Scope

The inspectors walked down the following trains or systems during the inspection period, while the redundant train was out of service or while the train was considered a protected train to manage plant risk:

- Unit 1, Class 1E, 4160 Volts Alternating Current (VAC), safety-related electrical distribution
- Unit 2, Class 1E, 4160 VAC, safety-related electrical distribution
- Unit 1, Class 1E, 480 VAC, safety-related electrical distribution
- Unit 2, Class 1E, 480 VAC, safety-related electrical distribution
- Unit 1 Low Pressure Core Spray System

The inspectors verified the correct valve position of all the valves in the primary system flowpath using the system P&IDs and system mechanical checklist, and verified breaker alignments using the system electrical checklist. The inspectors verified critical portions of the redundant or backup system and identified any discrepancies between the existing equipment lineup and the correct lineup. The inspectors verified by direct observation the lubrication and cooling of major components. The inspectors also observed instrumentation valve configurations and verified appropriate meter indications. The inspectors verified proper installation of hangers and supports during the walkdown, and verified operational status of support systems by direct observation of various parameters. The inspectors also evaluated other conditions such as adequacy of housekeeping, the absence of ignition sources, and proper labeling.

b. Findings

There were no findings identified.

1R05 Fire Protection

Fire Area Walkdowns

a. Inspection Scope

The inspectors walked down the following Unit 1 and Unit 2 risk-significant areas to identify fire protection degradations:

Zone 4E1: Unit 1 Auxiliary Electric Equipment Room (AEER), Elevation 731'
Zone 4E2: Unit 2 AEER, Elevation 731'
Zone 4E3: Unit 1, Division 2, Essential Switchgear Room, Elevation 731'
Zone 4E4: Unit 2, Division 2, Essential Switchgear Room, Elevation 731'
Zone 4F1: Unit 1, Division 1, Essential Switchgear Room, Elevation 710'
Zone 4F2: Unit 2, Division 1, Essential Switchgear Room, Elevation 710'
Zone 4D1: Unit 1 Auxiliary Building, Cable Spreading Room, Elevation 749'
Zone 4D2: Unit 2 Auxiliary Building, Cable Spreading Room, Elevation 749'
Zone 4D3: Unit 1 Auxiliary Building, Electrical Equipment Room, Elevation 749'
Zone 4D4: Unit 2 Auxiliary Building, Electrical Equipment Room, Elevation 749'
Zone 5A4: Unit 1 and Unit 2 Cable Area, Turbine Building Elevation 749'

Emphasis was placed on 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 inspectors also verified the physical condition of portable fire fighting equipment, such as portable fire extinguishers, 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 also inspected.

b. Findings

During a walkdown of the Unit 1, Division I, Essential Switchgear Room (Fire Zone 4F1) on July 5, 2000, the inspectors identified a 2.75-inch diameter corebore in the overhead that was not sealed with any fire retardant material. The inspectors brought the observation to the attention of the site fire marshall who verified that the corebore was not sealed. The open corebore contained a 1-inch diameter electrical grounding strap that passed from the overhead of the Division I Essential Switchgear Room to the inside of breaker cubicle 1AP02E-4, “B” Reactor Recirculation Pump, located in the Division II Essential Switchgear Room.

The Unit 1, Division II, Essential Switchgear Room (Fire Zone 4E3) was located directly above the Unit 1, Division I Essential Switchgear Room. The open corebore discovered by the inspectors compromised the 3-hour external fire rating and 2-hour internal fire rating between the two fire zones. The licensee initiated fire impairment 1-2000-110-ATR and generated action request 990099133 to repair the open penetration between the two safety-related electrical switchgear rooms. The issue was entered into the licensee's corrective action program under Problem Identification Form (PIF) L2000-03778.

During subsequent extent-of-condition walkdowns, the licensee discovered a second unsealed penetration on July 11, 2000. This open penetration, also located in the ceiling of the Division 1 Essential Switchgear Room, passed to the inside of breaker cubicle 1AP06E-4, "B" Residual Heat Removal Pump, located in the Division II Essential Switchgear Room in safety-related bus 142Y. The second open corebore was entered into the licensee's corrective action program under PIF L2000-03839. Fire impairment 1-2000-111-ATR was initiated and action request 990099442 was generated to repair the open penetration.

Pending additional inspection effort and review of the risk significance, this is an unresolved item **(URI 50-373/2000011-01(DRP))**.

Annual Fire Drill Observation

a. Inspection Scope

On July 21, 2000, the inspectors observed the fire brigade respond to a simulated fire in Fire Zone 2D (Unit 1 Reactor Building Elevation 786') to evaluate the readiness of licensee personnel to prevent and fight fires. Aspects of the response which were reviewed by the inspectors included:

- Proper use of self-contained breathing apparatus
- Proper use of protective clothing
- Verification that fire hoses were capable of reaching all necessary fire hazard locations, that the lines were laid out without flow constrictions, the hoses were simulated as charged with water, and the nozzle patterns were tested prior to entering the fire area of concern
- Entry into the fire area in a controlled manner
- Sufficient fire fighting equipment available at the scene by the fire brigade to properly perform fire fighting duties
- Fire brigade leader communications effectiveness
- Radio communications effectiveness
- Effective smoke removal operations
- Use of pre-planned fire fighting strategies
- Adherence to the pre-planned drill scenario and success in meeting drill objectives

b. Findings

There were no findings 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 following systems were evaluated during the inspection period:

- Process Radiation Monitoring
- Electro-Hydraulic Control (EHC)
- Station Air

The inspectors independently verified the licensee's implementation of maintenance rule requirements for these systems 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) 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 process radiation monitoring system, which included the liquid process radiation monitors, consists of two RHR monitors, two service water system monitors and a radioactive waste monitor, which were classified as (a)(1) due to exceeding the reliability criteria. The inspectors verified that the goals for the process radiation monitors identified above were adequate and that all other monitors were scoped properly within the maintenance rule. The inspectors conducted a walkdown of the process radiation monitoring system and verified through direct observation of equipment material condition that no conditions existed that jeopardized system functionality.

The EHC system was classified as (a)(1) due to several instances between August 1999 and May 2000 of EHC solenoids either failing to actuate or failing to reset during periodic testing. The inspectors verified that the goals for the EHC system were adequate and that the (a)(1) action plans were appropriate and being effectively completed. The inspectors focused attention on the potential for common mode failures of the EHC system through fluid degradation and contamination. Vendor reports, local temperature profiles, oxidation potentials of the EHC fluid, and chemistry analyses were reviewed. In addition, 35 PIFs were selected for review to determine if there were any functional failures that had not been included in the maintenance rule data base. The inspectors conducted a walkdown of accessible portions of the EHC system and verified through direct observation that no material condition deficiencies existed that jeopardized system functionality.

The station air system was classified as (a)(1) due to a number of recent functional failures associated with the Unit 2 Station Air Compressor (SAC). The inspectors verified that the goals planned for the system were adequate, and that the potential impact of problems associated with the Unit 2 SAC on the Unit 0 and Unit 1 SACs was considered. The inspectors also reviewed the PIFs generated in response to problems identified with the system and verified that the Maintenance Rule Program had captured the maintenance preventable functional failures.

b. Findings

There were no findings identified.

1R13 Maintenance Risk Assessment and Emergent Work Prioritization

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 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 inspectors verified that troubleshooting evolutions and maintenance activities for emergent work were adequately controlled at the job site to minimize risk to the system worked on and that all activities were within the work control boundary. The following specific activities were reviewed:

- The inspectors reviewed the maintenance risk assessment for work planned for the week of July 10, 2000 and July 23, 2000. This included work associated with the Unit 1 turbine building closed cooling water (TBCCW) system, and the Unit 1 and Unit 2 motor-driven reactor feedwater pumps. The inspectors verified that the initiating event frequency data for a loss of TBCCW used in the LaSalle probabilistic risk assessment was reasonable.

- The inspectors reviewed the increased plant risk associated with emergent work activities for a 1B standby liquid control system which did not meet the flow requirements specified in the acceptance criteria following performance of surveillance testing.

- The inspectors reviewed the increased plant risk associated with emergent work activities for the Unit 1 and Unit 2 reactor core isolation cooling (RCIC) system which was identified with abnormally high particulate in the turbine oil system following a routine turbine oil analysis.

b. Findings

There were no findings identified.

1R14 Personnel Performance During Non-Routine Plant Evolutions

(Closed) Licensee Event Report (LER) 050-374/00-003-00: Scram on Low Reactor Water Level Due to Loss of 2A Turbine-Driven Reactor Feed Pump Flow. On June 22, 2000, Unit 2 experienced an automatic reactor scram due to low reactor water level. The low reactor water level was caused by a loss of flow from the 2A turbine-driven reactor feedpump (TDRFP). The root cause of the event was intrusion of wear products into the high pressure control oil pressure regulating relief valve. The wear products became lodged between the relief valve seat and disk, allowing oil system pressure to be relieved, and the 2A TDRFP control valve to close, causing the loss of flow from the pump. The resulting reactor water level transient led to the automatic scram. Corrective actions included the installation of duplex oil filters on the 2A and 2B TDRFP high pressure control oil systems.

The inspectors responded to the reactor scram as documented in Section 4OA3 of NRC Inspection Report 50-373/2000006(DRP); 50-374/2000006(DRP). Based on the inspectors' observations and a review of this LER, the inspectors determined that the scram was uncomplicated, all systems responded as expected, no human performance errors complicated the event response, and no emergency core cooling systems were challenged. This LER is closed.

(Closed) LER 050-373/00-002-00: Inadvertent Reactor Water Cleanup (RWCU) Isolation During System Startup Due to Personnel Errors.

As discussed in Section 4OA4 of NRC Inspection Report 50-373/2000004(DRP); 50-374/2000004(DRP), on April 26, 2000, Unit 1 RWCU inlet isolations valves 1G33-F004 and 1G33-F001 isolated on high flow immediately upon opening during restoration from maintenance. The valves were opened in accordance with LaSalle Operating Procedure (LOP) RT-02. "Reactor Water Cleanup System - Startup and Pump Transfer," Revision 23. Licensed control room operators failed to adequately perform prerequisite B.3 of LOP-RT-02, which required that the system be filled and vented in accordance with LOP-RT-01, "Reactor Water Cleanup System Filling and Venting," Revision 23. The operators incorrectly concluded that the prerequisite had been met based on an evaluation of the scope of work that had been performed while the system was out-of-service. The operators were unaware that the RWCU system was operating with known leaks that resulted in the system being depressurized and partially drained during the time the system was removed from service. In addition, LOP-RT-02 was inadequate since the prerequisites did not clearly state that the RWCU system needed to be pressurized in accordance with LOP-RT-01 when starting up at rated conditions.

The safety significance of the event was low since operation of the reactor was not directly or immediately affected by the isolation of the system during the return to service. The RWCU isolation valves operated as designed, and there was no containment boundary leakage from the system.

The failure of operators to verify the prerequisites of LOP-RT-01 were completed prior to restoring the RWCU system resulted in an unplanned engineered safety features actuation. The failure to verify the prerequisites of LOP-RT-02 were complete prior to system restoration was a violation of 10 CFR 50, Appendix B, Criterion V, which required activities affecting quality be accomplished in accordance with prescribed procedures. However, this failure constitutes a violation of minor significance and is not subject to formal enforcement action in accordance with Section IV of the NRC's Enforcement Policy. This LER is closed.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations (OE) 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 inspectors also reviewed whether the licensee had identified operability evaluation issues at an appropriate threshold and entered them into the corrective action program. The following operability evaluations were reviewed:

- OE 95029 Drywell Sump Cover Plated Not Bolted or Sealed
- Determination of Average Bulk Suppression Pool Temperature in Accordance With LOP-CM-03, "Suppression Chamber Average Water Temperature Determination," Revision 11, dated March 11, 1998

b. Findings

Suppression Chamber Average Water Temperature Determination

Brief Overview

The Suppression Chamber water temperature monitoring system is arranged with two instrumentation divisions and with 14 temperature detectors (channels) per division. Each division has seven associated instrumentation wells with two temperature detectors located in each well. These instruments are located approximately 12 inches below the top of the normal suppression pool water surface. These instruments are designed to provide the capability to monitor the suppression chamber water temperature following an accident.

Discussion

On July 15, 2000, licensee management issued a policy to address frequent suppression pool cooling and mixing evolutions due to safety relief valve (SRV) leakage and increased lake temperatures due to the summer weather period. That policy letter referenced the use of LOP-CM-03 to determine average suppression pool temperature. The inspectors reviewed LOP-CM-03 and determined that Step E.1.7 allowed 2 degrees fahrenheit to be subtracted from the average temperature indicated by the digital average suppression pool temperature display. The inspectors determined that this allowance was based on Calculation NSLD 3C7-0788-001, "Assessment of Bulk Pool Temperature Calculation Methods," Revision 1, dated July 28, 1988, which referenced Sargent & Lundy In-Plant SRV Test, "Extended Blowdown Test Evaluation of

Suppression Pool Temperature Measurements," dated August 1, 1983. The inspectors reviewed page 5-4 of this test report and determined that thermal stratification due to factors which act over long periods of time and cause the pool temperature to rise very slowly (such as chronic SRV leakage) was not measured and, therefore, adjusting the suppression pool bulk pool temperature for pool stratification did not appear justified.

The inspectors discussed this issue with licensee personnel. Licensee engineering personnel subsequently researched later suppression pool monitoring data acquired during slow, chronic suppression pool heatup evolutions. Through these reviews, the licensee was able to demonstrate that the 2 degree fahrenheit subtraction used in Step E.1.7 of LOP-CM-03 was conservative with respect to maximum suppression pool heat capacities described in the Unit 1 and 2 licensing and design bases.

Significance Determination Process (SDP)

The inspectors assessed this issue using the NRC's Significance Determination Process (SDP). Since the licensee was able to demonstrate that no actual loss of a safety function or system had existed while the 2 degree subtraction of LOP-CM-03 had been used, the issue was determined to be "Green" using the SDP, Phase 1.

Regulatory Requirements

10 CFR 50, Appendix B, Criterion III, "Design Control," requires that measures shall be established to assure that the design basis for structures, systems, and components, are correctly translated into specifications, drawings, procedures, and instructions. The failure to adequately evaluate the methodology and bases for determining average suppression pool temperature by subtracting 2 degrees fahrenheit in LOP-CM-03, Step E.1.7 was an example where the requirements of 10 CFR 50, Appendix B, Criterion III, were not met and was a violation. However, this Severity Level IV violation is being treated as a Non-Cited Violation **(NCV 50-373/2000011-02(DRP); 50-374/2000011-02(DRP))**, consistent with Section VI.A.1 of the NRC Enforcement Policy. This item was entered into the licensee's corrective action program as PIF L2000-04295.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors reviewed operator workarounds (OWA) and operator challenges (OC) to identify any potentially adverse impact on the function of mitigating systems or the ability to implement a LaSalle Abnormal Operating Procedure (LOA) or LaSalle Emergency Operating Procedure (LGA). The following items were reviewed:

- OWA 318/319: Instrument Nitrogen System Manual Drain Blowdown

Issue Description: During high drywell humidity conditions, water can enter the system. Automatic drain traps were valved out due to reliability concerns. As a result, operators were required to manually drain water from the instrument nitrogen system about once per shift.

The inspectors determined through a review of the Updated Final Safety Analysis Report, station procedures, and plant drawings, that manual operator action to drain moisture from the instrument nitrogen system had no impact on the function of any mitigating system. The inspectors also verified that in the event of a design basis accident, in which conditions may preclude manually draining moisture from the system, that no impact on the function of a mitigating system would result. In particular, the instrument nitrogen compressors and associated equipment were not required to safely shutdown the plant following a postulated loss-of-coolant-accident and/or loss of offsite power. Valves associated with systems required for safe reactor shutdown following an event, such as the automatic depressurization system, were provided with individual pneumatic accumulators, nitrogen bottles, and an emergency nitrogen pressurization station, to shutdown the plant and maintain the plant in a safe shutdown condition. The inspectors conducted a plant walkdown and verified that nitrogen bottles and an emergency nitrogen pressurization station described in plant LOAs and LGAs, such as LOA-IN-101, “Loss of Pneumatic Drywell Air Supply,” LGA-MS-01, “Using Main Condenser as Heat Sink in ATWS [Anticipated Transient Without Scram],” and LGA-MS-02, “Using Main Steam Lines for Emergency RPV [Reactor Pressure Vessel] Blowdown,” were available.

- OC 243/295: Standby Liquid Control (SBLC) Pump Discharge Pressure Gauge Oscillations

Issue Description: During routine surveillance testing of the SBLC system, the SBLC pump pressure gauge oscillates significantly since the system utilizes positive displacement pumps.

The inspectors verified that the discharge pressure gauge is in the test loop of the system and is not relied upon to provide system performance indication during any design basis accident. Its use was not referenced in LGA-SC-101(201), “Unit 1(2) Initiation of Standby Liquid Control,” or LGA-10, “Failure to Scram.” The inspectors also verified that operators appropriately used the mid-range of the oscillations to establish pump discharge pressure to simulate reactor pressure when performing quarterly testing to satisfy American Society of Mechanical Engineers (ASME) requirements.

- OC 229/286/287: Installed Level Indication for Diesel Generator Fuel Day Tanks Is Inadequate for the Required Surveillance Testing

Issue Description: During routine inservice testing of the capacity of the emergency diesel generator fuel oil transfer pumps, the installed day tank level indication has been inadequate for the required surveillance testing requiring operator use of temporarily installed tygon tubing at the side of the diesel fuel oil day tank.

The inspectors verified that flow orifices had been installed in the fuel oil transfer pump discharge lines to more accurately measure fuel oil transfer pump capacity. The flow orifices were added under design change packages for all of the Division I, II, and III emergency diesel generators. The inspectors verified that the changes had been successful for the Division I and II diesels and that a further design change to the Division III fuel oil transfer pump discharge path was pending.

b. Findings

There were no findings identified.

1R19 Post-Maintenance Testing

a. Inspection Scope

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.

The inspectors reviewed and observed the following post-maintenance testing activities involving risk significant equipment associated with work requests (WRs):

- WR 990190924 - Unsealed Penetration in Floor of Unit 1
- WR 990191108 - 3 inch Corebore With 1 inch Grounding Strap Does Not Contain Fire Seal

  The inspectors observed the performance of these maintenance activities which restored a 3-hour fire barrier which was degraded due to unfilled corebore holes in the Unit 1, Division 1, Essential Switchgear Room ceiling.

- 1B Emergency Diesel Generator Maintenance

  On July 25, 2000, the inspectors observed post-maintenance testing of the 1B Emergency Diesel Generator (EDG) air receiver inlet check valve for leak tightness in accordance with LaSalle Operating Surveillance (LOS) DG-Q3, “1B DG ‘B’ Air Compressor Check Valve, Attachment 4.” The inspectors also observed portions of the work associated with the semi-annual rebuild of the 1B EDG air compressor piston heads and verified that the work activity was properly executed.

- WR 990059731 - Unit 2 Service Water Strainer Maintenance

  The inspectors observed post-maintenance testing of Unit 2 service water strainer backwash valves 2WS130A and 2WS130B in accordance with WR 990059731, and procedure LOP-WS-05, “Service Water Strainer Operations,” Revision 7. The inspectors observed the timing and alignment of the strainer backwash valves and performed a walkdown of portions of the service water system during the backwash evolution.

- WR 980080676 - 2VQ048 Drywell Nitrogen Makeup Isolation Valve

  Following a scheduled breaker inspection, valve 2VQ048 was cycled for post-maintenance testing. The valve was a containment isolation valve with a safety-related power supply from motor control center 235Y-1, cubicle H6. The inspectors observed the cycling from the control room, examined the material condition of the valve in the reactor building, and verified that the motor control center cubicle had been properly restored and the breaker reset following the breaker inspection.

- WR 990143039 - Functional Check of 1B Primary Containment Chiller 1VP01CB

  Following trips of the 1B primary containment chiller on high bearing and high discharge temperature indications, the associated trip relay was replaced. The unit was subsequently restarted and monitored for abnormal operation. The inspectors examined the temporary monitoring devices and output data from the running chiller. The inspectors also performed a walkdown of the chiller to identify material condition deficiencies and abnormal operating parameters.

b. Findings

There were no findings 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 (TS), the 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 that the equipment was returned to a condition in which it could perform its safety function.

The following surveillance testing activities were observed:

- LOS-DC-Q2, “Battery Readings for Safety-Related 250 VDC [Volts Direct Current] and Division 1, 2, 3 125 VDC Batteries”

The inspectors reviewed UFSAR Section H.3.5.28, “Turbine Building Ground Floor General Area - Fire Zone 5C11, UFSAR Section 8.3.2, “DC [Direct Current] Power Systems,” TS 3.8.2.3, “D.C. Distribution - Operating,” and the associated TS Bases Section to ensure that all requirements were consistent with licensing and design basis documentation. The inspectors also reviewed Institute of Electrical and Electronics Engineers (IEEE) 308-1974 Edition, “Criteria for Class IE Electric Systems for Nuclear Power Generating Stations,” and IEEE 450-1980 Edition, “IEEE Recommended Practices for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations.”

- LOS-RH-Q1, “RHR (LPCI) [Low Pressure Coolant Injection] and RHR Service Water Pump and Valve Inservice Test For Operational Conditions 1, 2, 3, 4, and 5" - Attachment 1A, “Unit 1A RHR System Operability and Inservice Test,” and Attachment 1D, “Unit 1A RHR Service Water System Operability and Inservice Test.”

  The inspectors compared the surveillance acceptance criteria to TS and the plant design basis for consistency and accuracy. The inspectors also completed a system walkdown to identify deficiencies such as evidence of waterhammer.

- LOS-RI-Q5, “Reactor Core Isolation Cooling (RCIC) System Pump Operability, Valve Inservice Tests in Conditions 1,2,3 and Cold Quick Start - Attachment 2A,” Revision 13. The inspectors observed RCIC pump operation both locally and from the control room. The inspectors focused on RCIC turbine governor response due to high particulate levels in the turbine oil sump that had been detected during monthly oil samples. The particulate levels were reduced to comply with vendor recommendations through a feed and bleed process prior to performance of the surveillance test. The inspectors also performed a system walkdown to identify deficiencies such as leaking injection and steam supply valves as well as evidence of waterhammer.

- LOS-SC-M1, “SBLC Pump Operability Test and Explosive Valve Continuity Check.” The inspectors observed portions of the surveillance test locally and interviewed operators to determine the discharge capacity of the pump relative to surveillance and TS acceptance criteria. The inspectors also observed restoration of the system to standby status and operator self-checking and communication practices.

- LIS-NR-209, “Unit 2 APRM [Average Power Range Monitor] Gain Adjustment,” Revision 9. The inspectors observed instrument maintenance (IM) personnel adjust APRM output so that indicated reactor power matched actual, calorimetric determined reactor power.

- LIS-NR-207, “Unit 2 APRM/RBM [Rod Block Monitor] Flow Converter to Total Core Flow Adjustment,” Revision 12. The inspectors observed IM personnel calibrate and adjust the Unit 2 APRM/RBM flow converter to total core flow instruments. The inspectors observed the self-checking, peer checking, and calibration techniques used by IM personnel to provide an accurate indication of total core flow.

- LIS-NR-301, “Unit 1 Source Range Monitor Rod Block Functional Test,” Revision 14. The inspectors observed instrument maintenance personnel perform channel functional tests of each of the source range monitor instruments and verify outputs of the same instruments to control room panel indications.

b. Findings

There were no findings identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed Temporary Modification 9900344 which defeated the Unit 1 drywell cooler condensate high flow rate alarm. The purpose of this alarm was to warn operators of a potential loss-of-coolant-accident due an increase in the measured drywell cooler condensate flow. The inspectors verified that the temporary modification was installed in accordance with WR 990168623 and WR 990174708 and that pre-installation and post-installation testing was adequate to confirm that there was no unintended impact on the plant. The inspectors reviewed the associated safety evaluation for the temporary modification installation and the subsequent removal of the temporary modification in accordance with licensee procedure CC-AA-112, “Temporary Modifications,” Revision 2, Attachment H, Step B.2.

b. Findings

There were no findings identified.

1EP6 Emergency Preparedness Drill Evaluation

a. Inspection Scope

The inspectors evaluated the adequacy of the licensee conduct of drills and critique of performance through the observation of drill scenario SEG 00C4-03, “Loss of 135X-1, Stuck Open Safety Relief Valve (SRV), Manual Scram, SRV Tailpipe Break in Drywell, Failure of VP [Primary Containment Chill Water] to Isolate,” Revision 00. This scenario utilized licensed operators in the plant simulator in conjunction with emergency response organization personnel in the Technical Support Center. 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.

Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

a. Inspection Scope

The inspectors reviewed reported 2nd quarter data for the Safety System Functional Failure performance indicator for Unit 1 and Unit 2 utilizing the performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Indicator Guideline,” Revision 0.

The inspectors reviewed LERs and operator log entries to identify the number of safety system functional failures that occurred during the previous four quarters and compared that number to the number in the performance indicator. The inspectors also reviewed the licensee’s basis for excluding events and conditions identified in LERs from reporting as a safety system functional failure.

b. Findings

There were no findings identified.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. C. Pardee and other members of licensee management on August 10, 2000. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

ComEd

C. Pardee, Site Vice President
J. Meister, Station Manager
D. Bost, Site Engineering Manager
K. Bartes, Nuclear Oversight Manager
G. Kaegi, Site Training Manager
R. Gilbert, Operations Manager
F. Spangenberg, Regulatory Assurance Manager
J. Pollock, System Engineering Manager
F. Gogliotti, Design Engineering Supervisor
T. Gierich, Work Control Manager
J. Henry, Shift Operations Superintendent

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-373/2000011-01	URI	Unit 1 Degraded Fire Barrier
50-373/2000011-02; 50-374/2000011-02	NCV	Inadequate Suppression Pool Temperature Correction Design Basis

Closed

50-373/2000011-02; 50-374/2000011-02	NCV	Inadequate Suppression Pool Temperature Correction Design Basis
50-373/00-002-00	LER	Inadvertent RWCU Isolation During Startup
50-374/00-003-00	LER	Reactor Scram Due to 2A TDRFP Loss

Discussed

None.