April 16, 2004

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INTEGRATED INSPECTION REPORT 05000373/2004002;
05000374/2004002

Dear Mr. Crane:

On March 31, 2004, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your LaSalle County Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on April 13, 2004, with the Site Vice President, Mr. G. Barnes, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, there were two NRC-identified and two self-revealed findings of very low safety significance. All four of these findings involved violations of NRC requirements. However, because these violations were determined to be non-willful and non-repetitive, and because they were entered into your corrective action program, the NRC is treating them as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. Additionally, a licensee identified violation is documented in Section 4OA7 of this report.

If you contest the subject or severity of any Non-Cited Violation in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector Office at the LaSalle County Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/ RA /

Bruce L. Burgess, Chief
Branch 2
Division of Reactor Projects

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

Enclosure: Inspection Report 05000373/2004002;
05000374/2004002;
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - LaSalle County Station
LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
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REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2004002; 05000374/2004002
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	January 1 through March 31, 2004
Inspectors:	D. Kimble, Senior Resident Inspector D. Eskins, Resident Inspector D. Jones, Regional Engineering Inspector M. Mitchell, Radiation Protection Specialist G. Wilson, Senior Resident Inspector – Duane Arnold J. Yesinowski, Illinois Dept. of Emergency Management
Observers:	M. Franke, Inspector-in-Training W. Macon, NRR Senior Project Manager C. Roque-Cruz, NRC Intern
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000373/2004002, 05000374/2004002; 01/01/2004 -03/31/2004; LaSalle County Station, Units 1 & 2; Operator Workarounds and Access Control to Radiologically Significant Areas Report.

The report covers a 3-month period of baseline resident inspection, and announced baseline inspections on radiation protection and the inservice inspection program. The inspections were conducted by both resident and region-based inspectors. Four Green findings and four associated Non-Cited Violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be “Green,” or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. <u>Inspector-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors after the licensee throttled an instrument nitrogen system pressure regulator isolation valve without adequate written instructions in an attempt to compensate for a degraded pressure regulator. The licensee failed to adequately assess the impact of the valve throttling on N_2 system performance prior to the evolution, and, therefore, did not provide appropriate acceptance criteria in plant procedures regarding the extent to which the valve could be throttled closed before system operability was impacted.

  This finding was greater than minor because it had the potential to be a more significant safety concern. If left uncorrected, operations personnel could have throttled the isolation valve closed to the extent that the safety function of the subject N_2 header was lost. The finding was of very low safety significance because a licensee engineering evaluation subsequently determined that the isolation valve had not been throttled closed far enough to have impacted any safety function. Enforcement for this finding resulted in a non-cited violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings.” (Section 1R16)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance was self-revealed when two technicians logged onto a general area Radiation Work Permit (RWP), entered the 1B Residual Heat Removal (RHR) Room, a posted high radiation area (HRA), and one of their electronic dosimeters alarmed.

  The cause of this event was failure to follow procedures. The finding was more than minor as it could be reasonably viewed as a precursor to a more significant event. The finding was of very low safety significance because the personnel were using electronic dosimeters that alarm to warn the workers of higher than expected dose rates or accumulated dose. The issue was a non-cited violation of Technical Specifications 5.7.1b and e., which required that an appropriate RWP be utilized by workers and a pre-job brief be provided prior to entry into a HRA. (Section 2OS1.6(2))

- Green. A finding of very low safety significance was self-revealed when a craft person, entered a posted HRA and highly contaminated area in the 1B Heater Bay without a HRA brief. This occurrence resulted in the person becoming contaminated and it was detected when the person exited the Radiologically Controlled Area (RCA).

  The cause of this event was failure to follow procedure. The finding was more than minor as it could be reasonably viewed as a precursor to a more significant event. The finding was of very low safety significance because the individual was using electronic dosimeters that alarm to warn the workers of higher than expected dose rates or accumulated dose. The issue was a non-cited violation of Technical Specifications 5.7.1b and e., which required that a pre-job brief be provided prior to entry into a HRA. (Section 2OS1.6(3))

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 20. The licensee failed to adequately evaluate the radiological hazards associated with radiation dose rates at a temporary walkway outside the radiologically controlled area in the turbine building.

  This finding was greater than minor because it had the potential to be more significant due to the location, adjacent to the main turbine bioshield during operation. The finding was of very low safety significance because no personnel had used the walkway. The issue was a non-cited violation of 10 CFR 20.1501(a). (Section 2OS1.7)

B. <u>**Licensee-Identified Violation**</u>

Cornerstone: Mitigating Systems

A violation of very low safety significance that was identified by the licensee has been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. The violation is discussed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period in end-of-cycle coastdown. The unit was shut down for scheduled refueling outage L1R10 on January 12, 2004. The unit was restarted on February 11, 2004, synchronized to the grid on February 12, 2004, and reached full power on February 15, 2004. On February 22, 2004, power was reduced to approximately 62 percent to permit a rod pattern adjustment. The unit was returned to full power on February 23, 2004. On March 7, 2004, power was reduced to approximately 66 percent to permit a rod pattern adjustment and the unit was returned to full power later that same day. The unit was operated at full power for the remainder of the inspection period.

Unit 2

The unit began the inspection period operating at full power. On February 29, 2004, power was reduced to approximately 74 percent to permit control rod scram time surveillance testing and scheduled maintenance on the feedwater heater and drain system. Testing and maintenance activities were completed and the unit was returned to full power later that same day. The unit operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial walkdowns of several equipment trains to verify operability and proper equipment lineup. These systems were selected based upon risk significance, plant configuration, system work or testing, or inoperable or degraded conditions. The following inspections performed represented two inspection samples:

- The 2B and 0 emergency diesel generators (EDGs) with the 2A EDG out-of-service for routine maintenance and testing;
- The Unit 2 Division 2 core standby cooling system (CSCS) with 1A EDG out-of-service for routine maintenance and testing.

The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors walked down the several risk significant areas looking for any fire protection issues. The inspectors selected areas containing systems, structures, or components that the licensee identified as important to reactor safety. The following twelve areas were selected as inspection samples:

- Fire zone 2D; Unit 1 reactor building - elevation 786'6"
- Fire zone 2E; Unit 1 reactor building - elevation 761'0"
- Fire zone 3E; Unit 2 reactor building - elevation 761'0"
- Fire zone 7B1; Unit 2 high pressure core spray (HPCS) diesel generator room - elevation 710'6"
- Fire zone 7B2; Unit 2 Division 2 diesel generator room - elevation 710'6"
- Fire zone 7B3; Unit 2 Division 1 diesel generator room - elevation 710'6"
- Fire zone 2G; Unit 1 reactor building - elevation 710'6"
- Fire zone 3G; Unit 1 reactor building - elevation 710'6"
- Fire zone 5C11; Turbine building ground floor general area - elevation 710'6"
- Fire zone 3D; Unit 2 reactor building - elevation 786'6"
- Fire zone 4D3; Unit 1 electrical equipment room - elevation 749'0"
- Fire zone 5A4; Turbine building cable zone - elevation 749'0"

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, barriers to fire propagation, and any compensatory measures the licensee had enacted due to degraded fire protection features.

b. Findings

An Unresolved Item (URI) was opened to track the NRC’s assessment of procedural noncompliance associated with compensatory fire watch patrols performed by the licensee to fulfill Technical Requirements Manual (TRM) action statements.

During routine quarterly fire protection inspections from February 5, 2004, through February 23, 2004, inspectors noted that logs documenting the performance of several licensee fire watch patrols recorded the performance of those patrols with unusual precision. Specifically, the inspectors noted that the times logged for the performance of each patrol were exactly 1 hour apart, with little or no variation seen over a period of days. It was further noted that these logs were not routinely carried by the fire watch during the performance of their rounds, thus requiring the logs to be filled in at the completion of each patrol. Direct observations by the inspectors identified that several fire watch patrols were logged at times the fire watch was not physically in the plant, contrary to licensee procedures that required the fire watch patrol logs to reflect the actual time of each patrol.

Direct observations by the inspectors also identified that fire watch patrols for some TRM fire door impairments were not routinely conducted by patrolling the fire zones on both sides of the affected barrier, as specified by written instructions for the impairment.

The inspectors identified that certain fire watch patrols being performed by the licensee were, at times, merely a simple visual observation of the impaired doorway area only. Reviews of completed fire watch patrol logs and interviews with licensee personnel and contractors indicated that these practices have been ongoing for many months, despite multiple opportunities by licensee management to have identified the issues and taken corrective action.

The inspectors determined that the issues with compensatory fire watch patrols may constitute licensee performance deficiencies and violations of regulatory requirements which require additional evaluation. As a result, the issue is considered unresolved pending further NRC investigation and review. (URI 05000373/05000374/ 2004002-01)

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee's testing of the 1A residual heat removal (RHR) heat exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing criteria.

The inspectors' review of the 1A RHR heat exchanger testing represented a single inspection sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (71111.08)

a. Inspection Scope

The inspectors reviewed the licensee's implementation of the inservice inspection program for monitoring degradation of the reactor coolant system boundary and the risk significant piping system boundaries.

Specifically, the inspectors conducted onsite and/or record reviews of the following three nondestructive examination activities to evaluate compliance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and to verify that indications and defects were dispositioned in accordance with the ASME Code. These reviews constituted two inspection samples:

- Ultrasonic examination of reactor core isolation cooling system elbow to pipe weld IRI-1002-16
- Ultrasonic examination of feedwater system pipe to elbow weld 1FW-1001-68
- Ultrasonic examination of residual heat removal system pipe to elbow weld 1RH-1004-24

The inspectors also reviewed the following two examinations from the previous outage with recordable indications that had been accepted by the licensee for continued service to verify that the licensee's acceptance for continued service was in accordance with the ASME Code. These reviews constituted one inspection sample:

- Recordable indication over 20 percent distance amplitude correction (DAC) found during ultrasonic examination of reactor pressure vessel feedwater line (LCS-1-N4E)
- Root geometry found during ultrasonic examination of feedwater line weld (FW-1002-19)

The inspectors reviewed the following two pressure boundary welds for Class 1 or 2 systems which were completed since the beginning of the previous refueling outage, to verify that the welding acceptance (e.g., radiography) and preservice examinations were performed in accordance with ASME Code requirements. These reviews constituted one inspection sample:

- Radiography of feedwater line 1FW02FA 24-inch weld FW-01 (Check Valve 1B21-F010-A)
- Radiography of feedwater line 1FW02FB 24-inch weld FW-01 (Check Valve 1B21-F010-B)

The inspectors reviewed one ASME Section XI Code repair or replacement to verify the repair and replacement met ASME Code requirements. This review constituted one inspection sample:

- Reactor core isolation cooling Class 1 repair and replacement of pipe with gate valve, and repair of pipe and disc (Job # 990174553)

The inspectors reviewed a sample of inservice inspection related problems documented in the licensee's corrective action program to assess conformance with 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requirements. In addition, the inspectors verified that the licensee correctly assessed operating experience for applicability to the ISI group.

b. <u>Findings</u>

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors conducted a quarterly observation of a training crew during an evaluated simulator scenario and reviewed licensed operator performance in mitigating the consequences of events. The scenario included a reactor scram with the failure of several control rods to properly insert, as well as a steam line fault that threatened primary containment integrity. The scenario also resulted in an emergency plan declaration of a site area emergency.

Areas observed by the inspectors included: clarity and formality of communications, timeliness of actions, prioritization of activities, procedural adequacy and implementation, control board manipulations, managerial oversight, emergency plan execution, and group dynamics. In addition, the inspectors also observed the licensee's post-scenario instructor/evaluator crew performance critique.

This training observation constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's handling of performance issues and the associated implementation of the Maintenance Rule (10 CFR 50.65) to evaluate maintenance effectiveness for the selected system. The following system was selected based on being designated as risk significant under the Maintenance Rule, being in the increased monitoring (Maintenance Rule category a(1)) group, or due to an inspector identified issue or problem that potentially impacted system work practices, reliability, or common cause failures:

- Maintenance issues with the 2B reactor protection system (RPS) motor-generator (MG) set

The inspectors review included verification of the licensee's categorization of specific issues including evaluation of the performance criteria, appropriate work practices, identification of common cause errors, extent of condition, and trending of key parameters. Additionally, the inspectors reviewed the licensee's implementation of the Maintenance Rule requirements, including a review of scoping, goal-setting, performance monitoring, short-term and long-term corrective actions, functional failure determinations associated with the condition reports (CRs) reviewed, and current equipment performance status.

The inspectors' review of this issue constituted a single inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

For each inspection sample, the inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The following four inspection samples were included:

- Unit 2 instrument nitrogen (IN) system failures and performance issues
- Unit 1 reactor recirculation system jet pump repairs
- Unit 1 channel A reactor vessel level 8 spurious trip signals troubleshooting and repair
- Unit 1 Division 1 RHR service water heat exchanger relief valve problems and troubleshooting

The inspectors also reviewed the licensee's evaluation of plant risk, risk management, scheduling, and configuration control for these activities in coordination with other scheduled risk significant work. The inspectors verified that the licensee's control of activities considered assessment of baseline and cumulative risk, management of plant configuration, control of maintenance, and external impacts on risk. In-plant activities were reviewed to ensure that the risk assessment of maintenance or emergent work was complete and adequate, and that the assessment included an evaluation of external factors. Additionally, the inspectors verified that the licensee entered the appropriate risk category for the evolutions.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Plant Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors monitored the licensee's response to a trip of the 2B RPS MG output breaker on February 1, 2004. The breaker trip caused the loss of the Unit 2 'B' RPS bus, which resulted in a half scram on Unit 2 and numerous primary containment isolation system actuations. The inspectors verified that operator responses to the event, as well as subsequent recovery actions, were in accordance with approved plant procedures. Additionally, the inspectors reviewed the licensee's notifications made pursuant to 10 CFR 50.72, and changes to the station's on-line risk profile that resulted from the event.

The inspectors' review of this event constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of several operability evaluations to determine the impact on Technical Specifications, the significance of the evaluations, and to ensure that adequate justifications were documented. The following five inspection samples were reviewed:

- Unit 1 drywell bulk average temperature measurement
- Unit 2 IN system bottle bank gas pressure regulator
- Unit 1 IN system bottle bank gas pressure regulator
- Unit 1 reactor recirculation jet pump RS-9 welds
- Unit 1 and Unit 2 reactor water cleanup differential flow isolation circuit wires

Operability evaluations were selected based upon the relationship of the safety-related system, structure, or component to risk.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed an Operator Workaround (OWA) involving the frequent replacement of nitrogen bottles to support IN system automatic depressurization system (ADS) safety-relief valve (SRV) operability. The inspectors reviewed the workaround's potential to impact the operators' ability to utilize the ADS system during the coping period for a station blackout, and for long term decay heat removal in a post-accident environment.

This review constituted a single inspection sample.

b. Findings

Introduction: A Green finding and associated NCV were identified by inspectors for the licensee's failure to provide adequate written instructions, as described in 10 CFR 50, Appendix B, Criterion V, for throttling activities associated with the 1IN090, "Gas Manifold System 1IN09MB Pressure Regulator Downstream Stop Valve."

Description: On February 11, 2004, LaSalle Unit 1 restarted from refuel outage L1R10. During this time, Operating Surveillance LOS-IN-R3, "Drywell Pneumatics Bottle Bank Regulator Adjustment and Emergency Pressurization Station Check Valve Exercise," was also in progress as a post-maintenance test for Unit 1 N_2 pressure regulators 1IN035 and 1IN038 following rebuild. Step D.1.1.2 of LOS-IN-R3 cautioned personnel performing the procedure that closure of the 1IN090 valve would render automatic depressurization system (ADS) safety-relief valves (SRVs) ‘D’, ‘S’, and ‘V’ inoperable from their backup compressed gas supply source.

On that same day, the Unit 1 North N_2 header pressure, an input to the instrument nitrogen (IN) system trouble alarm, annunciated on high pressure. In response to the alarm, operations personnel throttled closed 1IN090 as directed by the alarm response procedure, LOR-1PM13J-B404, “Instrument Nitrogen System Trouble,” Step B.4.d. No guidance was provided in LOR-1PM13J-B404 as to how far closed 1IN090 could be throttled before impacting ADS SRV backup gas supply operability.

From February 11 through February 21, 2004, in response to a malfunctioning 1IN038, “North N_2 Header Pressure Regulator Valve,” operators continued to throttle 1IN090 to maintain ADS header pressure between 150 psig and 190 psig. These actions were necessary to prevent the downstream North N_2 header relief valve from lifting at 210 psig. Based on interviews with the operators, the inspectors determined that at least once during the period, valve 1IN090 was throttled “almost fully closed.” On February 20, 2004, a non-licensed operator (NLO) questioned an operations field supervisor about IN system operability with the 1IN090 valve throttled almost closed. Following discussions between engineering and operations, the licensee returned valve 1IN090 to its full open position on February 21, and implemented a continuous vent on the North N2 header to maintain system pressure within the required range.

Subsequent to the licensee restoring 1IN090 to the fully open position, inspectors questioned licensee personnel about the previous use of the “almost closed” 1IN090 valve to maintain North N_2 header pressure. An engineering evaluation performed in response to the inspectors’ questions concluded that, based on how far throttled closed operations personnel remembered the 1IN090 valve being, ADS SRV backup compressed gas system operability had not been impaired. The licensee documented this evaluation in their corrective action program as CR 204354.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to determine how far closed 1IN090 could have been without impacting ADS SRV operability constituted a licensee performance deficiency. Using IMC 0612, Appendix B, “Issue Screening,” the inspectors determined that the issue constituted a finding of more than minor significance in that had it been left uncorrected it would have become a more significant safety concern. Specifically, had the licensee continued to throttle closed 1IN090 in accordance with LOR-1PM13J-B404, ADS SRV backup compressed gas supply operability would have eventually been lost at some point prior to the valve being fully closed when the gas flow path had become sufficiently restricted.

The inspectors analyzed the finding using the Phase 1 SDP for Mitigating Systems in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.” Because the finding did not result in an actual loss of any safety function for any system, train, or component, and because it did not screen as potentially risk significant due to a seismic, fire, flooding, or a severe weather initiating event, the inspectors determined the finding to be of very low safety significance (Green) and within the licensee's response band.

Enforcement: 10 CFR 50, Appendix B, Criterion V states: "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished."

Contrary to this requirement, the licensee's instructions and procedures relating to the operation of the 1IN090 valve, a component subject to the requirements of 10 CFR 50, Appendix B, per the licensee's UFSAR, failed to include appropriate quantitative or qualitative acceptance criteria for determining the extent to which the 1IN090 valve could be throttled closed without impacting ADS SRV backup compressed gas system operability. Because the finding associated with this violation is of very low safety significance and has been entered in the licensee's corrective action program (CRs 204354 and 206183), this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2004002-02)

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed two permanent plant modifications to verify that the design basis, licensing basis, and performance capability of risk significant systems were not degraded by the installation of the modifications. The inspectors also verified that the modifications did not place the plant in an unsafe configuration.

- Unit 1 Division 2 residual heat removal service water keep fill elimination project (EC 341950)
- Unit 1 drywell fillup rate conductor termination relocation (EC 340547)

The inspectors considered the design adequacy of each modification by performing a review, or partial review, of the modification's impact on plant electrical requirements, material requirements and replacement components, response time, control signals, equipment protection, operation, failure modes, and other related process requirements.

These reviews constituted two inspection samples.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected several post-maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact overall plant risk. The following reviews represented six inspection samples:

- Unit 1 drywell floor drain sump fill-up rate functional test and calibration after drywell penetration modification
- Unit 1 Division 2 residual heat removal service water leak check and flow test after keep-fill system elimination modification
- 2B reactor protection system motor generator set operational testing after voltage regulator replacement work
- Unit 2 reactor core isolation cooling (RCIC) system operational testing after instrument and mechanical maintenance
- Unit 1 main steam line drain isolation valves radiography and local leak rate testing after code weld repairs
- Discharge capacity testing of the Unit 1 Division 1 125 volt direct current (Vdc) battery following replacement

The inspectors verified by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance activities. The inspectors' reviews included, but were not limited to, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use and compliance, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, system restoration, and evaluation of test data. Also, the inspectors verified that maintenance and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, Technical Specifications, and Updated Final Safety Analysis Report (UFSAR) design requirements.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated activities associated with a Unit 1 refueling outage (L1R10) that began on Tuesday, January 13, 2004, and ended on Thursday, February 12, 2004. Evaluated activities were assessed to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment and containment close-out activities, startup and heatup activities, and identification and resolution of problems associated with the outage.

All activities by inspectors associated with L1R10 under this inspection procedure constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors selected various surveillance test activities for review. Activities were selected based upon risk significance and the potential risk impact from an unidentified deficiency or performance degradation that a system, structure, or component could impose on the unit if the condition were left unresolved. Review of the following surveillances constituted five inspection samples:

- Unit 1 Division 3 response time test
- Unit 1 main steam drain line isolation valves 1B21-F016 and 1B21-F019 local leak rate test
- Unit 1 reactor core isolation cooling pump operability test
- 1A emergency diesel generator idle start
- Unit 2 Division 2 residual heat removal pump quarterly operability test

The inspectors observed the performance of surveillance testing activities, including reviews for preconditioning, integration of testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, control of temporary modifications or jumpers required for test performance, documentation of test data, Technical Specification applicability, impact of testing relative to performance indicator reporting, and evaluation of test data.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

Review of the following temporary modifications constituted three inspection samples:

- Lifted shield leads on Unit 2 drywell floor drain sump drywell penetration cable to suppress noise induction (TCCP 346343)
- Temporary power to the 0 EDG H-1 immersion heater, B-7 oil circulating pump, and B-7A engine lube oil soak back pump (TCCP 346749)

- Temporary alternate nitrogen gas supply system to supplement the Unit 1 automatic depressurization system north bottle bank (TCCP 347585)

The inspectors reviewed the safety screening, design documents, UFSAR, and applicable Technical Specifications to determine that the temporary modification was consistent with modification documents, drawings, and procedures. The inspectors also reviewed the post-installation test results to confirm that tests were satisfactory and that the actual impact of the temporary modification on the permanent system and interfacing systems were adequately verified.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's records to determine if any occupational exposure control cornerstone performance indicators (PIs) had been identified during the previous five calender quarters. If PIs had been identified, the inspectors determined whether or not the conditions surrounding the PIs had been evaluated and identified problems had been entered into the corrective action program for resolution. This review represented one sample.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit (RWP) Reviews

a. Inspection Scope

The inspectors reviewed three exposure significant work areas within radiation areas, high radiation areas (<1R/hr), and reviewed associated licensee controls and surveys of these areas to determine if controls were acceptable. The three areas were the Low Pressure Heater Bay, Turbine Building Turbine Deck, and the Drywell.

With a survey instrument, the inspectors walked down these areas to determine whether prescribed RWP, procedure, and engineering controls were in place and licensee survey and posting were complete and accurate, and air samplers, if needed, properly located.

The inspectors reviewed radiation work permits used to access these and other high radiation areas to identify what work control instructions or control barriers were specified. The inspectors used plant Technical Specification HRA requirements as the standard for necessary barriers. The inspectors reviewed the electronic personnel dosemeter (EPD) alarm set points for conformity with survey indications and plant policy and personnel required response when the EPD malfunctions or alarms.

The inspectors reviewed records to determine if airborne radioactivity areas with the potential for individual worker internal exposures of >50 millirem committed effective dose equivalent (CEDE) had been identified within the facility. Work areas having a history of, or the potential for, airborne transuranics were also evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

The inspectors reviewed the adequacy of the licensee's internal dose assessment process for internal exposures > 50 millirem CEDE. This review represented five samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, Licensee Event Reports and Special Reports relating to the access control program since the last inspection to determine if identified problems are entered into the corrective action program for resolution.

The inspectors reviewed approximately 10 corrective action reports related to access controls, including three HRA radiological incidents (non-performance indicators (PI) occurrences identified by the licensee in high radiation areas <1R/hr). Radiation protection (RP) staff were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking
- Disposition of operability/reportability issues
- Evaluation of safety significance/risk and priority for resolution
- Identification of repetitive problems
- Identification of contributing causes
- Identification and implementation of corrective actions

The inspectors evaluated the licensee's process for problem identification, characterization, prioritization, and verified that problems were entered into the corrective action program and resolved.

For repetitive deficiencies, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates >25 R/hr at 30 centimeters or >500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures >100 millirem total effective dose equivalent (or >5 rem shallow dose equivalent or >1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure. This review represented three samples.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews and Review of Work Practices in Radiologically Significant Areas

a. Inspection Scope

The inspectors observed feed water check valve welding, radiography of the check valve in the drywell and RHR service Water Keep Fill Modification work. The inspectors reviewed radiological job requirements for this activity including the RWP requirements and those provided in the As-Low-As-Reasonably-Achievable (ALARA) plan, and the associated Total Effective Dose Equivalent (TEDE) ALARA evaluation. Additionally, the inspectors attended an ALARA pre-job briefing for work in the Low Pressure Heater Bay and Drywell to assess the adequacy of the information exchanged.

Job performance was observed to verify that radiological conditions in the work areas were adequately communicated to workers through the pre-job brief and postings. The inspectors also verified the adequacy of radiological controls provided by the RP staff including the radiological surveys and RP technician job coverage which consisted of continuous visual surveillance as a filter was removed from its housing and transported to another area of the plant.

The inspectors also reviewed the licensee's procedure and practices for dosimetry placement and use of multiple dosimetry and for extremity monitoring for work in high radiation areas. Specifically, the inspectors reviewed dosimetry use during underwater diving to repair cracks on the steam dryer, a job having significant dose gradients, for compliance with the requirements. This review represented three inspection samples.

b. Findings

No findings of significance were identified.

.5 High Risk Significant, High Dose Rate HRA, and Very High Radiation Area (VHRA) Controls

a. Inspection Scope

The inspectors held discussions with the Radiation Protection Manager concerning high dose rate/high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to verify that any procedure modifications did not substantially reduce the effectiveness and level of worker protection.

The inspectors discussed with a RP supervisor the controls that were in place for special areas that had the potential to become very high radiation areas during certain plant operations to determine if these plant operations required communication beforehand with the RP group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to numerous Locked High Radiation Areas (LHRA) in the Turbine and Reactor Buildings. This review represented three samples.

b. Findings

No findings of significance were identified

.6 Radiation Worker Performance

a. Inspection Scope

During job observations, the inspectors reviewed radiation worker performance with respect to stated RP work requirements to determine if they were aware of the significant radiological conditions in their workplace and the RWP controls and limits in place.

The inspectors reviewed ten radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause, and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. This review represented two inspection samples.

b. Findings

(1) Contract Personnel Entry Into the 1A Turbine-Driven Reactor Feed Pump Room HRA on December 30, 2003

An Unresolved Item (URI) was opened to track the NRC’s evaluation of a contractor supervisor who directed contractor craft personnel to enter a HRA boundary without craft personnel having received an HRA briefing.

On December 30, 2003, three contractor craft personnel entered a posted HRA in the 1A turbine-driven reactor feed pump (TDRFP) room to build scaffolding without receiving the required HRA briefing. The personnel had been briefed by RP technicians for work in the non-HRA radiation portion of the 1A TDRFP room. However, several days earlier, the supervisor walked down the job site and recognized that the scaffolding would also be built in the HRA portion of the 1A TDRFP room. Apparently, the supervisor had not informed the RP technicians that portions of the work would be done in a HRA and, despite the fact the RP briefing only addressed work in a non-HRA, the supervisor directed the three craft personnel to enter and work in the HRA. In addition, the supervisor continued to direct the craft personnel to enter the HRA even after one of the craft personnel questioned this direction.

The HRA was not equipped with any type of entry gate to facilitate personnel access, as RP technicians were unaware that personnel would be working in the area. In order to gain entry to the HRA, the foreman instructed the three personnel to either "duck under" or move the HRA barrier rope that was in place.

After approximately 15 minutes in the HRA, the three craft personnel noted that their electronic dosimeters were recording substantially more dose than the 4-6 mrem the RP technician had indicated would be their expected dose for the job. A craft person discussed the situation with a second contractor supervisor while the supervisor passed through the room. This supervisor discussed the situation with the RP technicians stationed at the RP desk which was not located at or near the work site. Subsequently, all work was halted, the craft personnel exited the HRA, and the licensee initiated a prompt investigation into the issue.

This issue may indicate a performance deficiency resulting in craft personnel unintended dose and requires addition NRC review. Pending further NRC review of the circumstances surrounding this occurrence, this issue is identified as unresolved. (URI 05000373/2004002-03)

(2) <u>Contract Personnel Entry Into the 1B RHR Room HRA on January 20, 2004</u>

<u>Introduction</u>: A Green self-revealing finding and associated NCV were identified when two technicians who were logged onto a general area RWP entered the 1B RHR Room, a posted HRA, contrary to the licensee's Technical Specifications.

<u>Description</u>: On January 20, 2004, two technicians returning from a job in the plant decided to enter the 1B RHR Room to walkdown another job coming up in the next several days. One of the individual's EPDs alarmed at a rate above the 50 mrem/hr set point, but this apparently went unnoticed by the worker. The error was detected when the individual later logged out of his RWP and received an "ERROR - CONTACT HP" message due to his EPD recording a rate above its alarm set point. The entry into a HRA is contrary to Technical Specification 5.7.1b, requiring that an appropriate RWP be utilized by workers, and Technical Specification 5.7.1e requiring a pre-job brief be provided prior to entry into a HRA.

The individual received a total dose of 8 millirem, and the maximum dose rate measured by the EPD was 50.5 millirem/hour.

The licensee's investigation determined the cause to be a failure of human performance error prevention techniques. Specifically, the technicians misunderstood the RWP requirements, lacked a questioning attitude, lacked self-checking and peer checking in making the decision to enter a marked and gated HRA, and lacked the use of proper radiation safety practices. Both individuals were locked out of the stations radiologically controlled area (RCA) and the licensee initiated a prompt investigation. Additionally, all site personnel were notified of this event through a station safety alert.

Analysis: The performance deficiency associated with this event was failure to follow procedure. The finding, which is under the Occupational Radiation Safety Cornerstone, does not involve the application of traditional enforcement because it did not result in actual safety consequences or potential to impact the NRC's regulatory function and was not the result of any willful actions. The finding was more than minor as it could be reasonably viewed as a precursor to a more significant event.

Enforcement: The licensee's Technical Specification 5.7.1b requires that an appropriate RWP be utilized by radiation workers and Technical Specification 5.7.1e requires a pre-job brief be provided prior to entry into a HRA. Contrary to the above, on January 20, 2004, radiation workers entered a HRA inside the 1B RHR Room, failed to sign on to a RWP that authorized entry into the HRA, and did not receive a briefing prior to entry into the HRA. Because entry into the RCA was properly conducted under a general entry RWP, the entry into the HRA was monitored by EPDs. Therefore, the event is of low safety significance and the finding is within the licensee response band. The licensee had entered the issue into their corrective action system as Condition Report (CR) 196455. The associated violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2004002-04)

(3) Licensee Craft Personnel Entry Into a Unit 1 Heater Bay HRA on January 21, 2004

Introduction: A Green self-revealing finding and associated NCV were identified when a craft person, logged onto a Heater Bay RWP, entered a posted HRA in the Unit 1 heater bay without a HRA brief, contrary to the licensee's Technical Specifications.

Description: On January 21, 2004, a craft person, on loan from another Exelon station was working on an elevated platform in the Unit 1 heater bay. The actual valve the mechanic needed to access was located on another platform in the heater bay approximately 12 to 15 feet away. Rather than descend the ladder from the current platform and climb back up on the other platform's ladder, the mechanic exited his platform by crawling through the platform's guard railing, across some piping and through another guard railing to the second platform. The second platform, a HRA, was posted at the base of it's ladder, it's sole point of normal access, as a contaminated high radiation area. The finding was self-revealing when the mechanic, who exited the second platform via the same path that he used to get there and had no knowledge that he ever was in a contaminated HRA, alarmed the RCA personnel contamination monitors when he attempted to leave the RCA. The entry into a HRA is contrary to Technical Specification 5.7.1e requiring a pre-job brief be provided prior to entry into a HRA.

Analysis: The performance deficiency associated with this event was failure to follow procedure. The finding, which is under the Occupational Radiation Safety Cornerstone, does not involve the application of traditional enforcement because it did not result in actual safety consequences or potential to impact the NRC's regulatory function and was not the result of any willful actions. The finding was greater than minor as it could be reasonably viewed as a precursor to a more significant event.

Enforcement: The licensee's Technical Specification 5.7.1e requires a pre-job brief be provided prior to entry into an HRA. Contrary to the above, on January 21, 2004, a radiation worker entered an HRA inside the Unit1 heater bay and did not receive a briefing prior to entry into that HRA. Because entry into the heater bay was properly conducted under the heater bay RWP and the entry into the HRA was monitored by an EPD, the event is of low safety significance and the finding is within the licensee response band. The licensee had entered the issue into their corrective action system as CR 196819. The associated violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2004002-05)

(4) Contract Personnel Entry Into a Unit 1 694' Reactor Building Raceway HRA on January 25, 2004

A URI was opened to track the NRC's evaluation of a contractor supervisor who lead contractor craft personnel into a HRA before craft personnel had received a HRA briefing.

On January 25, 2004, three craft personnel and their supervisor entered a High Radiation Area without the proper brief and in violation of proper postings in the Unit 1 694 foot reactor building raceway. The foreman directed the workers to assemble the needed tools and sign on to a general area RWP. The workers logged on to a non-outage RWP for all building minor maintenance. The foreman signed on to a RWP for the outage that did not allow HRA entries. No pre-job brief was conducted. The foreman thought that no pre-job brief was required for set-up activities when there was no work involved. The foreman had entered that area earlier in the day to locate a valve work area. All individuals failed to review survey maps. Subsequent to the entry, a craft Superintendent met the group in the HRA and did not know that they were not on the correct RWP. This fact was later identified when EPD dose rate alarms sounded on two craft personnel. While they directed the Superintendent to leave immediately and notify RP, they remained in the area for approximately 8-10 minutes. The highest dose received by any of the craft personnel was 5 millirem. Upon identification of the occurrence, the licensee initiated a prompt investigation into the issue.

The entry into a HRA is contrary to Technical Specification 5.7.1b, requiring that an appropriate RWP be utilized by workers, and Technical Specification 5.7.1e requiring a pre-job brief be provided prior to entry into a HRA requires additional NRC review. Pending further NRC review of the circumstances surrounding this occurrence, this issue is identified as unresolved. (URI 05000373/2004002-06)

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

During job observations and general plant walkdowns, the inspectors evaluated RP technician performance with respect to RP work requirements to determine whether they were aware of the radiological conditions in their workplace, the RWP controls and limits in place, and if their performance was consistent with the radiological hazards that existed.

The inspectors reviewed several radiological problem reports generated in the fourth quarter of 2003 through this inspection date in 2004, with the cause of the event attributed to RP technician error, to determine if trends were traceable to similar causes. These reviews represented two inspection samples.

b. Findings

Introduction: A Green finding and associated NCV were identified by inspectors for failure to adequately evaluate the radiological hazards associated with measurements of radiation dose rates at a temporary walkway outside the radiologically controlled area, as described in 10 CFR 20.1501(a).

Description: During a routine tour of the power block on January 7, 2004, resident inspectors identified an elevated platform set up in a radiologically uncontrolled area adjacent to the Unit 1 main turbine biological shield wall. The platform was set up by the licensee as part of their preparations for an upcoming Unit 1 refuel outage, with its purpose being to provide an uncontaminated bridge over a contaminated pathway on the turbine floor that allowed personnel to access the control room without entering the RCA. Under outage conditions with the main turbine secured, dose rates on the platform, which had been used by the licensee during past Unit 1 refuel outages, are historically less than 1 mrem/hour. During the platform placement, the RP technician covering the activity misread the meter settings on the survey instrument and recorded the dose rate as less than 1 mrem/hour. Shortly after being put in place and surveyed by the RP technicians, but before being put into routine use by the licensee, the inspectors questioned the lack of radiation area signs on the platform. The licensee investigated the inspectors' concerns and subsequent radiological surveys revealed a dose rate of 7-8 mrem/hour on the platform. The licensee then posted the platform as a radiation area, as required by 10 CFR 20.1902(a), "Posting of Radiation Areas."

Analysis: The inspectors determined that the licensee's failure to adequately evaluate the radiological hazards associated with radiation dose rates at a temporary walkway outside the RCA is a performance deficiency because the licensee is expected to meet the requirements of 10 CFR 20.1501(a). Traditional enforcement does not apply because the issue did not have any actual safety consequences, potential for impact on the NRC's regulatory function, and was not the result of any willful violation of NRC requirements or the licensee's procedure. This finding is greater than minor because it is associated with program and process attributes and affected the objective of the Radiation Safety Cornerstone to protect workers from exposure to radiation. In addition, if left uncorrected, this finding could result in more significant safety concern (i.e., actual unintended and unmonitored exposure outside the RCA). This finding was evaluated using the Occupational Radiation Safety SDP and was preliminarily determined to be a finding of low to moderate safety significance. The Occupational Radiation Safety SDP defines a compromised ability to assess dose, for external dose, as 100 mrem whole body from external exposure, per individual and an individual or isolated failure to survey, or monitor, does not constitute a compromised ability to assess dose. However, each event should be considered as a failure of a radiation safety barrier.

Enforcement: 10 CFR 20.1501(a) states, in part, "Each licensee shall make or cause to be made surveys that: (1) may be necessary for the licensee to comply with the regulations in this part; and (2) are reasonable under the circumstance to evaluate the magnitude and extent of radiation levels and the potential radiological hazards." Contrary to the above, the licensee's failure to effectively evaluate the radiological hazard presented by elevating a walkway adjacent to the turbine bioshield could have resulted in an actual unintended and unmonitored exposure outside the RCA. Because the failure to effectively evaluate the radiological hazard is of very low safety significance (i.e., it could not have reasonably exceeded 100 mrem wholebody dose) and has been entered in the licensee's corrective action program (CR 194434), this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000373/2004002-07)

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

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends, ongoing and planned activities in order to assess current performance, and exposure challenges. This included determining the plant's current 3-year rolling average for collective exposure in order to help establish resource allocations and to provide a perspective of significance for any resulting inspection finding assessment. The inspectors reviewed the outage work scheduled during the inspection period and associated work activity exposure estimates for the following work activities which were likely to result in the highest personnel collective exposures:

- Unit 1 drywell safety relief valve (SRV) activities
- Unit 1 drywell control rod drive (CRD) pull/puts and support activities
- Replacement of 1B21-F010A/10B feed water check valves
- 1B RHR service water keep fill modification
- L1R10 drywell snubber activities

The inspectors evaluated site specific trends in collective exposures and source-term measurements. The inspectors reviewed procedures associated with maintaining occupational exposures, ALARA, and processes used to estimate and track work activity specific exposures. This review represented four samples.

b. Findings

No findings of significance were identified.

.2 Radiological Work Planning.

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following work activities of highest exposure significance during the Unit 1 refueling outage:

- Unit 1 drywell safety relief valve (SRV) activities
- Unit 1 drywell control rod drive (CRD) pull/puts and support activities
- Replacement of 1B21-F010A/10B feed water check valves
- 1B RHR service water keep fill modification
- L1R10 drywell snubber activities

For these five activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to verify that the licensee had established procedures and engineering and work controls that were based on sound radiation protection principles in order to achieve occupational exposures that were ALARA. This also involved determining that the licensee had reasonably grouped the radiological work into work activities based on historical precedence, industry norms, and/or special circumstances.

The inspectors compared the results achieved including dose rate reductions and person-rem used with the intended dose established in the licensee's ALARA planning for these five work activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed. This review represented three samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The inspectors reviewed the assumptions and bases for the current annual collective exposure estimate including procedures, in order to evaluate the licensee's methodology for estimating work activity-specific exposures and the intended dose outcome. Dose rate and man-hour estimates were evaluated for reasonable accuracy.

The licensee's process for adjusting exposure estimates or re-planning work when unexpected changes in scope, emergent work, or higher than anticipated radiation levels were encountered was evaluated. This included determining that adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process. This review represented two samples.

b. Findings

No findings of significance were identified.

.4 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors observed the following five jobs that were being performed in radiation areas, airborne radioactivity areas, or high radiation areas for observation of work activities that presented the greatest radiological risk to workers:

- Unit 1 drywell safety relief valve (SRV) activities
- Unit 1 drywell control rod drive (CRD) pull/puts and support activities
- Replacement of 1B21-F010A/10B feed water check valves
- 1B RHR service water keep fill modification
- L1R10 drywell snubber activities

The licensee's use of ALARA controls for these work activities was evaluated using the following:

The licensee's use of engineering controls to achieve dose reductions was evaluated to verify that procedures and controls were consistent with the licensee's ALARA reviews, that sufficient shielding of radiation sources was provided for, and that the dose expended to install/remove the shielding did not exceed the dose reduction benefits afforded by the shielding.

This review represented one sample.

b. Findings

No findings of significance were identified.

.5 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and determined that the licensee was making allowances and had developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry. This review represented one sample.

b. Findings

No findings of significance were identified.

.6 Radiation Worker Performance

a. Inspection Scope

Radiation worker and Radiation Protection Technician performance was observed during work activities being performed in radiation areas, airborne radioactivity areas, and high radiation areas that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas and that work activity controls were being complied with. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved. This review represented one sample.

b. Findings

No findings of significance were identified.

.7 Declared Pregnant Workers.

a. Inspection Scope

The inspectors reviewed dose records of declared pregnant workers for the current assessment period to verify that the exposure results and monitoring controls employed by the licensee complied with the requirements of 10 CFR Part 20. This review represented one sample.

b. Findings

No findings of significance were identified.

.8 Problem Identification and Resolutions

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports related to the ALARA program since the last inspection to determine if the licensee's overall audit program's scope and frequency for all applicable areas under the Occupational Cornerstone met the requirements of 10 CFR 20.1101(c). The licensee's corrective action program was also reviewed to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed. This review represented two samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Initiating Events and Barrier Integrity

.1 Reactor Safety Performance Indicator Verification

a. Inspection Scope

The inspectors reviewed Licensee Event Reports (LERs), licensee data reported to the NRC, plant logs, and NRC inspection reports to verify the following performance indicators for the 1st quarter of 2004:

- Unplanned Power Changes per 7000 Critical Hours, Units 1 and 2
- Reactor Coolant System Specific Activity, Units 1 and 2

The inspectors verified that the licensee accurately reported performance as defined by the applicable revision of Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline."

The review of these performance indicators for both LaSalle units constituted four inspection samples.

b. Findings

No findings of significance were identified.

.2 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 1st quarter 2004 performance indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter (IMC) 0608, "Performance Indicator Program."

b. Findings

No findings of significance were identified.

The inspectors noted that the licensee's method for screening data for inclusion in the reactor coolant system specific activity performance indicator excluded data from samples taken during balance of plant transients such as trips of the hydrogen water chemistry system or isolation of the reactor water cleanup system. The licensee stated that this was due to their misunderstanding of the data reporting elements specified in Nuclear Energy Institute (NEI) 99-02, Revision 2, for this performance indicator. The licensee corrected this error for the current quarter's data prior to submission and performed an extent-of-condition review of data submitted for the past year.

The licensee's extent-of-condition review revealed that the inclusion of data for all qualifying samples per NEI 99-02, Revision 2, would increase the already submitted value for maximum monthly reactor coolant system activity for Unit 2 in August of 2003. The inspectors determined that this error constituted a minor issue, since the corrected value, while higher, did not result in the reactor coolant system specific activity performance indicator crossing a threshold value. The licensee entered this issue into their corrective action program as CR 210248.

4OA2 Identification and Resolution of Problems (71152)

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the routine baseline inspections performed during this period, the inspectors verified that the licensee entered the problems identified during the inspection into their corrective action program. Additionally, the inspectors verified that the licensee was identifying issues at an appropriate threshold and entering them in the corrective action program, and verified that problems included in the licensee's corrective action program were properly addressed for resolution. Attributes reviewed included: complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue.

b. Findings

No findings of significance were identified.

.2 Selected Issue Follow-up Inspection: 2B Reactor Protection System (RPS) Motor-Generator (MG) Set Troubleshooting and Repairs

a. Introduction

On January 31, 2004, the licensee identified a low voltage condition on the 2B RPS MG set during routine operator rounds. Follow on investigation by a component specialist did not identify any apparent problems with the MG set, and increased monitoring and trending of the machine's voltage was initiated. The licensee documented the condition and these actions in CR 198665. Additionally, maintenance work requests were initiated to have the voltage (2C71-R902B) and current (2C71-R901B) meters checked for accuracy.

On January 1, 2004, the 2B RPS MG set ‘D’ electrical power monitoring assembly (EPMA) output breaker tripped open resulting in the loss of the 2B RPS bus. The loss of the bus resulted in a half scram and associated primary containment isolation system (PCIS) group 1 (partial), 2, 3, 5, 6, 7 and 10 isolations. The instrument nitrogen (IN) and reactor water clean up (RWCU) systems were lost when their containment isolation valves closed on group 10 and group 5 isolation signals. The event required entry into multiple abnormal procedures and Technical Specification action statements, as well as a non-emergency 8-hour notification to the NRC Headquarters Operations Officer (HOO) in accordance with 10 CFR 50.72(b)(3)(iv)(A).

The licensee initiated troubleshooting with the 2B RPS MG left running in an unloaded condition. During this troubleshooting, output voltage was observed to drift between 115 and 120 Vac. The licensee determined that an electrical relay (2K) on the machine was faulty and the cause of the voltage issues. In addition to replacing the subject relay, the licensee also replaced the locally mounted MG set voltage and current indicating meters (2C71-R901B and 2C71-R902B), replaced the ‘D’ EPMA circuit board since it was coming due for routine replacement, and calibrated the ‘B’ and new ‘D’ EPMA circuit boards. The MG set was subsequently run for 24 hours under load for testing.

During the confidence load test following the troubleshooting and repair efforts, the 2B MG set and both EPMAs experienced another trip when technicians connected an oscilloscope across a discrete component in the voltage regulator circuit. When the MG set was re-started and re-loaded, voltage read almost 3 volts higher then before the trip. The licensee documented this condition in CR 208783. After reviewing the trip data, licensee personnel concluded that the new 2K relay had an oxide film on it that was burned off during the trip of this relay. This, they concluded, caused enough circuit resistance change to account for the higher output voltage. The trip of the unit was attributed to a static charge on the oscilloscope that, when connected, caused a spike in the feedback loop and ultimately drove output voltage up above its set point. Once the MG set tripped, the EPMA cards tripped three seconds later on under voltage as expected. After discussing this trip with a corporate voltage regulator expert, the technicians performed additional performance tests on the MG set in an attempt to duplicate the problem. However, the voltage regulator performed as designed, and on February 7, 2004, the 2B RPS MG set was returned to operational service.

A little over 2 days later, on February 10, 2004, operations personnel noted that the 2B RPS MG set output voltage was at approximately 118 Vac, vice the normal 120 Vac. This condition was documented by the licensee in CR 200667. The licensee decided at this point to place the 2B RPS bus on to its alternate voltage source to allow for additional troubleshooting of the MG set. During the subsequent troubleshooting, technicians noted that insertion of a screwdriver into the adjustment slot for the voltage regulator gain adjustment potentiometer caused large voltage spikes on the MG set generator output. After technicians performed this action a second time, the MG set and EPMA breakers tripped. The entire voltage regulator was replaced, and the remainder of the troubleshooting plan was completed with no other anomalies noted. The MG set was restarted and the voltage regulator calibrated while the MG set was on an artificial load bank. The MG set was then allowed to run for 3 days while connected to the load bank and special monitoring instrumentation. No abnormalities were noted and the 2B MG set was returned to normal operation on February 12, 2004.

On February 26, 2004, the 2B group RPS scram lights were observed to be "flickering" by control room operators. Further investigation by the licensee revealed that the 2B RPS Generator "bus alive" lights on panel 2H13-P610 and the DS30 and DS35 Group 6 isolation lights on panel 2H13-P609 were also flickering. However, no abnormal indications were reported from the 2B RPS MG set. The licensee documented these conditions with CR 204449.

Troubleshooting was again initiated by the licensee, and live circuit checks confirmed that the 2B MG set was causing bus voltage fluctuations. After conferring with the vendor and other sites, the licensee's troubleshooting team concluded that the recently replaced voltage regulator required minor adjustments for regulator stability. The MG set was removed from service and connected to a load bank. Minor adjustments to the MG voltage regulator were performed with the unit on a load bank, and the voltage regulator was successfully load tested for 12 hours before the MG set was returned to service.

b. <u>Effectiveness of Problem Identification</u>

(1) <u>Inspection Scope</u>

The inspectors selected the licensee's repeated maintenance efforts on the 2B RPS MG set for a more in-depth review. The inspectors considered the nature and significance of the issue with respect to safety, risk, and the licensee's corrective action procedural requirements. Attributes reviewed included complete and accurate identification of the problem, and that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent of condition reviews, and previous occurrences reviews were proper and adequate.

(2) <u>Issues</u>

During a period of approximately 4 weeks, the 2B RPS MG set tripped or had to be secured for troubleshooting and repair of output voltage issues on four separate occasions. Troubleshooting efforts following the first three of these instances failed to fully identify all issues associated with the MG set's faulty voltage regulator, and ultimately led to additional component unavailability. Initially, the licensee chose to only address the problem through "simple troubleshooting" under a work order. It was not until the MG set was removed from service on February 10, 2004, that the licensee began addressing the voltage problem via a "complex troubleshooting" effort, which consisted of an in-depth plan of attack for the problem. It was only after this complex troubleshooting plan was enacted that the licensee was able to identify issues with the MG set's voltage regulator.

c. <u>Effectiveness of Corrective Actions</u>

(1) <u>Inspection Scope</u>

The inspectors reviewed the licensee's repeated maintenance efforts on the 2B RPS MG set to determine if the corrective action program actions addressed generic implications, and also to determine if the focus and timeliness of corrective actions were sufficient to prevent recurrence.

(2) Issues

Following replacement of the 2B RPS MG set's voltage regulator on February 12, 2004, additional problems with voltage regulation were experienced. The licensee performed the initial voltage regulator calibration using instructions from the vendor's manual. However, vendor manual set up instructions are typically generic, and the details do not address plant-specific factors that may affect the initial setting. Historically, the licensee had relied upon a corporate voltage regulator expert or third party vendor to assist with the setup of RPS MG set voltage regulators. During this particular evolution, licensee technicians utilized the vendor manual information to set up the voltage regulator with out the usual external assistance, or even a component-specific procedure. As a result, fine tuning adjustments to the voltage regulator were not made. A component-specific procedure could have provided the necessary guidance for properly fine tuning the MG set voltage regulator, as was evidenced by the procedure site engineering was able to eventually provide that enabled the voltage regulator to be appropriately fine tuned.

4OA3 Event Follow-up (71153)

Cornerstones: Initiating Events and Mitigating Systems

.1 (Closed) Licensee Event Report 05000373/2003-004-00: High Pressure Core Spray Inoperable Due to Improperly Seated Fuse.

On November 17, 2003, during a routine Unit 1 high pressure core spray (HPCS) system surveillance, licensee personnel discovered that a fuse in the circuitry for the system's low level initiation/high level trip logic was not fully seated in its holder. Licensee engineering personnel subsequently evaluated the as-found condition of the fuse and determined that circuit continuity could not be assured during a seismic event.

The licensee's investigation into the issue determined that the most probable cause of the fuse not being fully seated was personnel error during the last time the fuse had been manipulated. The investigation determined that this last manipulation had been in March 2002, when Operations personnel restored the fuse to its holder upon restoration from a routine maintenance clearance order. The improperly inserted fuse went undetected until the November 17, 2003 surveillance because circuit continuity had been maintained by the fuse even though it was not fully seated. Had circuit continuity ever been lost during the period from March 2002 to November 2003, a main control room annunciator would have alarmed and alerted operators to the condition.

The inspectors reviewed the issue and the licensee's investigation and determined the matter to have been of very low safety significance. Although not fully qualified for seismic events, the actual low level initiation/high level trip function for HPCS was never lost, as the improperly seated fuse had maintained circuit continuity. Additionally, even had the low level initiation/high level trip logic function been lost due to a seismic or other such event, the HPCS system was always capable of being started and secured manually by control room operators, and procedures were in place to direct them to do so had it been necessary. As a result, no findings of significance were identified by the inspectors. The licensee had entered this event into their corrective action program under CR 186839, and corrective actions taken and planned by the licensee appeared to be adequate. The details of a licensee-identified violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," are discussed in Section 4OA7 of this report, and aspects associated with cross-cutting area of Human Performance are discussed in Section 4OA4.

.2 (Closed) Licensee Event Report 05000373/2003-005-00: Manual Reactor Scram on Low Reactor Vessel Water Level Due to Stuck Open Feedwater Pump Discharge Check Valve.

On November 27, 2003, while conducting a Unit 1 shutdown to perform various maintenance activities, the 1A turbine-driven reactor feed pump (TDRFP) discharge check valve stuck open during a feed pump transfer at approximately 20 percent power. This component fault caused a significant amount of feedwater flow to be diverted through the minimum flow line back to the main condenser instead of being sent to the reactor vessel. With a large difference existing between steam flow and feedwater flow, reactor vessel level began to decrease. At a reactor vessel level of 20 inches with a significant lowering trend established, operators followed existing procedural guidance and inserted a manual reactor scram.

Following the scram, all systems responded as expected and no emergency core cooling systems were actuated or challenged. Operators, per procedure, closed the 1A TDRFP discharge isolation valve during normal efforts to secure that pump. This effectively isolated the feedwater flow being diverted back to the condenser and allowed operators to restore reactor vessel level with the 1C motor-driven reactor feed pump (MDRFP).

The licensee initiated a root cause investigation into the cause of the scram and found that the design of the 1A TDRFP check valve rendered the valve susceptible to the type of failure observed. A licensee risk analysis of the event revealed that it was of very low safety significance, and bounded by the plant's existing analysis for a loss of reactor feedwater flow. The inspectors reviewed the issue and the licensee's investigation and determined the matter to have not involved any licensee performance deficiencies as defined in IMC 0612, Appendix B, Section 1. As a result, no findings of significance were identified by the inspectors. Similarly, the inspectors did not identify any violations of regulatory requirements associated with the event. The licensee had entered this event into their corrective action program under CR 188345, and corrective actions taken and planned by the licensee appeared to be adequate.

4OA4 Cross-Cutting Aspects of Findings

.1 Human Performance

Several of the findings described elsewhere in this report had as the majority of their causes various human performance deficiencies.

- A finding described in Section 2OS1.6(2) involved failures on the part of contractor personnel to follow established plant procedures and radiological practices with respect to HRAs during a pre-job walkdown in the 1B RHR Room.

- A finding described in Section 2OS1.6(3) also involved the failure of personnel to follow established plant procedures and radiological practices with respect to HRAs, albeit in this case it was a licensee employee, rather than a contractor, conducting a pre-job inspection in the Unit 1 heater bay.

- A finding described in Section 2OS1.7 involved procedural errors on the part of a radiation protection technician conducting a survey that resulted in a platform in the Unit 1 turbine building not being properly posted as a radiation area when radiological conditions on the platform required such a posting.

- A licensee-identified violation described in Section 4OA3.1 involved the failure of plant personnel to have properly reseated a safety-related fuse for the Unit 1 HPCS pump in accordance with plant procedures and clearance order instructions.

As can be seen from the above descriptions of each issue, these human performance deficiencies were procedure compliance/procedure use and adherence related. In each of the findings described in Section 2OS1, as well as the licensee-identified violation discussed in Section 4OA3.1, had the personnel involved followed existing plant procedures and instructions it is likely that the occurrences never would have taken place.

4OA5 Other

.1 Spent Fuel Material Control and Accounting At Nuclear Power Plants
(TI [Temporary Instruction] 2515/154)

a. Inspection Scope

The inspectors interviewed the station and Exelon corporate special nuclear material (SNM) custodians. The inspectors reviewed licensee procedures regarding the movement and accountability of SNM. The inspectors also reviewed a sample of recent inventories of nuclear fuel and SNM. Documents reviewed as part of this TI are listed in the Attachment. This TI was not a part of the baseline inspection program and was therefore not considered an inspection sample. Phases I and II of the TI are considered complete for LaSalle Station.

b. Findings

No finding of significance were identified.

.2 <u>(Closed) Unresolved Item 05000373/2003005-03</u>: Unit 1 Reactor Power Transient Due to Failed Open Reactor Feed Pump Minimum Flow Valve.

Since 1995, the licensee has documented at least five reactor feed pump minimum flow valve performance issues due to malfunctions of the conoflow snap acting (SNAP) relays within the valves' pneumatic control systems. The licensee determined through failure analysis that upgrading from Buna-N to Viton elastomers would serve as appropriate corrective action for the SNAP relay performance problems.

On November 27, 2003, the licensee entered a Unit 1 maintenance outage (L1M14). Several performance problems were noted with reactor feed pump minimum flow valves:

- The 1A turbine-driven reactor feed pump (TDRFP) minimum flow valve exhibited dual position indication, and a position feedback of 3.2 percent open with a full closed demand signal. Actual position was 10 percent open.
- The 1B TDRFP minimum flow valve exhibited a demand verses actual position mismatch.
- The 1C motor-driven reactor feed pump (MDRFP) minimum flow valve would not travel fully closed with a full close demand signal. Actual position was 23 percent open.

Though potential failure of the SNAP relays was considered, the licensee planned to defer repair/replacement of these components until the next scheduled refueling outage (L1R10 and L2R10) on each unit. Licensee management, supported by the site's engineering staff and based upon the symptoms that were evident, concluded that the possibility of a catastrophic failure and the probability of incurring a plant transient from the degraded condition were low.

On December 7, 2003, the licensee noted increased air leakage on the 1C MDRFP minimum flow valve SNAP relay. Licensee management made the decision to remove the 1C MDRFP from service and replace the minimum flow valve SNAP relay. The replacement relay was outfitted with Viton elastomers.

On December 10, 2003, the 1B TDRFP minimum flow valve failed open due to a SNAP relay air leak. The resulting reactor feedwater transient forced operators to reduce reactor power by lowering core recirculation flow. When core flow had been reduced to approximately 60 million pounds per hour, an unrelated trip of several feedwater heaters caused reactor power to increase. Operators noted that this resulting power increase placed the unit in Region 'B' of the power-to-flow instability map, and inserted control rods to reduce reactor power per applicable procedures. The total time the unit was within the bounds of Region 'B' was approximately 5 minutes, and reactor power was ultimately stabilized at 45 percent.

The validity of the licensee's decision to defer repairs for the SNAP relays until the next scheduled refueling outage (L1R10 and L2R10) on each unit was reviewed by the inspectors. In examining the licensee's root cause analysis for the transient, the inspectors determined that there were no licensee performance deficiencies associated with the event. As a result, no findings of significance or violations of regulatory requirements were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. G. Barnes, and other members of licensee management on April 13, 2004. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary materials were identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- A refueling outage inservice inspection program inspection with the Site Vice President, Mr. G. Barnes, on January 23, 2004.

- An occupational radiation safety radiological access control and ALARA inspection with the Site Vice President, Mr. G. Barnes, on January 28, 2004.

4OA7 Licensee-Identified Violations

Cornerstone: Mitigating Systems

The following violation of very low significance (Green) 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 an NCV.

Section 4OA3.1 of this report discusses a routine Unit 1 HPCS system surveillance in which licensee personnel discovered that a fuse in the circuitry for the system's low level initiation/high level trip logic was not fully seated in its holder. Licensee engineering personnel subsequently evaluated the as-found condition of the fuse and determined that circuit continuity could not be assured during a seismic event. Inspectors reviewing the issue determined it to have been of very low safety significance in that the actual function provided by the fuse had never been lost; and even had the circuit function been lost, the inspectors determined that the HPCS system was still capable of being manually initiated and secured by control room operators who had adequate procedural guidance to do so if necessary. However, contrary to the requirements of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," the inspectors determined that by failing to ensure that the HPCS fuse was fully seated upon restoration from a March 2002 clearance order, licensee personnel had failed to adhere to clearance order restoration instructions and procedural requirements.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
G. Barnes, Site Vice President
S. Landahl, Plant Manager
T. Connor, Maintenance Director
L. Coyle, Operations Director (2)
D. Czufin, Site Engineering Director
D. Enright, Operations Director (1)
S. Fatora, Radiation Protection Department Manager
A. Ferko, Nuclear Oversight Manager
F. Gogliotti, System Engineering Manager
P. Holland, Regulatory Assurance Coordinator
R. Jacobs, NDE Level 3
G. Kaegi, Regulatory Assurance Manager
B. Kapellas, Radiation Protection Manager
J. Rappeport, Nuclear Oversight
W. Riffer, Emergency Planning Manager
J. Rommel, Programs Engineering Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission
B. Burgess, Chief, Region 3 Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000373/2004002-01 05000374/2004002-01	URI	Discrepancies with Fire Watch Practices and Fire Watch Logs (Section 1R05)
05000373/2004002-02	NCV	Lack of Proper Procedural Guidance for Throttling an Instrument Nitrogen System Isolation Valve (Section 1R16)
05000373/2004002-03	URI	Unauthorized Entry into 1A TDRFP Room HRA by Contract Personnel (Section 2OS1.6(1))
05000373/2004002-04	NCV	Unauthorized Entry into 1B RHR Room HRA by Contract Personnel (Section 2OS1.6(2))
05000373/2004002-05	NCV	Unauthorized Entry into Unit 1 Heater Bay HRA by Licensee Craft Personnel (Section 2OS1.6(3))
05000373/2004002-06	URI	Unauthorized Entry into Unit 1 694’ Reactor Building Raceway HRA by Contract Personnel (Section 2OS1.6(4))
05000373/2004002-07	NCV	Inadequate Survey Results in Unposted Radiation Area (Section 2OS1.7)

Closed

05000373/2004002-02	NCV	Lack of Proper Procedural Guidance for Throttling an Instrument Nitrogen System Isolation Valve (Section 1R16)
05000373/2004002-04	NCV	Unauthorized Entry into 1B RHR Room HRA by Contract Personnel (Section 2OS1.6(2))
05000373/2004002-05	NCV	Unauthorized Entry into Unit 1 Heater Bay HRA by Licensee Craft Personnel (Section 2OS1.6(3))
05000373/2004002-07	NCV	Inadequate Survey Results in Unposted Radiation Area (Section 2OS1.7)
05000373/2003-004-00	LER	High Pressure Core Spray Inoperable Due to Improperly Seated Fuse (Section 4OA3.1)
05000373/2003-005-00	LER	Manual Reactor Scram on Low Reactor Vessel Water Level Due to Stuck Open Feedwater Pump Discharge Check Valve (Section 4OA3.2)
05000373/2003005-03	URI	Unit 1 Reactor Power Transient Due to Failed Open Reactor Feed Pump Minimum Flow Valve (Section 4OA5.2)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

Procedures:
- LOP-RH-04E; Unit 2 Residual Heat Removal System Electrical Checklist; Revision 13
- LOP-RH-03E; Unit 2 RHR Service Water System Electrical Checklist ; Revision 5
- LOP-DG-09E; 2A Diesel Generator Cooling System Electrical Checklist; Revision 4
- LOP-DG-09M; 2A Diesel Generator Cooling System Mechanical Checklist; Revision 7
- LOP-DG-03E; Unit 0 Diesel Generator Electrical Checklist; Revision 9
- LOP-DG-03M; Unit 0 Diesel Generator Mechanical Checklist; Revision 8
- LOP-DG-05E; 2B Diesel Generator Electrical Checklist; Revision 10
- LOP-DG-05M; 2B Diesel Generator Mechanical Checklist; Revision 7

1R05 Fire Protection

Updated Final Safety Analysis Report; Revision 13:
- Appendix H; Fire Hazards Analysis
- Section 9.5.1; Fire Protection System

Technical Requirements Manual:
- Section 3.7.j; Fire Suppression Water System; Revision 1
- Section 3.7.k; Deluge and Sprinkler Systems; Revision 1
- Section 3.7.m; Fire Hose Stations; Revision 1

Mechanical Maintenance Procedures:
- LMS-FP-15; TRM Fire Hose Stations Inspection; Revision 18

Exelon Procedures:
- OP-MW-201-007; Fire Protection System Impairment Control; Revision 0
- CC-AA-201; Plant Barrier Control Program; Revision 3

Surveillances:
- LMS-ZZ-03; Inspection of Fire Doors Separating Safety Related Fire Areas; Revision 7

Logs:
- Fire Watch Inspection Logs for Fire Doors 272, 417, 321, & 406; 2/1/2004 – 2/5/2004
- Fire Watch Inspection Logs for Fire Doors 231, 272, 256, 417, 406, Unit 1 Drywell, and Refuel Floor; 2/5/2004 – 2/6/2004
- Fire Watch Inspection Logs for Fire Door 272; 2/19/2004 – 2/21/2004
- Fire Watch Inspection Logs for Fire Door 256; 1/24/2004 – 2/5/2004
- Fire Watch Inspection Logs for Fire Impairment 1-03-132 TRM U1 DG Rooms; 1/17/2004 – 2/1/2004
- Fire Watch Inspection Logs for Fire Impairment 1-03-181 TRM U1 DG Corridor; 1/17/2004 – 1/28/2004
- Fire Watch Inspection Logs for Fire Door 471; 1/26/2004 –1/28/2004
- Fire Watch Inspection Logs for Fire Door 336; 1/24/2004 – 2/5/2004

- Fire Watch Inspection Logs for Fire Door 469; 1/13/2004 – 1/16/2004
- Fire Watch Inspection Logs for Fire Door 220; 1/17/2004 – 1/26/2004
- Fire Watch Inspection Logs for 2A EDG Room; 10/6/2003 – 10/9/2003
- Fire Watch Inspection Logs for Fire Door 469; 10/15/2003 – 10/16/2003
- Fire Watch Inspection Logs for Fire Door 351; 2/17/2003 – 2/17/2003
- Fire Watch Inspection Logs for Fire Door 256; 4/8/2003 – 4/10/2003
- Fire Watch Inspection Logs for Fire Door 503; 3/19/2003 – 3/19/2003
- Fire Watch Inspection Logs for Fire Door 128; 2/20/2003 – 2/22/2003
- Fire Watch Inspection Logs for Fire Door 731; 2/15/2003 – 2/19/2003
- Fire Watch Inspection Logs for Fire Door 406; 2/23/2003 – 2/27/2003
- Fire Watch Inspection Logs for Fire Door 615; 1/24/2003 – 2/19/2003
- Fire Watch Inspection Logs for U1 Drywell; 1/15/2004 – 2/6/2004

Condition Reports:
- 198921; Fire Barriers Exceeding 7 Day Time Clock; 1/31/2004
- 197970; Fire Watch was Inattentive to Duties; 1/28/2004
- 202427; Paper Work Not Maintained in the WEC; 2/18/2004
- 201611; NRC Concerns with Fire Protection System Impairment Control; 2/17/2004
- 200093; Hourly Fire Watch; 2/6/2004
- 201982; Fire Watch Inspection Log; 2/16/2004

Miscellaneous Documents:
- Memo from LaSalle Fire Marshall William Collins to Sun States Personnel; RE: Fire Protection Impairment Control; 2/6/2004
- Memo from LaSalle Fire Marshall William Collins to Sun States Personnel; RE: Door 272 Fire Watch Tour; 2/23/2004

1R07 Heat Sink Performance

Condition Reports:
- 196765; RHR Heat Exchanger Flange; 1/21/2004

Data Sheets:
- HX/Component Inspection Data Sheet for 1A RHR HX; 1/17/2004

Work Orders:
- 992275462-01; Unit 1 ‘A’ RHR Heat Exchanger Test; 1/23/2002

Surveillances:
- LTS-200-17; RHR Heat Exchanger Thermal Performance Monitoring; Revision 5

1R08 Inservice Inspection Activities

Procedures:
- GE-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Piping Welds; dated December 9, 2003
- ER-AA-335-005; Radiographic Examination; dated September 13, 2002

Condition Reports:
- 192589; Radiography of 1B21-F010A/B Check Valve Spool Pieces
- 193777; Radiography of 1B21-F010 Replacement Check Valve S/N AU-577
- 194308; Additional UT Indications on FW Check Valve 1B21-F010A
- 196552; UT Indications on Jet Pump Beam #15
- 195976; Enhancement-Calibration Blocks Difficult to ID and Move
- 195996; Visual Indications of Cracking on Reactor Steam Dryer
- 196072; Additional Cracking Detected on Reactor Steam Dryer

1R11 Licensed Operator Requalification Program

ESG00C5-36; [Operator Simulator Exam Scenario – Title Withheld]; Revision 2

EP-MW-114-100, Attachment 1; Nuclear Accident Reporting System; Revision 3

1R12 Maintenance Effectiveness

Condition Reports:
- 198665; B RPS MG Set Reading 3 Volts Lower than Nominal; 1/31/2004
- 198850; B RPS Trip Caused Half SCRAM and PCIS Isolations; 2/2/2004
- 200667; B RPS MG Output Voltage Degradation; 2/10/2004
- 203177; Crew 2 Critique of Crew Response to Loss of U2 RPS; 2/21/2004
- 204449; 2B RPS Scram Lights Flickering after LOS-RP-W1; 2/26/2004
- 205298; NOS ID Ineffective Maintenance on Unit 2 RPS System; 3/1/2004
- 208783; RPS MG Set Output Breaker Tripped During Load Test; 3/16/2004
- 209028; 2B RPS EPA Trip Settings Need to Be Proceduralized; 3/17/2004
- 209582; Found Silver Plating on Contacts Removed for Relay 2K in RPS; 3/19/2004

1R13 Maintenance Risk Assessments and Emergent Work Control

Drawings and Prints:
- M-66; Drywell Pneumatic System; Revision AE

Procedures:
- LOA-IN-201; Loss of Drywell Pneumatic Air Supply; Revision 4
- LOR-2H13-P601-F102; Automatic Depressurization System (ADS) Valve Accumulator Pressure Low; Revision 5
- LOP-IN-05; Replacing Nitrogen Bottles on Instrument Nitrogen System; Revision 4
- LOP-IN-01; Drywell Pneumatic System Startup and Operation; Revision 22
- LIS-FW-301; Unit 1 Reactor Vessel High Water Level 8 Main Turbine/Feedwater Pump Trip Functional Test; Revision 12

Condition Reports:
- 148142; Jet Pump Operability Tests; 3/8/2003
- 198860; B IN Compressor Failed to Trip on Low Suction Pressure; 2/2/2004
- 198871; ADS Accumulators U and E Low Pressure in Alarm; 2/2/2004
- 198858; Spurious Loss of North ADS Bottle Bank Pressure; 2/2/2004
- 199734; 2A IN Compressor Failed to Load; 2/5/2004

- 199264; Found TDR Unplugged for ‘B’ IN Compressor
- 201867; Receiving Spurious Level 8 Trips on A Channel; 2/16/2004

Operability Evaluations:
- OE 04-001; Unit 2 Nitrogen Bottle Bank Pressure Regulators for ADS Valves; Revision 0
- OE 04-002; Unit 1 Reactor Jet Pumps 5/6 and 9/10; Revision 0

Work Orders:
- 667998; LMT-01-LIS-FW-301 U1 Rx Vsl Lvl 8 Turb/FW Pmp Trip; 2/20/2004

1R14 Operator Performance During Non-Routine Plant Evolutions and Events

Condition Reports:
- 198850; Loss of the Unit 2 ‘B’ RPS Bus; 2/1/2004

1R15 Operability Evaluations

Procedures:
- LOS-AA-S101; Unit 1 Shiftly Surveillance; Revision 18
- LOS-IN-R2; ADS Accumulator Unregulated N_2 Header Drywell Supply Check Valve Test; Revision 1
- LOS-IN-R3; Drywell Pneumatics Bottle Bank Regulator Adjustment and Emergency Pressurization Station Check Valve Exercise; Revision 5
- LOR-1PM13J-B404; Instrument Nitrogen System Trouble; Revision 3
- LOR-2H13-P601-F102; Automatic Depressurization System (ADS) Valve Accumulator Pressure Low; Revision 5
- LOP-IN-05; Replacing Nitrogen Bottles on Instrument Nitrogen System; Revision 4
- LOA-IN-101; Loss of Drywell Pneumatic Air Supply; Revision 3

Engineering Changes:
- EC 341543; Additional Margin for Drywell Bulk Temperature; 4/25/2003

Standing Orders:
- S04-006; Compensatory Actions for Unit 1 IN Op Eval 04-003; 2/26/2004

Calculations:
- L-002880; Additional Margin for Drywell Bulk Temperature; Revision 0A

Condition Reports:
- 148142; Jet Pump Operability Tests; 3/8/2003
- 203368; Deficient Equipment Design of IN Regulator; 2/23/2004
- 204643; ADS Bottle Bank Installed Bottle Size Below Design Calculation Value; 2/27/2004
- 204664; LOS-IN-R3 Has Errors; 2/27/2004
- 206063; Operability Determination 04-003 Nitrogen Bottle Size; 3/4/2004
- 206183; Summary of Issues: IN/ADS on Unit 1; 3/4/2004
- 204713; NOS Id: Inadequate Safety Plans for Induced N_2 Leakage

Operability Evaluations:
- OE 04-001; Unit 2 Nitrogen Bottle Bank Pressure Regulators for ADS Valves; Revision 0
- OE 04-002; Unit 1 Reactor Jet Pumps 5/6 and 9/10; Revision 0
- OE 04-003; Unit 1 Nitrogen Bottle Bank Pressure Regulator for ADS Valves; Revisions 0 and 1
- OE 04-004; Unit 1 and Unit 2 Reactor Water Cleanup Differential Flow Isolation Circuit Wires Not Enclosed in Flexible Conduit per Design; Revision 0

Drawings and Prints:
- M-66; Drywell Pneumatic System; Revision AE

1R16 Operator Workarounds

Condition Reports:
- 203368; Deficient Equipment Design of IN Regulator; 2/23/2004
- 204664; LOS-IN-R3 has Errors; 2/27/2004
- 204643; ADS Bottle Bank Installed Bottle Size Below Design Calc Value; 2/27/2004

Operability Evaluations:
- OE 04-003; Unit 1 Nitrogen Bottle Bank Pressure Regulator for ADS Valves; Revisions 0 & 1

Procedures:
- LOS-AA-S101; Unit 1 Shiftly Surveillance; Revision 18
- OP-AA-102-103; Operator Work-Around Program; Revision 1
- LOS-IN-R2; ADS Accumulator Unregulated N_2 Header Drywell Supply Check Valve Test; Revision 1
- LOS-IN-R3; Drywell Pneumatics Bottle Bank Regulator Adjustment and Emergency Pressurization Station Check Valve Exercise; Revisions 4 & 5
- LOR-1PM13J-B404; Instrument Nitrogen System Trouble; Revision 3
- LOR-2H13-P601-F102; Automatic Depressurization System (ADS) Valve Accumulator Pressure Low; Revision 5
- LOP-IN-05; Replacing Nitrogen Bottles on Instrument Nitrogen System; Revision 4
- LOA-IN-101; Loss of Drywell Pneumatic Air Supply; Revision 3
- LOP-AP-101, Attachment K; Station Blackout Contingencies; Revision 17

Operations Standing Orders:
- S04-006; Compensatory Actions for Unit 1 IN Op Eval #04-003; 2/26/2004

Engineering Changes:
- EC 347585; Installation of Temporary Alternate Nitrogen Gas Supply System to Supplement the North Bottle Bank 1IN09MB That is Being Vented Off due to Leaking Pressure Regulator Valve 1IN038; Revisions 0, 1, & 2

LaSalle Operators Open Challenge and Workarounds Database as of 3/1/2004

1R17 Permanent Plant Modifications

Engineering Changes:
- EC 340547; Drywell Fill Up Rate Conductor Termination Relocation; Revisions 0 & 1
- EC 341950; Unit 1 RHR WS Keep Fill Elimination Project; Revisions 0,1, & 5

Drawings:
- 1E-1-4374AS; Wiring Diagram Miscellaneous Equipment "RE" & "RF" Systems; Revision K
- 1E-1-4315AA; Internal/ External Wiring Diagram Instrument "A" Electrical Penetration 1LV94E (E-20) Part 1; Revision J
- 1E-1-4315AB; Internal/ External Wiring Diagram Instrument "A" Electrical Penetration 1LV94E (E-20) Part 2; Revision F
- 1E-1-4315AC; Internal/ External Wiring Diagram Instrument "A" Electrical Penetration 1LV94E (E-20) Part 3; Revision K
- M-1302; Reactor Building Instrument Piping Plan El. 710'-6" Area 2 Sheet 26; Revision AM
- M-1302; Reactor Building Instrument Piping Plan El. 694'-6" Area 2 Sheet 30; Revision AG
- M-1340; Instrument Detail V-Cone 20" Carbon Steel with ANSI Class 150 Raised Face Welding Neck Sheet 181; Revision A

Work Orders:
- 531335; Drywell Fillup Rate Movement; 1/28/2004
- 560654; Unit 1 RHR WS Keep Fill Elimination; 2/5/2004
- 465580; Drywell Floor Drain Sump Fillup Rate; 1/25/2004

Procedures:
- LLP-2004-001; Startup of the Unit 1 Division 2 RHR Service Water System with the 1E12-F014B Throttled; Revision 0

Surveillances:
- LTS-100-17; Local Leak Rate Test of Conax Electrical Penetrations Unit 1(2); Revision 15
- LIS-PC-112; Unit 1 Drywell Floor Drain Sump Fillup Rate Calibration; Revision 13

Calculations:
- L-000120; Recalibration of RHR Service Water Flow Transmitters; Revisions 0, 0A, & 0B

Condition Reports:
- 198307; High Indicated Pressure During Div. 2 RHR WS Run; 1/29/2004

1R19 Post-Maintenance Testing

Surveillances:
- LIS-PC-112; Unit 1 Drywell Floor Drain Sump Fillup Rate Calibration; Revision 13
- LES-RP-101; Inspection of the Reactor Protection Motor-Generator Sets; Revision 13

- LOS-RI-Q3; Unit 2 Reactor Core Isolation Cooling System Pump Operability and Inservice Test in Conditions 1,2, and 3; Revision 38
- LTS-700-6; Unit 1(2) Division 1 Battery Service Test Discharge; Revision 20
- LTS-100-2; As-Left LLRT of 1B21-F016/19 Inbd MSIV Drain Valves; Revision 26
- LTS-100-4; Inboard MSIV's Drain Isolation Valves Local Leak Rate Test (1(2)B21-F016 and 1(2)B21-F019); Revision 18

Procedures:
- MA-LA-773-462; U-2, Electrical Protection Assembly (EPA) and RPS MG Set Calibration by OAD; Revision 0
- LLP-2004-001; Startup of the Unit 1 Division 2 RHR Service Water System with the 1E12-F014B Throttled; Revision 0

Work Orders:
- 465580; Unit 1 Drywell Floor Drain Sump Fillup Rate Cal; 1/25/2004
- 657383; Unit 2 RCIC Pump Op & Inservice Test, Att 2A; 2/24/2004
- 465643; Unit 1 125 Vdc Battery Division 1 Service Test Discharge; 1/20/2004
- 465595; LLRT 1B21-F016, 1B21-F019; 1/14/2004
- 660937; LLRT 1B21-F016, 1B21-F019; 2/2/2004
- 661734; Loss of Unit 2 'B' RPS Bus- Suspect M-G Set Voltage Problem; 2/7/2004

Condition Reports:
- 203741; U2 RCIC Window Extended Due to Emergent Work on 2E51-N010 Sw; 2/24/2004
- 203802; Tripped RCIC Turbine Due to Low Oil Level During Start; 2/24/2004
- 151164; LOS-RI-Q5 Observations; 3/28/2003
- 195007; L1R10 LLRT on MSLD 1B21-F019 Exceed Admin Alarm Limit; 1/14/2004
- 198307; High Indicated Pressure During Div. 2 RHR WS Run; 1/29/2004

Engineering Changes:
- EC 340547; Drywell Fill Up Rate Conductor Termination Relocation; Revisions 0 & 1
- EC 341950; Unit 1 RHR WS Keep Fill Elimination Project; Revisions 0,1, & 5

1R20 Outage Activities

Condition Reports:
- 197490; Near Miss During Upgrade of Bus Potential Transformer (PT) for Live 6.9KV - Cable Shield Ground Lifted on Energized 6.9KV Cable; 1/26/2004
- 197457; Workers Entered High Radiation Area Without Brief, Without Knowledge of Area Dose Rates and Contamination Levels, and on Wrong RWP; 1/25/2004
- 196006; Wrong Control Rod Unlatched; 1/17/2004

NF0300069; LaSalle Unit 1 Cycle 11 Design Basis Loading Plan; Revision 1

Procedures:
- LOP-AA-03; Reactor Mode Changes; Revision 19
- LOP-NB-01; Reactor Vessel Leakage Test; Revision 40
- LOP-FC-16; Reactor Vessel/Cavity Draindown via RHR Shutdown Cooling; Revision 13

- LGP-1-1; Normal Unit Startup; Revision 71
- LGP-1-S1; Master Startup Checklist; Revision 54
- LTP-1500-2; Alternate Decay Heat Removal Lineup Capabilities; Revision 3
- LTS-1100-4; Scram Insertion Times; Revision 22
- LOP-DW-01; Drywell Close Out (After Outage); Revision 36

Work Orders:
- 560679; LOP-DW-01, U-1 Drywell Closeout (After Outage); 2/10/2004

Engineering Calculations:
- EC 346082; Use of Fuel Pool Cooling as an ADHR system; Revision 0

Miscellaneous Documents:
- L1R10 Shutdown Safety Plan; 1/6/2004

1R22 Surveillance Testing

Surveillances:
- LOS-DG-111; Integrated Division 3 Response Time Surveillance; Revision 0
- LTS-500-111; Integrated Division 3 Response Time Surveillance; Revision 96
- LTS-100-4; Inboard MSIV's Drain Isolation Valves Local Leak Rate Test (1(2) B21-F016 and 1(2) B21-F019); Revision 18
- LOS-RI-R3; Reactor Core Isolation Cooling System Pump Operability Test; Revision 27
- LOS-DG-M2; 1A Diesel Generator Idle Start Attachment 1A; Revision 54
- LOS-RH-Q1; RHR(LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1,2,3,4 and 5; Revision 54

Work Orders:
- 99116084; Integrated Division 3 ECCS Response Time; 1/10/2002
- 465595-01; LLRT for 1B21-F016, 1B21-F019; 1/14/2004
- 666335; 1A Diesel Generator Idle Start; 2/25/2004
- 623791; LOS-RH-Q1 2B RHR System Att 2B; 1/7/2004
- 465662; LOS-RI-R3 U1 RCIC Att 1A; 2/11/2004

Condition Reports:
- 193753; Various Minor Deficiencies Noted During LOS-RH-Q1; 1/6/2004
- 195007; L1R10 LLRT on MSLD 1B21-F019 Exceed Admin Alarm Limit; 1/14/2004

1R23 Temporary Plant Modifications

Procedures:
- CC-MW-112-1001; Temporary Configuration Change Packages; Revision 3
- LIS-PC-412; Unit 2 Drywell Floor Drain Sump Fillup Rate Functional Test; Revision 5
- LOS-IN-R2; ADS Accumulator Unregulated N_2 Header Drywell Supply Check Valve Test; Revision 1
- LOS-IN-R3; Drywell Pneumatics Bottle Bank Regulator Adjustment and Emergency Pressurization Station Check Valve Exercise; Revisions 4 & 5
- LOR-1PM13J-B404; Instrument Nitrogen System Trouble; Revision 3

- LOR-2H13-P601-F102; Automatic Depressurization System (ADS) Valve Accumulator Pressure Low; Revision 5
- LOP-IN-05; Replacing Nitrogen Bottles on Instrument Nitrogen System; Revision 4
- LOA-IN-101; Loss of Drywell Pneumatic Air Supply; Revision 3
- LOP-AP-101, Attachment K; Station Blackout Contingencies; Revision 17

Work Orders:
- 641882; Contingency Troubleshooting Task for Unit 0,1, & 2; 12/31/2003
- 645535; LMT-02-LIS-PC-412 Drywell Flr Drn Smp Fillup Rate; 12/9/2003

Temporary Change Control Packages:
- 346600; Lift Leads for Hi-Hi Level Switch 2LS-RF003; 1/13/2004
- 346343; DWFDS Lift Shield Leads from Drywell Due to Noise Induction; 12/18/2003
- 346749; Provide Temporary Power to DG-0 Immersion Heater H1, Oil Circulating Pump B7, and Engine Lube Oil Soak Back Pump B7A; 1/17/2004
- 347585; Installation of Temporary Alternate Nitrogen Gas Supply System to Supplement the North Bottle Bank 1IN09MB That is Being Vented Off due to Leaking Pressure Regulator Valve 1IN038; Revisions 0, 1, & 2

Engineering Changes:
- 346472; Manual Operation of RF Sump Pumps with Failed Hi-Hi Level Switch; Revision 0
- 346757; Evaluation of EC 346749, MR90 Temporary Power to 0 D/G; Revision 0

Operational Evaluations:
- OE 04-003; Unit 1 Nitrogen Bottle Bank Pressure Regulator for ADS Valves; Revisions 0 & 1

Condition Reports:
- 189668; 2UR-RF002 Recorder Erratic During LIS-PC-412; 12/9/2003
- 191226; U2 DWFDS Fill Up Rate Loop Noise Troubleshooting Results; 12/16/2003
- 203368; Deficient Equipment Design of IN Regulator; 2/23/2004
- 204664; LOS-IN-R3 has Errors; 2/27/2004
- 204643; ADS Bottle Bank Installed Bottle Size Below Design Calc Valu; 2/27/2004

2OS1 Access Control to Radiologically Significant Areas

Procedures:
- RP-LA-460-1002; Additional High Radiation Exposure Controls; Revision 0
- RP-AA-462; Controls for Radiographic Operations; Revision 3

Radiation Work Permits:
- 10002615; L1R10 I/B MSIV Work Activities; Revision 0
- 10002629; L1R10 RHR Service Water Keep Fill Modification; Revision 1

Condition Reports:
- 138407; Unposted Radiation Area Violation of 10 CFR 20; January 7, 2003
- 140276; RPT Left Steam Sensitive Area While Occupied; January 21,2003

- 140976; Qualification Verification of Workers Using Air Sample Equipment; January 24, 2003
- 144763; Technical Specification 5.7 High Radiation Area Unguarded; January 27, 2003
- 167884; HRA/LHRA Assessment Deficiencies During Attachment Review; July 16, 2003
- 169794; Locked Hi-Rad Door Degradation; July 31, 2003
- 173490; Radiological Posting Found on Ground; August 28, 2003
- 182312; Access Control and PI FASA Identified Issues; October 22, 2003
- 187284; Radiation Protection FASA Results; November 12, 2003
- 188354; Locked High Rad Area Boundary Degraded- Near Miss; November 27, 2003
- 192902; Workers Enter HRA Without Brief; December 30, 2003
- 194434; Unposted Radiation Area Identified on Turbine Deck Elevated Platform; January 7, 2004
- 196455; Workers Enter HRA on Wrong RWP; January 20, 2004
- 196585; Incorrect RWP Usage; January 20, 2004
- 196819; Inadvertent Entry into a Contaminated High Radiation Area Without a Radiation Protection Briefing; January 21, 2004
- 196885; Worker Received a Dose Rate Alarm; January 20, 2004
- 196894; EPD Dose Rate Alarm Sounded While Moving the Separator; January 22, 2004
- 197375; Carpenter Received Dose Alarm; January 23, 2004
- 197457; Workers Enter HRA Without Brief; January 25, 2004

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

Procedures:
- RP-AA-220; Intake Investigation; Revision 1
- RP-AA-400; ALARA Program; Revision 3
- RP-AA-400-1001; Establishing Collective Radiation Exposure Estimates and Goals; Revision 0
- RP-AA-400-1002; Dose Equalization; Revision 0
- RP-AA-401; Operation ALARA Planning and Controls; Revision 3

Radiation Work Permits:
- 10002580; U-1 Drywell SRV Activities; Revision 0
- 10002591; Remove and Replace Drywell Snubbers; Revision 0
- 10002600; L1R10 CRD Pull/Put; Revision 2
- 10002610; Replace 1B21-F010A/10B Feed Water Check Valves; Revision 0
- 10002617; Repair of the 1B33-F023A Valve (L1R10); Revision 3
- 10002629; 1B RHR Service Water Keep Fill Modification; Revision 1

Condition Reports:
- 158381; NOS RP First Quarter Ineffective and Declining Trend; April 30, 2003
- 168009; NOS Identifies Second Quarter RP Rating Ineffective and Declining; July 17, 2003
- 173730; Task Not on Dose Goal Sheet; September 8, 2003
- 173749; Not Dose Goal Assigned to IM Task; September 8, 2003
- 173889; Emergent Work/Management Leads to Unnecessary Dose; September 2, 2003

- 175631; Seven of Twelve Cameras (High Rad) Not Working; September 13, 2003
- 176151; Observation of RP-AA-460 Attachments; September 17, 2003
- 176333; RWP DIGI Setpoints Did Not Match RWP; September 18, 2003
- 176437; Radiological Controls Not Met as Defined in ALARA Plan; September 17, 2003
- 177706; Rad Worker Arrived at Job Site Without Proper DIGI; September 26, 2003
- 185232; Contamination Spread From Use of Grinder on Clean Materials; November 7, 2003
- 191198; Venture PSA Laborer Entered High Radiation Area On Wrong RWP; December 16, 2003
- 193737; Escalation Notice – Limited RP Program Effectiveness; January 13, 2004

4OA1 Performance Indicator Verification

Procedures:
- LS-AA-2090; Monthly Performance Indicator (PI) Data Elements for Unplanned Power Changes per 7000 Critical Hours; Revision 3

Condition Reports:
- 210248; DEI Sample Result Not Used in NRC PI for Unit 2 Aug 03; 3/22/2004

4OA2 Identification and Resolution of Problems

Condition Reports:
- 198665; B RPS MG Set Reading 3 Volts Lower than Nominal; 1/31/2004
- 198850; B RPS Trip Caused Half SCRAM and PCIS Isolations; 2/2/2004
- 200667; B RPS MG Output Voltage Degradation; 2/10/2004
- 203177; Crew 2 Critique of Crew Response to Loss of U2 RPS; 2/21/2004
- 204449; 2B RPS Scram Lights Flickering after LOS-RP-W1; 2/26/2004
- 205298; NOS ID Ineffective Maintenance on Unit 2 RPS System; 3/1/2004
- 208783; RPS MG Set Output Breaker Tripped During Load Test; 3/16/2004
- 209028; 2B RPS EPA Trip Settings Need to Be Proceduralized; 3/17/2004
- 209582; Found Silver Plating on Contacts Removed for Relay 2K in RPS; 3/19/2004

4OA3 Event Follow-up

Licensee Event Reports:
- 05000373/2003-004-00; High Pressure Core Spray Inoperable Due to Improperly Seated Fuse; 1/6/2004
- 05000373/2003-005-00; Manual Reactor Scram on Low Reactor Vessel Water Level Due to Stuck Open Feedwater Pump Discharge Check Valve; 1/23/2004

Work Orders:
- 642124; 1A TDRFP Discharge Check Valve Possibly Sticking in the Open Position; 11/27/2003

Drawings and Prints:
- M-57, Sheet 1; Unit 1 Feedwater and Zinc Injection; Revision F
- M-54, Sheet 6; ASME Section XI ISI Classification Descriptions; Revision A

Procedures:
- LAP-200-7; Post Event Review Program; Revision 8

Engineering Changes/Analyses:
- EC 346036; Review of Effects of Lost Parts From TDRFP Discharge Check Valves 1FW001A/B; 11/29/2003
- EC 334921: Review Effects of Debris Found in 16B Feedwater Heater; 1/21/2002

Condition Reports:
- 186839; Unit 1 HPCS Fuse (1B21A-F8) not Fully Seated; 11/17/2003
- 188345; Unit 1 Reactor Scram; 11/27/2003
- 188393; Closure Failure of the 1A TDRFP Discharge Check Valve; 11/28/2003

4OA5 Other

Condition Reports:
- 190091; 1B TDRFP Minimum Flow Valve Failed Open; 12/10/2003

Procedures:
- LTP-1200-2; Attachments for Special Nuclear Material (SNM) Inventories and Piece Counts to Support NF-AA-330; Revision 0
- NF-AA-330; Special Nuclear Material Physical Inventories; Revision 1

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
ALARA	As-Low-As-Reasonably-Achievable
APRM	Average Power Range Monitor
ARM	Area Radiation Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CAP	Corrective Action Program
CEDE	Committed Effective Dose Equivalent
CFR	Code of Federal Requirements
CR	Condition Report
CRD	Control Rod Drive
CSCS	Core Standby Cooling System
DAC	Distance Amplitude Correction
DBD	Design Basis Document
DG	Diesel Generator
d/p	Differential Pressure
DRP	Division of Reactor Projects
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPD	Electronic Personnel Dosimeter
EPMA	Electrical Power Monitoring Assembly
FSAR	Final Safety Analysis Report
HCU	Hydraulic Control Unit
HOO	Headquarters Operations Officer
HPCI	High Pressure Core Injection
HPCS	High Pressure Core Spray
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IN	Instrument Nitrogen
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LHRA	Locked High Radiation Area
LOS	LaSalle Operating Surveillance
MDRFP	Motor-Driven Reactor Feed Pump
MG	Motor-Generator
MOV	Motor-Operated Valve
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NLO	Non-Licensed Operator
NRC	U.S. Nuclear Regulatory Commission
OWA	Operator Workaround
PARS	Publicly Available Records

PCIS	Primary Containment Isolation System
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Testing
psid	Pounds Per Square Inch Differential
psig	Pounds Per Square Inch Gauge
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RP	Radiation Protection
RPS	Radiation Protection Specialist
RPS	Reactor Protection System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SLC	Standby Liquid Control
SNM	Special Nuclear Material
SRV	Safety Relief Valve
TDRFP	Turbine-Driven Reactor Feed Pump
TEDE	Total Effective Dose Equivalent
TI	Temporary Instruction
TRM	Technical Requirements Manual
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
Vdc	Volts Direct Current
VHRA	Very High Radiation Area