May 2, 2006

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 05000373/2006003;
05000374/2006003

Dear Mr. Crane:

On March 31, 2006, 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 results of this inspection, which were discussed on April 6, 2006, with the Site Vice President, Ms. Susan Landahl, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and 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, one NRC-identified finding and one self-revealed finding of very low safety significance were identified. Both of these findings also involved violations of NRC requirements. However, because the findings associated with these violations were of very low safety significance and because the issues were entered into the licensee's corrective action program, the NRC is treating these issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRCs Enforcement Policy. Additionally, one licensee-identified violation of very low safety significance is listed 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 copies 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 Inspectors' Office at the LaSalle County Station.

In accordance with 10 CFR 2.390 of the NRCs "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 NRCs 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 05000373/2006003; 05000374/2006003
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
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2006003; 05000374/2006003
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, 2006
Inspectors:	D. Kimble, Senior Resident Inspector D. Eskins, Resident Inspector C. Brown, Project Engineer (Acting) D. Jones, Engineering Inspector B. Jorgensen, NRC Contractor D. McNeil, Reactor Engineer (Lead Inspector) M. Mitchell, Radiation Protection Inspector C. Phillips, Senior Resident Inspector – Dresden Station D. Schrum, Engineering Inspector N. Shah, Project Engineer M. Sheikh, Resident Inspector – Dresden Station S. Sheldon, Reactor Engineer J. Yesinowski, Illinois Dept. of Emergency Management
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000373/2006003, 05000374/2006003; 01/01/2006 - 03/31/2006; LaSalle County Station, Units 1 and 2; Operator Performance During Nonroutine Evolutions and Events and Access Control To Radiologically Significant Areas Report.

The inspection was conducted by resident inspectors and regional inspectors. The report covers a 3-month period of resident baseline inspection, and announced baseline inspections in the areas of refueling outage inservice inspection and radiation protection. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using NRC 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 NRCs 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. Inspector-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. Inspectors identified a finding of very low safety significance for the lack of written procedures and instructions related to the Technical Specification administrative control of a primary containment isolation valve (PCIV). An associated Non-Cited Violation (NCV) of 10 CFR 50, Appendix B, Criterion V, was also identified.

  The finding was determined to be more than minor in that it directly affected the configuration control and procedure quality attributes of the Barrier Integrity cornerstone (containment) and affected the cornerstone objective of providing reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events. Because the finding did not represent a degradation of the radiological barrier function provided for the control room, auxiliary building, reactor building, or the standby gas treatment (SBGT) system, and did not represent a degradation of the smoke or toxic gas barrier function for the control room, and did not represent an actual open pathway in the physical integrity of the primary containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the primary containment, the inspectors determined it to be of very low safety significance (Green) and within the licensee’s response band. The licensee had entered this issue into their corrective action program as Issue Report (IR) 475214. Corrective actions planned by the licensee included development of a formal process for using administrative controls to meet Technical Specification requirements. (Section 1R14.2)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance was self-revealed as a result of an alarm on a worker's electronic dosimeter. The issue was identified when an instrument maintenance technician logged onto the wrong radiation work permit (RWP) and entered the assigned work area in the radiologically controlled area (RCA), Unit 1 Division 1 residual heat removal (RHR) room, a posted high radiation area (HRA). The primary cause of this finding was related to human performance. The technician failed to verify that he/she was on the correct RWP for the assigned work.

  The finding was more than minor because the occurrence involved an individual worker's potential unplanned, unintended dose resulting from actions or conditions contrary to licensee procedures, and could be reasonably viewed as a precursor to a more significant event. The finding was determined to be of very low safety significance because the finding did not involve an as low as reasonably achievable (ALARA) issue, as collective dose was not an issue and the individual's radiation exposure was low relative to regulatory limits; there was not a substantial potential for a worker overexposure; and the licensee's ability to assess worker dose was not compromised. The finding was a Non-Cited Violation of Technical Specification 5.4.1.a., which requires the licensee to establish, implement and maintain procedures recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Corrective actions planned by the licensee included increased management oversight during RWP log in and issuance of a site communication regarding the event. (Section 2OS1.4)

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

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 and associated corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Unit 1

The unit began the inspection period operating at full power. Due to reaching the end of the reactor core cycle, three load reductions were required during the inspection period to facilitate control rod pattern adjustments:

- On January 9, 2006, power was reduced to approximately 71 percent;
- On January 28, 2006, power was reduced to approximately 83 percent; and
- On February 11, 2006, power was reduced to approximately 71 percent.

In each of the above cases, the unit returned to full power operation later the same day. On February 20, 2006, reactor power was reduced to approximately 6 percent as part of a shutdown for a scheduled unit refueling outage. However, due to a steam transient induced by a malfunction of the turbine control system, the reactor was shutdown by an automatic scram from the reactor protection system on this date (Sections 1R14 and 4OA3). Following completion of the refueling outage, the unit commenced startup operations on March 17, 2006, and achieved criticality that same day. The unit returned to full power on March 21, 2006, and operated at or near full power for the remainder of the inspection period.

Unit 2

The unit began the inspection period operating at full power. On January 7, 2006, power was reduced to approximately 82 percent to perform minor maintenance on the turbine control and reactor feed systems and to perform a control rod sequence exchange. The unit returned to operation at full power that day. On February 12, 2006, power was reduced to approximately 66 percent to perform maintenance on the turbine control system, make repairs to the heater drain system, perform control rod surveillance testing, and perform a control rod sequence exchange. The unit returned to full power later that day and remained operating at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown of the following 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:

- Unit 1 reactor core isolation cooling (RCIC) system and automatic depressurization system (ADS) while the Unit 1 high pressure core spray (HPCS) system was inoperable during the week ending January 14;
- Unit 1 Division 1 emergency core cooling system (ECCS) during maintenance activities on the Unit 1 Division 2 residual heat removal service water (RHRSW) system during the week ending January 21; and
- Unit 2 Division 2 and Division 3 ECCS protected paths during a Technical Specification extended allowed outage time for valve and pump replacement maintenance on the Division 1 core standby cooling system (CSCS) during the week ending March 11.

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

These reviews constituted three partial equipment alignment inspection samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors walked down the following 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:

- Fire Zone 4A, auxiliary building upper ventilation equipment floor, elevation 815'0";
- Fire Zone 4B, auxiliary building lower ventilation equipment floor, elevation 786'6";
- Fire Zone 4D3, Unit 1 electrical equipment room, elevation 749'0";
- Fire Zone 5D1, Unit 1 HPCS switchgear zone, elevation 687'0";
- Fire Zone 5D2, Unit 2 HPCS switchgear zone, elevation 687'0";
- Fire Zone 8C3, Unit 2 HPCS diesel pump room, elevation 674'0";
- Fire Zone 8C4, Unit 2 Division 2 RHRSW pump room, elevation 674'0";
- Fire Zone 8C5, Unit 2 Division 1 RHRSW pump room, elevation 674'0"; and
- Fire Zone 9D1, technical support center and operational support center, elevation 694'6".

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive suppression capabilities, automatic suppression capabilities, and barriers to fire propagation. Areas of particular focus for these inspection samples were hot work (e.g., welding, grinding, etc.) conducted during the licensee's L1R11 refueling outage.

These reviews constituted nine quarterly fire protection inspection samples..

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

During the licensee's L1R11 refueling outage, the inspectors reviewed the thermal performance testing of the 1A RHR heat exchanger and the visual inspection of the 1B RHR heat exchanger to verify that any 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, 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.

This review constituted one inspection sample.

b. Findings

No findings of significance were identified. One unresolved item (URI) was identified.

The licensee's RHR heat exchangers on both units are tested for thermal performance and inspected in accordance with a licensee program intended to meet the recommendations of NRC Generic Letter (GL) 89-13. In their review of this program, the inspectors raised several questions regarding its implementation.

During the years 1999 and 2000, all four RHR heat exchangers (two on each unit) were chemically cleaned. In accordance with their GL 89-13 program, the licensee was required to conduct a baseline thermal performance test on each heat exchanger following these chemical cleanings to establish initial heat exchanger performance parameters. However, during their review of the licensee's program the inspectors were unable to verify that all these initial baseline tests had been performed as required. One specific example involved the 2B RHR heat exchanger, which appeared to have never had a successful thermal performance test since it was chemically cleaned.

In addition to the initial baseline testing following chemical cleaning, the requirements within GL 89-13 specified that an initial test program of at least three thermal performance tests with no intervening cleaning be performed on each RHR heat exchanger. During their review, the inspectors identified that the licensee has routinely performed non-chemical cleaning of the service water side of the RHR heat exchangers between tests using high pressure water. When questioned by the inspectors about this routine cleaning and its potential effect on the GL 89-13 testing program, the licensee had no immediate answer.

Once the initial test program was completed, the GL 89-13 continuing program for periodic RHR heat exchanger performance monitoring established a test frequency of once per outage and allowed the licensee to alter the frequency for testing based on established performance trends. During their review, the inspectors identified that the licensee has established a current RHR heat exchanger test frequency of 4 years. Again, when questioned by the inspectors regarding the basis for this 4 year frequency, the licensee had no immediate answer.

The inspectors also questioned the licensee's present practice of testing a single RHR heat exchanger in lieu of performance testing all four RHR heat exchangers. Upon their review, the inspectors found no justification concerning comparable service conditions or other appropriate reasoning within the licensee's program that would allow for the performance test results from one RHR heat exchanger to be applied to all four.

The licensee has entered these issues into their corrective action program as Issue Reports (IRs) 458571, 463253, and 473455, and has indicated plans to perform an apparent cause evaluation on RHR heat exchanger thermal performance testing deficiencies. This issue is considered unresolved, pending the inspectors' review of the licensee's corrective action program evaluations. (URI 05000373/2006003-01; 05000374/2006003-01)

1R08 Inservice Inspection (ISI) Activities (71111.08)

.1 Piping Systems ISI

a. Inspection Scope

From February 27, 2006, to March 2, 2006, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system (RCS) boundary, and the risk significant piping system boundaries during Unit 1 refueling outage L1R11. The inspectors selected the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, required examinations and code components in order of risk priority as identified in Section 71111.08-02 of NRC Inspection Procedure (IP) 71111.08, "Inservice Inspection Activities," based upon the ISI activities available for review during the on-site inspection period.

The inspectors conducted an on-site review of the following types of nondestructive examination activities to evaluate compliance with the ASME Code, Section XI and Section V requirements, and to verify that indications and defects, if present, were dispositioned in accordance with the ASME Code Section XI requirements. Specifically, the inspectors observed/reviewed the following examinations:

- Ultrasonic examination (UT) of a pipe-to-valve weld (weld 1-RH-1003-2), RHR;
- Magnetic particle examination (MT) of the reactor pressure vessel top head-to-flange weld (GEL-1009-AG) from 180 to 360 degrees azimuth; and
- An automated phased array UT of reactor core shroud weld H6.

The inspectors reviewed an examination with recordable indications that was accepted for continued service to verify that the licensee's acceptance was in accordance with the ASME Code or an NRC approved alternative. Specifically, the inspectors reviewed the following record:

- The inspectors reviewed L1R10 UT record (Data Report Number 1R10-70) of a feedwater (FW) valve-to-pipe weld (1FW-1001-21) performed on February 13, 2004. A recordable indication was dispositioned as being outside the required examination volume, however, it was evaluated and found to be acceptable per ASME Section XI, IWB-3514.

There were no pressure boundary welds, for Class 1 or 2 systems, completed by the licensee, and hence the inspectors did not perform the step of the inspection procedure that verifies that the welding process, and welding examinations were performed in accordance with ASME Code requirements, or an NRC approved alternative.

The inspectors performed a review of ISI related problems that were identified by the licensee, and entered into the corrective action program. Additionally, the inspectors' review included confirmation that the licensee had an appropriate threshold for identifying issues, and had implemented effective corrective actions. The inspectors evaluated the threshold for identifying issues through interviews with licensee staff, and review of licensee actions to incorporate lessons learned from industry issues related to the ISI program. The inspectors performed these reviews to ensure compliance with 10 CFR Part 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.

The reviews as discussed above constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

The inspectors observed a training crew during an evaluated simulator scenario and reviewed licensed operator performance in mitigating the consequences of events. The scenario included multiple failures which resulted in a loss of coolant accident (LOCA), reactor scram, and an Alert declaration. 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.

The inspectors' observation of this simulator scenario constituted one 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 a 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 potential impacted system work practices, reliability, or common cause failures:

- Unit 1 and Unit 2 ECCS room cooling service water system

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 reviewed, and current equipment performance status.

This maintenance effectiveness review 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

The inspectors reviewed and observed emergent work, preventive maintenance, or planning for risk significant maintenance activities. The inspectors observed maintenance or planning for the following activities or risk significant systems undergoing scheduled or emergent maintenance:

- Unit 1 Division 1 CSCS pump room floor plug removal and reinstallation;
- Unit 2 load drop activities on January 7, 2006;
- Unit 1 Division 3 battery online cell replacement and cell jumper installation to maintain battery availability; and
- Various Unit 1 issues associated with plant start up from refueling outage L1R11.

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.

The inspectors' reviews of these issues constituted four inspection samples.

b. Findings

No findings of significance were identified.

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

.1 Scheduled Unit 2 Load Drops For Maintenance and Surveillance Activities

a. Inspection Scope

The inspectors performed several hours of control room observation to evaluate operator performance during two planned Unit 2 power reductions, which were performed to facilitate several minor maintenance/repair activities, periodic surveillances, and rod pattern adjustments. The non-routine evolutions were conducted during the weekends of January 7-8, 2006, and February 11-12, 2006.

The inspectors reviewed operator logs and plant computer data to determine how the unit responded and to verify that operator actions were appropriate and consistent with operator training and plant procedures. The licensee's troubleshooting, repair strategy, planned recovery actions, procedures, reactivity manipulation briefings, and contingency plans were also reviewed by the inspectors to identify any personnel performance issues. In addition, the inspectors verified that any problems encountered during these non-routine evolutions were identified by the licensee, and appropriately entered into the corrective action program.

The observation of these non-routine evolutions by the inspectors constituted two inspection samples.

b. Findings

No findings of significance were identified.

.2 <u>February 20, 2006, Unit 1 Reactor Scram with Site Area Emergency</u>

a. <u>Inspection Scope</u>

As part of the event response documented in Section 4OA3 of this report, inspectors conducted multiple hours of direct control room observations in order to evaluate operator performance during a complicated Unit 1 reactor scram with a declared Site Area Emergency.

The inspectors reviewed operator logs and plant computer data to determine how the unit responded and to verify that operator actions were appropriate and consistent with operator training and plant procedures. The licensee's troubleshooting, repair strategy, planned recovery actions, procedures, reactivity manipulation briefings, and contingency plans were also reviewed by the inspectors to identify any personnel performance issues. In addition, the inspectors verified that any problems encountered during the periods of control room observation associated with this event were identified by the licensee, and appropriately entered into their corrective action program.

The observations associated with this complex reactor scram and Site Area Emergency declaration constituted a single inspection sample.

b. <u>Findings</u>

<u>Introduction</u>

Inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V, for the lack of written procedures and instructions related to the Technical Specification administrative control of a primary containment isolation valve (PCIV).

<u>Description</u>

After terminating the Site Area Emergency on February 20, 2006, plant operators elected to use the RCIC system to assist with the cooldown of the Unit 1 reactor in preparation for its scheduled refueling. Several months earlier, the RCIC barometric condenser vacuum pump return line to the suppression pool had been isolated to comply with plant Technical Specifications. Recurring performance problems associated with the 1E51-F028 PCIV, a plug-style check valve, had resulted in the licensee maintaining the companion 1E51-F069 PCIV, a motor-operated valve, closed and deenergized as required by Technical Specification 3.6.1.3. However, Note 1 associated with Technical Specification 3.6.1.3 allowed for the unisolation of the containment penetration "intermittently under administrative controls," and provided the means by which plant operators could routinely operate the RCIC system and still comply with the requisite PCIV Technical Specification. Further, the Bases of Technical Specification 3.6.1.3 stated that, "These controls consist of stationing a dedicated operator at the controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated."

In preparation for the use of RCIC on February 20, 2006, operations personnel established a dedicated operator for the 1E51-F069 PCIV and logged that individual by name in the Unit 1 control room log. Inspectors conducting a review of plant logs identified that following shift turnover on the evening of February 20, 2006, operators failed to denote what operator on the Unit 1 control room crew had assumed the 1E51-F069 dedicated operator duties. Further inspection revealed that licensee operations management and supervision had not established any written direction or instructions for the dedicated operator in any form. Specifically, the inspectors found that there was no written guidance as to how the valve would be administratively controlled, when and under what conditions it could or should be opened, when or under what conditions it should be closed, or any written guidance or assessment as to how long it could be "intermittently" opened as discussed in Note 1 to Technical Specification 3.6.1.3. When questioned by the inspectors regarding these issues, licensee management indicated that although the lack of any formal written guidance for administratively controlling the 1E51-F069 PCIV did not meet station expectations, they believed that the Technical Specification 3.6.1.3 requirements of Note 1 had always been met in that the dedicated operator for 1E51-F069 was stationed and had been briefed on his responsibilities by operating shift supervisors.

Analysis

The inspectors determined that there was a licensee performance deficiency associated with this issue. Specifically, the inspectors identified that the licensee had failed to establish and maintain written instructions or procedures governing the administrative control of the 1E51-F069 PCIV. The finding was determined to be more than minor in that it directly affected the configuration control and procedure quality attributes of the Barrier Integrity cornerstone (containment) and affected the cornerstone objective of providing reasonable assurance that physical design barriers (i.e., containment) protect the public from radionuclide releases caused by accidents or events.

The inspectors determined that the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," and conducted a Phase 1 characterization and initial screening. Because the finding did not represent a degradation of the radiological barrier function provided for the control room, auxiliary building, reactor building, or the SBGT system, and did not represent a degradation of the smoke or toxic gas barrier function for the control room, and did not represent an actual open pathway in the physical integrity of the primary containment or involve an actual reduction in defense-in-depth for the atmospheric pressure control or hydrogen control functions of the primary containment, the inspectors determined it to be of very low safety significance (Green) and within the licensee's response band.

Enforcement

Table 3.2-1 of the licensee's Updated Final Safety Analysis Report (UFSAR) indicates that the 1E51-F069 PCIV is subject to the requirements of 10 CFR 50, Appendix B. Criterion V, "Instructions, Procedures, and Drawings," of this appendix states, in part, that: "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." Contrary to this requirement, the licensee failed to provide any documented instructions or procedures for the administrative control of the 1E51-F069 PCIV while it was energized and subject to the requirements of Note 1 of Technical Specification 3.6.1.3.

The licensee had entered this issue into their corrective action program as IR 475214. Corrective actions planned by the licensee included development of a formal process for using administrative controls to meet Technical Specification requirements. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion V, is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2006003-02)

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Low cooling water flow through the Unit 1 HPCS room cooler;
- Failure of the Unit 2 number 4 Main turbine control valve (TCV) to exhibit fast closure during surveillance testing;
- OE 04-006, Revision 4, CSCS pump room ventilation;
- Evaluation of CSCS system operability during installation of temporary mechanical line stops;
- Questions associated with reactor pressure vessel (RPV) minimum debris retention injection rate (MDRIR); and
- Secondary containment operability issues with tarpaulins installed over crane bay openings on the reactor refueling floor.

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

The inspectors' review of these operability evaluations and issues constituted six inspection samples.

b. Findings

No findings of significance were identified. One URI was identified.

During the licensee's Unit 1 L1R11 refueling outage, a large scale manual valve replacement project was undertaken for the CSCS Division 1 subsystem. Portions of the project affected both units due to the common nature of some of the Division 1 piping. The licensee had requested and received a Technical Specification amendment from the NRC staff to facilitate the work. One of the provisions associated with the granting of this amendment was that the licensee would, throughout the entire valve replacement project, maintain the seismic qualification of the CSCS piping systems.

From March 4, 2006, through March 10, 2006, the licensee worked the portion of the project designated to replace the 1&2 DG032 manual valves. These valves were the last isolations for each unit's Division 1 room cooler CSCS discharge piping, and required the installation of temporary line stop plugs to prevent lake water from running back through the CSCS discharge piping and flooding out the lower elevations of both reactor buildings. Additionally, temporary seismic supports were required in the work packages for each valve's replacement, since the CSCS discharge piping was no longer fully supported once each line was severed to accomplish the replacement of the valves.

On March 10, 2006, a licensee work control supervisor identified that the required temporary seismic supports had not been installed during the replacement of 1&2 DG032 during work package closeout. Since by this point, the 1&2 DG032 valves had been replaced and the CSCS piping restored to normal, there was no immediate operability issue and no Technical Specification required actions were entered. The licensee did enter the issue into their corrective action program as IR 464917, and performed an evaluation for past operability. On March 11, 2006, the licensee made an 8-hour non-emergency report (ENS 42405) notifying the NRC Operations Center of the existence of the potentially unanalyzed condition for both units.

The licensee retracted their previous ENS 42405 notification on March 14, 2006, after having completed calculations that indicated that the CSCS piping remained seismically supported even without the requisite temporary supports installed. This issue is currently considered unresolved, pending completion of the review of the licensee's structural and seismic calculations by NRC Region III engineering inspectors.
(URI 05000373/2006003-03; 05000374/2006003-03)

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed the following 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 or introduce any new failure mechanisms.

- Upgrade of wide range reactor level indication to mitigate ringing (EC 347739); and
- Division 2 RHR service water orifice 1E12-D304B resizing (EC 347065).

The inspectors considered the design adequacy of the 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 the following post-maintenance activities for review. Activities were selected based upon the structure, system, or component's ability to impact risk:

- Unit 1 Division 3 125 Vdc battery testing following replacement of cell 38;
- Unit 1 Division 1 125 Vdc battery charger 1DC09E testing following charger maintenance;
- Unit 1 main steam line high flow main steam isolation valve (MSIV) isolation switch 1E-31-N008 testing after replacement;
- Unit 1 'A' RHR heat exchanger outlet valve leakage testing after repair;
- Unit 1 turbine stop valve number 4 testing after limit switch replacement;
- Unit 2 scram time testing following hydraulic control unit (HCU) maintenance;
- Testing following replacement of 1A RHRSW pump valves;
- Testing following replacement of 1B RHRSW pump valves;
- Common emergency diesel generator (EDG) testing following L1R11 outage maintenance activities; and
- Unit 1 primary plant ASME Code Class 1 pressure test following vessel reassembly from refueling outage L1R11.

The inspectors verified by witnessing the test or reviewing the test data that post-maintenance testing activities were adequate for the above maintenance and repairs. 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, repair, and post-maintenance testing activities adequately ensured that the equipment met the licensing basis, Technical Specifications, and UFSAR design requirements.

The inspectors' review of these post maintenance testing activities constituted ten inspection samples.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated outage activities for the Unit 1 L1R11 refueling outage that began on February 20, 2006, and ended on March 18, 2006. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

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

These outage inspection activities constituted a single refueling outage inspection sample.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 General Surveillance Tests

a. Inspection Scope

The inspectors selected the following general 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:

- Monthly test run of the 2A EDG;
- Unit 1 Division 2 125 Vdc 1DC17E battery charger capacity test; and
- Unit 1 ECCS injection check valve testing.

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.

The review of these general surveillance testing activities by the inspectors constituted three inspection samples.

b. Findings

No findings of significance were identified.

.2 Inservice Testing (IST) Required by the ASME Operations and Maintenance (OM) Code

a. Inspection Scope

The inspectors selected the following ASME OM Code pump IST activity for review. This IST was selected due to the risk significance of the system, HPCS, which represented over a 2 percent contribution to the unit's baseline core damage frequency.

- Unit 2 quarterly HPCS pump IST

The inspectors observed the performance of the test, including reviews for preconditioning, applicability of acceptance criteria, test equipment calibration and control, procedural use, documentation of test data, Technical Specification applicability, compliance with 10 CFR 50.55a, "Codes and Standards," impact of testing relative to performance indicator reporting, and evaluation of the test data.

The review of this IST quarterly pump surveillance constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.3 Containment Isolation Valve (CIV) Local Leak Rate Testing (LLRT)

a. Inspection Scope

The following LLRT activities required by 10 CFR 50, Appendix J, were selected by the inspectors for review. These LLRT activities were performed as part of the licensee's L1R11 refueling outage work:

- Unit 1 RHR CIV Type C LLRTs; and
- Unit 1 RCIC F028 Type C LLRT following valve replacement.

The inspectors observed the performance of LLRTs, including reviews for preconditioning, integration of the testing activities, applicability of acceptance criteria, test equipment calibration and control, procedural use, documentation of test data, Technical Specification applicability, compliance with 10 CFR 50, Appendix J, and evaluation of the test data.

The review of these LLRTs by the inspectors constituted two inspection samples.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary modification “Temporary Power Temporary Configuration Change Procedure (TCCP) to Support De-Energizing Switchgear 142Y/ Temporary Power TCCP to Support De-Energizing Switchgear 142X.” 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.

The review of this temporary modification by the inspectors constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

Inspectors observed an emergency preparedness drill to evaluate drill conduct and the adequacy of the licensee’s critique of performance to identify weaknesses and deficiencies. The inspectors selected a scenario that the licensee had scheduled as providing input to the Drill/Exercise Performance Indicator. The inspector observed, when applicable, the classification of events, notifications to off-site agencies, protective action recommendation development, and drill critiques. Inspector observations were compared to the licensee’s observations and corrective action program entries. The inspectors verified that there were no discrepancies between observed performance and performance indicator reported statistics. The scenario observed resulted in an unusual event, alert, and site area emergency classifications.

This drill evaluation constituted one inspection sample.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas and airborne radioactivity areas in the plant and reviewed work packages which included associated licensee controls and surveys of these areas to determine if radiological controls including surveys, postings and barricades were acceptable:

- Drywell;
- Low pressure heater bay; and
- Refuel floor.

The inspectors walked down and surveyed (using a NRC survey meter) these two areas to determine if the prescribed RWP, procedure, and engineering controls were in place, that licensee surveys and postings were complete and accurate, and that air samplers were properly located.

The inspectors reviewed the RWPs and work packages used to access these two areas and other high radiation work areas to identify the work control instructions and control barriers that had been specified. Electronic dosimeter alarm set points for both integrated dose and dose rate were evaluated for conformity with survey indications and plant policy. Workers were interviewed to determine if they were aware of the actions required when their electronic dosimeters noticeably malfunctioned or alarmed.

These plant walkdowns and radiation work permit reviews constituted three inspection samples.

b. Findings

No findings of significance were identified.

.2 Problem Identification and Resolution

a. Inspection Scope

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

The inspectors reviewed 15 corrective action reports related to access controls and one high radiation area radiological incident (non-PIs identified by the licensee in high radiation areas <1R/hr). Staff members were interviewed and corrective action documents were reviewed to determine if 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 effective corrective actions;
- Resolution of NCVs tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

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 and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

These problem and resolution reviews constituted three inspection samples.

b. Findings

No findings of significance were identified.

.3 Job-In-Progress Reviews

a. Inspection Scope

The inspectors observed the following three 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:

- Drywell control rod drive exchange;
- Refuel floor primary containment head removal; and
- Low pressure heater bay maintenance.

The inspectors reviewed radiological job requirements for these three activities including RWP requirements and work procedure requirements, and attended ALARA job briefings.

Job performance was observed with respect to these requirements to determine the radiological conditions in the work area were adequately communicated to workers through pre-job briefings and postings. The inspectors also verified the adequacy of radiological controls including required radiation, contamination, and airborne surveys for system breaches; radiation protection job coverage which included audio and visual surveillance for remote job coverage; and contamination controls.

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to determine if licensee controls were adequate. These work areas involved areas where the dose rate gradients were severe which increased the necessity of providing multiple dosimeters and/or enhanced job controls.

These job-in-progress reviews constituted three inspection samples.

b. Findings

No findings of significance were identified.

.4 Radiation Worker Performance

a. Inspection Scope

During job performance observations, the inspectors evaluated radiation worker performance with respect to stated radiation protection work requirements and evaluated whether workers were aware of the significant radiological conditions in their workplace, the RWP controls and limits in place, and that their performance had accounted for the level of radiological hazards present.

The inspectors reviewed radiological problem reports that 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. The problems included a January 26, 2006, incident, which involved an entry into a high radiation area on the wrong RWP. These problems, along with planned and taken corrective actions, were discussed with the Radiation Protection Manager.

These reviews of radiation worker performance constituted two inspection samples.

b. Findings

Introduction:

A Green self-revealed finding and associated NCV were identified when an Instrument Maintenance Technician logged onto a surveillance and calibration activities RWP and entered the assigned work area in the RCA, contrary to the licensee's procedures. The area was the 1A RHR room, and was posted as a HRA. The event was self-revealed when the individual's electronic dosimeter alarmed briefly as he entered a 103.5 millirem/hour dose field, and was identified upon exiting the RCA and logging out of the RWP.

Description:

On January 26, 2006, two Instrument Maintenance Technicians were assigned to work in the 1A RHR room. The assigned workers proceeded to the radiation protection desk to sign onto the RWP 1000-5652, "L1R11 Pre-outage Activities." This RWP contained proper controls for the assigned activity in a HRA. The workers received the required HRA radiation protection briefing.

This was the first time the workers had used this RWP, so they electronically signed the RWP prior to proceeding to the electronic dosimeter station and logging onto the RWP for RCA entry. Following the radiation protection HRA briefing, the workers returned to the shop for completion of the maintenance specific pre-job briefing. Both workers then logged onto the RWP and activated their electronic dosimeters. One technician proceeded to the RCA entrance to stage equipment and the other returned to the instrument shop to retrieve paperwork authorization for the work package. One of the technicians then logged out of the RWP prior to lunch.

Following lunch, the technician that had logged out of the RWP logged back in and improperly selected RWP 1000-6284, "Surveillance/Cals/EQ Checks," which did not authorize HRA access. Both technicians then proceeded to the 1A RHR room and into the HRA.

When the technician exited the 1A RHR room and checked out of the RCA, he received a warning on the computer screen that he had received a dose rate alarm during the entry. He immediately notified radiation protection (RP) staff of the warning. The RP staff investigated the event and identified that he had signed the wrong RWP.

The individual entered a maximum dose rate field as measured by the electronic dosimeter of 103.5 millirem/hour for 3 seconds and received a total dose of 9 millirem during the entry into the RCA.

The failure through self-checking to select the appropriate RWP that includes specification of radiation dose rates in the immediate work area and other appropriate radiation protection equipment and measures is contrary to RP-LA-2101, "Operation and Use of the Siemens ED-MK2 Electronic Dosimeter."

The licensee's initial prompt investigation determined the cause to be a failure of human performance error prevention techniques. Specifically, the electrician failed to adequately self-check and exhibited complacency in the acknowledgment of the access screen statements that are presented as barriers to incorrect RWP selection. As immediate corrective actions, the individual was locked out of the station's RCA and removed from duty pending completion of the investigation. A site communication was distributed by management reinforcing the expectations for self-checking when obtaining an electronic dosimeter. Senior plant management conducted a stand-down with all the first line supervision. First line supervisors were then required to peer check the electronic dosimeter login process for all high radiation area entries by their assigned workers for a period of time following the event.

Analysis:

The inspectors determined that the performance deficiency associated with this event was failure to follow established written procedures and instructions. Specifically, the individual did not electronically sign the correct RWP. The issue, under the occupational radiation safety cornerstone, does not involve the application of traditional enforcement because it did not result in actual safety consequences or the potential to impact the NRC's regulatory function, and was not the result of any willful actions. The inspectors determined that the issue was of more than minor significance, as it could be reasonably viewed as a precursor to a more significant event. The finding was associated with the occupational radiation safety cornerstone program/process attribute and affected the cornerstone objective to ensure adequate protection of worker health and safety from exposure to radiation.

The finding was evaluated using the SDP for the occupational radiation safety cornerstone and was determined to be of very low safety significance (Green), and within the licensee's response band. The finding did not involve an ALARA issue, as collective dose was not an issue. Additionally, the individual's radiation exposure was low relative to regulatory limits; there was not a substantial potential for a worker overexposure; nor was the licensee's ability to assess worker dose compromised.

The inspectors determined that the primary cause for finding was related to the cross-cutting area of human performance, specifically, procedure use and adherence.

Enforcement:

Technical Specification 5.4.1.a. requires the licensee to establish, implement, and maintain procedures recommended by Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Procedure RP-LA-2101, "Operation and Use of the Siemens ED-MK2 Electronic Dosimeter," Step 4.3.4, requires the radiation worker to select the appropriate RWP during routine issuance of a dosimeter through the RCA access control system.

Contrary to the above, on January 26, 2006, an Instrument Maintenance Technician selected the wrong RWP during routine issuance of a dosimeter and received a dose rate alarm when the worker entered a HRA in the 1A RHR room. Since the finding is of very low safety significance and had been entered into the corrective action system as IR 446416, the associated violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000373/2006003-04).

.5 Radiation Protection Technician Proficiency

a. Inspection Scope

During job performance observations, the inspectors evaluated Radiation Protection Technician (RPT) performance with respect to radiation protection work requirements and evaluated 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 their training and qualifications with respect to the radiological hazards and work activities.

This review of RPT proficiency constituted a single inspection sample.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning And Controls (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 four work activities which were likely to result in the highest personnel collective exposures:

- Drywell under vessel sump activities;
- Drywell reactor recirculation pump seal activities;
- Drywell insulation activities; and
- Tendon Inspection/repair refuel floor cavity and tendon tunnel activities.

The inspectors reviewed procedures associated with maintaining occupational exposures ALARA and processes used to estimate and track work activity specific exposures.

These reviews constituted three inspection 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 four work activities of highest exposure significance:

- Drywell under vessel sump activities;
- Drywell reactor recirculation pump seal activities;
- Drywell insulation activities; and
- Tendon Inspection/repair refuel floor cavity and tendon tunnel activities.

For these four activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements in order to determine if 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.

These radiological work planning reviews constituted two inspection samples.

b. Findings

No findings of significance were identified.

.3 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

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 by the inspectors constituted a single inspection sample.

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:

- Drywell control rod drive exchange;
- Drywell under vessel sump activities;
- Drywell Reactor recirculation pump seal activities;
- Refuel floor primary containment head removal; and
- Low pressure heater bay maintenance.

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

This job site inspection and ALARA control review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance

a. Inspection Scope

Radiation worker and RPT 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 radiation worker performance review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Occupational Radiation Safety, and Public Radiation Safety

.1 Data Submission

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 4th Quarter 2005 performance indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This review was performed as part of the inspectors' normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

As part of the various baseline inspection procedures conducted during the 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: the 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.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program (CAP) Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews did not constitute any additional inspection samples. Instead, by procedure they were considered part of the inspectors' daily plant status monitoring activities.

b. Findings

No findings of significance were identified.

.3 Selected Issue Follow-up Inspection – (Closed) Unresolved Item 05000373/2004005-01; 05000374/2004005-01: RHR Pump Seal Operability with Elevated Seal Water Temperatures.

Introduction:

The following concerns were identified during a biennial heat sink performance inspection in 2004:

- A non-conservative coolant water temperature of 250 degrees Fahrenheit (F) at 2 gallons per minute (gpm) was being used in calculations for cooling the RHR mechanical seals;
- The 2 gpm coolant flow to the seals (stated in the first concern) could be as low as 0.3 gpm; and
- The RHR seals were original unbalanced seals subject to expected shorter seal life when used at elevated temperatures.

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

a. Prioritization and Evaluation of Issues

(1) Inspection Scope

In reviewing the licensee's CAP entries and actions associated with this issue, the inspectors considered the licensee's evaluation and disposition of performance issues, evaluation and disposition of operability issues, and application of risk insights for prioritization of issues.

(2) Issues

The inspectors met with licensee engineering personnel and pump seal technical experts of the John Crane Company on January 18, 2006. The licensee and vendor representatives provided additional information regarding pump seal testing. The results of these tests led to their conclusion that the seal is suitable for running continuously in demineralized water at temperatures up to 250 degrees F. At 250 degrees F, the vapor pressure margin is approximately 185 psi, which is more than sufficient to minimize flashing in the seals. The second concern was addressed by a John Crane calculation that indicted that 0.3 gpm coolant flow was more than adequate to cool the seals. The third concern was addressed by a series of tests that were run totaling over 10,000 hours of dynamic operation. These tests were run both at steady state conditions and then exposing the seal to both dynamic and static conditions, along with temperature and pressure transients. The goal of a minimum of 2 years of seal life was met during this testing.

b. Findings

No findings of significance or violations of regulatory requirements were identified by the inspectors.

4OA3 Event Follow-up (71153)

Cornerstones: Initiating events, Mitigating Systems, and Emergency Preparedness

.1 Unit 1 Reactor Scram and Site Area Emergency on February 20, 2006

a. Inspection Scope

Inspectors responded to the station following a reactor scram and declaration of a Site Area Emergency on Unit 1 in the early morning hours on Presidents' Day, February 20, 2006. While in the process of shutting down to begin a scheduled L1R11 refueling outage, Unit 1 experienced a transient and subsequent scram as a result of a perturbation in the main turbine electric-hydraulic control (EHC) system at 00:23 a.m. A failure in one of two EHC negative 22 Vdc power supplies caused a temporary step change in reactor set pressure that caused all five main turbine bypass valves to go fully open. The resulting reactor water level and pressure transients with the reactor in the run mode at approximately 6 percent power resulted in a reactor scram and Group 1 (Main Steam Isolation Valves) containment isolation. For several hours following the scram, plant operators were unable to verify that all control rods had inserted into the core as designed and declared a Site Area Emergency in accordance with the station's emergency plan.

In response to the event, the inspectors observed plant parameters and status, including mitigating systems and fission product barriers; evaluated the performance of mitigating systems and licensee actions; and confirmed that the licensee properly reported the event as required by 10 CFR 50.72. The inspectors remained on station in the site's control room and Technical Support Center providing independent assessment and communication to NRC managers in the Region III Incident Response Center and Headquarters Operations Center until after 04:27 a.m., when the licensee had completed determinations that the reactor was shut down and the Site Area Emergency could be terminated.

The inspectors' response to this reactor scram and Site Area Emergency constituted a single inspection sample.

b. Findings

No findings of significance were identified. However, as a result of the complications associated with this scram, the event was determined to meet the criteria within NRC Management Directive 8.3, "NRC Incident Investigation Program," for transition to a special inspection due to the occurrence of a significant operational event that involved repetitive failures or events involving safety-related equipment or deficiencies in operations. The special inspection was conducted using NRC IP 93812, “Special Inspection,” and IP 71153, “Event Followup.” See NRC Inspection Report 05000373/2006009 (ADAMS Accession No. ML060820574), dated March 23, 2006, for additional details.

4OA5 Other

.1 Implementation of Temporary Instruction (TI) 2515/165 - Operational Readiness of Offsite Power and Impact on Plant Risk

The objective of TI 2515/165, “Operational Readiness of Offsite Power and Impact on Plant Risk,” was to confirm, through inspections and interviews, the operational readiness of offsite power systems in accordance with NRC requirements. On March 27 through 30, 2006, the inspectors reviewed licensee procedures and discussed the attributes identified in TI 2515/165 with licensee personnel. In accordance with the requirements of TI 2515/165, the inspectors evaluated the licensee’s operating procedures used to assure the functionality/operability of the offsite power system, as well as, the risk assessment, emergent work, and/or grid reliability procedures used to assess the operability and readiness of the offsite power system.

The information gathered while completing this TI was forwarded to the Office of Nuclear Reactor Regulation for further review and evaluation.

The performance of this TI by the inspectors represented a single inspection sample.

.2 (Closed) Unresolved Item 05000373/2005005-02; 05000374/2005005-02: Two Partial Shifts Added Together to Meet the Requirement for a Shift Specified in 10 CFR 55.53(e) for License Re-activation.

During an inspection of the Licensed Operator Requalification Training (LORT) program (IP 71111.11B), NRC inspectors identified that the licensee appeared to be in violation of 10 CFR 55.53(e) in that the requirements of 10 CFR 55.53(f) had not been correctly completed prior to resuming watchstanding duties by one of the station’s Senior Reactor Operators Limited (LSRO) to fuel handling. In accordance with 10 CFR 55.53(f), to re-activate an inactive LSRO license, the licensed operator shall actively perform the functions of an LSRO for one shift. The station’s procedures state that an LSRO must stand one 8-hour shift to reactivate their license. An LSRO was recently re-activated by standing two shifts, one shift of 6.25 hours and a second shift the following day of 1.75 hours. The two partial shifts were added together to obtain one 8-hour shift. The question as to whether the licensee can add two partial shifts together to meet the requirement for a shift specified in 10 CFR 55.53(e) for license re-activation was an unresolved item until further review by the NRC. Additionally, at the time of re-activation, the station was standing 12-hour shifts rather than the 8-hour shift required for reactivation by station procedures. Upon further review by the NRC, it was determined that it was reasonable for the station to accept an 8-hour shift versus a 12-hour shift as their minimum requirement for an LSRO. Also, because the LSROs normally rotate in 2-hour intervals (with repetitive turnovers) it appears that adding two partial shifts together under these conditions was acceptable and met the requirement for LSRO license re-activation at the LaSalle County Station. Unresolved Item 05000373/2005005-02; 05000374/2005005-02 is considered closed with no violation of NRC requirements for this issue.

.3 (Closed) White Finding and Associated Notice of Violation 05000373/2005010-01; 05000374/2005010-01: Failure to Maintain Required Design Redundancy Against a Single Failure Involving Safety-Related 4160 Vac Division 1 and Division 2 Bus Metering Circuitry.

A supplemental inspection was performed by the NRC in accordance with IP 95001 to follow up on a White inspection finding related to a single point vulnerability (SPV) in the 4160 Vac safety-related electrical system common metering circuit. A design deficiency in a metering circuit for the site's normal 4160 Vac offsite power supply induced a vulnerability whereby a single fault in the common metering circuitry, for a given unit, could have resulted in the loss of all Division 1 and Division 2 safety-related 4160 Vac power.

The supplemental inspection assessed the licensee's root and contributing cause evaluations, extent of condition and extent of cause, and completed and proposed corrective actions relating to the SPV of the 4160 Vac system common metering circuitry. Based on the results of this supplemental inspection, the NRC staff concluded that the licensee had performed a comprehensive root cause evaluation and developed corrective actions to address the concerns associated with SPV. Consistent with the guidance in NRC IMC 0305, the NRC staff classified the White finding as an "Old Design Issue." See NRC Inspection Report 05000373/2006002; 05000374/2006002 (ADAMS Accession No. ML060440293), dated February 10, 2006, for additional details.

.4 Compliance with Confirmatory Order EA-04-170, dated November 22, 2005

a. Inspection Scope

The inspectors initiated review of compliance with NRC Confirmatory Order EA-04-170, dated November 22, 2005. The Order has requirements for specific activities by the licensee during the two refueling outages following the date of the Order. The inspectors reviewed the licensee actions to determine compliance.

The inspectors reviewed LaSalle station procedures and training material and attended dynamic learning activity training sessions, to assure that the licensee:

- Revised initial radiation worker training material to highlight HRA entry requirements and consequences for the radiation worker if requirements were not met;
- Revised RWP instructions that allow HRA entry to state, "High radiation entry brief required;"
- Added warnings to worker acknowledgments on the computer screen during the access control electronic dosimetry log-in process;

- Added the radiation protection aid for conducting HRA briefings; and
- Required a signature from transient refueling outage workers prior to issuance of dosimetry that acknowledges their understanding of HRA entry requirements and the consequences for violating them.

The inspectors observed licensee activities associated with the first outage since the date of the Confirmatory Order to assure that:

- During the first 10 days, or longer as necessary, of the L1R11 refueling outage, LaSalle had greeters at primary access points to the radiologically controlled area to enhance awareness of radiological controls;
- For the L1R11 outage, all transient refueling outage workers, except as specifically authorized by the Radiation Protection Manager, were required to attend and pass a dynamic learning activity on proper HRA entry; and
- LaSalle was committed to perform an industry benchmark evaluation of HRA controls, and evaluate changes to existing practices prior to the next refueling outage.

The inspectors reviewed the corrective actions outlined in Exelon's letter dated December 17, 2004, to assure that the licensee’s contractor, Venture, revised its operating procedures to further assure compliance with HRA entry requirements. The inspectors verified through review of selected records and observations that:

- A discussion of pertinent radiological practices were conducted at each Venture daily shift brief during L1R11;
- Venture employees who worked in radiation areas read, understood, and signed a pledge to attest to his/her commitment to follow all radiological requirements and that each pledge was co-signed by the Venture site manager, project superintendent, or site ALARA coordinator, and were retained for future audit during a period of at least 1 year;
- Venture superintendents were present at select pre-job briefs involving HRA entries; and
- Venture participated in Exelon Radiation Protection Manager peer group meetings at least once prior to the L1R11 outage, and had plans for a semi-annual evaluation with the resultant commitment to take necessary action on radiation protection issues.

The inspectors reviewed the Exelon Corporate audit of Confirmatory Order action implementation to assure that Exelon conducted a review of the implementation of its and Venture's corrective actions covered in the Order. The inspectors verified that the review was conducted by knowledgeable individuals independent of the LaSalle facility.

The inspectors reviewed records of management meetings and attended a LaSalle Plant Manager meeting with contract leadership, specifically first line supervisors, prior to their access to the plant and start of contract work to assure that during the L1R11 outage the plant management clearly established personnel expectations in following radiological work requirements.

These Confirmatory Order reviews by the inspectors did not constitute any distinct inspection samples. Rather they were an integral part of the refueling and radiation protection inspections documented elsewhere in this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Ms. Susan Landahl, and other members of licensee management on April 6, 2006. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- A scheduled baseline refueling outage ISI engineering inspection with the Site Vice President, Ms. Susan Landahl, on March 2, 2006.

- A scheduled baseline refueling outage radiation protection inspection with the Site Vice President, Ms. Susan Landahl, on March 3, 2006.

- Biennial Operator Requalification Program Inspection with Mr. L. Blunk, Operations Support Manager, on March 30, 2006.

4OA7 Licensee-Identified Violation

Cornerstone: Barrier Integrity

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.

- Criterion V of 10 CFR 50, Appendix B, “Instructions, Procedures, and Drawings,” states, in part, that: “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.” Contrary to this requirement, on February 28, 2006, plant operators directed to clear status tags and install fuses for Unit 1 PCIVs associated with 0PLC9J HRSS (High Radiation Sample System) Panel inadvertently cleared the status tags and installed the fuses for the companion PCIVs for Unit 2. The error was discovered by the licensee on March 4, 2006, while licensee personnel were preparing for an outage related surveillance test for the Unit 1 PCIVs.

Although Unit 2 was at power at the time and the PCIVs associated with the HRSS panel were to be tagged out and deenergized to maintain administrative controls, the valves were always closed and no open pathway through the primary containment was ever established as a result of the error. As a result, the violation was determined to be of very low safety significance. The licensee had entered this issue into their corrective action program as IR 461998. Corrective actions by the licensee included immediately replacing the status tags and removing the fuses for the Unit 2 PCIVs, suspension of all operations equipment operators from concurrent verification activities, pending completion of remedial training, and a planned apparent or common cause evaluation in accordance with the licensee's CAP.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Landahl, Site Vice President
D. Enright, Plant Manager
J. Bashor, Site Engineering Director
R. Bassett, Emergency Preparedness Manager
L. Blunk, Operations Support Manager
T. Connor, Maintenance Director
L. Coyle, Operations Director
R. Ebright, Site Training Director
F. Gogliotti, System Engineering Manager
B. Kapellas, Radiation Protection Manager
A. Kochis, ISI Coordinator
H. Madronero, Engineering Programs Manager
S. Marik, Shift Operations Superintendent
J. Rappeport, Nuclear Oversight Manager (Acting)
D. Rhodes, Work Management Director
T. Simpkin, Regulatory Assurance Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission

B. Burgess, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000373/2006003-01; 05000374/2006003-01	URI	RHR Heat Exchanger Thermal Performance Testing and NRC GL 89-13 Conformance Issues (Section 1R07)
05000373/2006003-02	NCV	Failure to Establish and Maintain Written Procedures and Instructions for the Technical Specification Administrative Control of Primary Containment Isolation Valve 1E51-F069 (Section 1R14.2)
05000373/2006003-03; 05000374/2006003-03	URI	Operability and Calculational Issues Associated with the Failure to Install Procedurally Required Temporary Seismic Supports During L1R11 CSCS Valve Replacement Work (Section 1R15)
05000373/2006003-04	NCV	Instrument Maintenance Technician Enters a High Radiation Area on the Wrong RWP (Section 2OS1.4)

Closed

05000373/2006003-02	NCV	Failure to Establish and Maintain Written Procedures and Instructions for the Technical Specification Administrative Control of Primary Containment Isolation Valve 1E51-F069 (Section 1R14.2)
05000373/2006003-04	NCV	Instrument Maintenance Technician Enters a High Radiation Area on the Wrong RWP (Section 2OS1.4)
05000373/2004005-01; 05000374/2004005-01	URI	RHR Pump Seal Operability with Elevated Seal Water Temperatures (Section 4OA2.3)
05000373/2005005-02; 05000374/2005005-02	URI	Two Partial Shifts Added Together to Meet the Requirement for a Shift Specified in 10 CFR 55.53(e) for License Re-activation (Section 4OA5.2)
05000373/2005010-01; 05000374/2005010-01	VIO	Failure to Maintain Required Design Redundancy Against a Single Failure Involving Safety-Related 4160 Vac Division 1 and Division 2 Bus Metering Circuitry (Section 4OA5.3)

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

1R05 Fire Protection

LaSalle County Station - Fire Protection Report (FPR)

Procedures:
- OP-AA-201-008; Pre-Fire Plans; Revision 1
- LMS-FP-05; Annual Inspection, Maintenance, and Weight Check of Portable fire Extinguishers; Revision 23

Table T3.7.m-1; LaSalle Technical Requirements Manual; Revision 2

1R07 Heat Sink Performance

Issue Reports:
- 458571; Results of 1A RHR Thermal Performance Test Indeterminate; 2/21/2006

Procedures:
- LTS-200-17; RHR Heat Exchanger Thermal Performance Monitoring; Revision 8

1R08 Inservice Inspection Activities

Issue Reports:
- 458950; Jet Pump 12 Main Wedge Wear; No Change; 2/26/2006
- 458993; Jet Pump 13 Main Wedge Wear; No Change; 2/26/2006
- 457474; Stud Nut on the North Rod from Can to the Clamp Was Loose; 2/24/2006

Procedures and Other Documents:
- GE ROP-002; Reactor and Field Services Operating Experience Program; January 2005
- GE ROP-004; Briefings and Shift Turnover; January 2005
- GE-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds; July 2005
- GE-MT-100; Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle, Fluorescent); September 2004

1R11 Licensed Operator Requalification Program

ESG-69; Dynamic Simulator Scenario Guide; Revision 0

Procedures:
- LGA-001; RPV Control; Revision 6
- LGA-002; Secondary Containment Control; Revision 3
- LGA-003; Primary Containment Control; Revision 5
- LGP-3-2; Reactor Scram; Revision 51

1R12 Maintenance Effectiveness

Engineering Documents:
- EC 3494192; VY Area Cooler Throttle Valves Locking Device Equivalent Change; Revision 0
- EC 359017; Physical Restraint of Cubical Cooler Outlet Throttle Valve Position; Revision 0

Issue Reports:
- 325231; U1 SW Cubicle Area Cooler Flow Low During LOS-DG-Q3 Att A5; 4/15/2005
- 442006; Low Flow on Cooler 2VY02A During LOS-DG-Q3; 1/13/2006
- 451664; Low Flow Issue on Unit-1 Division III VY Cooler, 1VY02A; 1/13/2006
- 455276; Mechanical Locking Device Broken; 2/17/2006
- 455471; Could not Complete LOS-DG-SR6; 2/23/2006

1R13 Maintenance Risk Assessments and Emergent Work Control

Procedures:
- LEP-DC-114; Installing Jumper Around Cell in Division 1,2 or 3 125 Volt Battery; Revision 4
- LAP-300-44; Floor Plug Removal and Installation; Revision 10
- LOA-COND-101; Unit 1 Reactor Water/Condensate High Conductivity; Revision 5
- OP-AA-102-101; Unit Load Changes; Revision 3
- OP-AA-300; Reactivity Management; Revision 1
- OP-AB-300-1001; BWR Control Rod Movement Requirements; Revision 2

Work Orders:
- 881923-01; Replace Degrade Cell; 1/13/2006
- 556242-22; Remove and Reinstall Floor Plugs in Unit 1 Diesel Generator Corridor; 2/7/2006

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

Procedures:
- OP-AA-102-101; Unit Load Changes; Revision 3
- OP-AA-300; Reactivity Management; Revision 1
- OP-AB-300-1001; BWR Control Rod Movement Requirements; Revision 2
- LTS-1100-4; Scram Insertion Times; Revision 26

Issue Reports:
- 475214; NRC ID – Inadequate Administrative Controls for U1 RCIC; 4/5/2006

1R15 Operability Evaluations

Operability Evaluations:
- OE 04-006; CSCS Pump Room Ventilation; Revision 4

Calculations:
- L-002313; Minimum Debris Retention Injection Rate; Revision 3

Procedures:
- LOS-RP-Q5; Turbine Control Valve Quarterly Surveillance; Revision 4
- LRP-1470-14; ALARA Engineering Contamination Control Applications; (Installation/ Removal of Refuel Floor Temporary Equipment Hatch Covers); Revision 1

Issue Reports:
- 435069; Turbine Control Valve No. 4 Did Not Exhibit Fast Acting Characteristics; 12/18/2005
- 442006; Low Flow on Cooler 2VY02A During LOS-DG-Q3; 1/13/2006
- 442964; Effect of Slow Turb. Control Valve Penalty on U2 Rod Pattern; 1/17/2006
- 451664; Low Flow Issue on Unit-1 Division III VY Cooler, 1VY02A; 1/13/2006
- 454258; Recommend Removal of MCPR Penalty for U2 TCV4; 2/20/2006
- 466702; NRC Identified - LRP-1470-14 Requirements & SBGT Operability
- 464917; Temporary Seismic Support Not Installed During 1(2)DG032 Valve Replacement; 3/11/2006

Engineering Documents:
- EC 346099; Refueling Floor Equipment Hatch Covers to Support Refueling Operations and Cavity Draindown and Decon Tasks; Revision 0
- EC 349192; VY Area Cooler Outlet Throttle Valve Locking Device Equivalent Change; Revision 0
- EC 359955; CSCS Division 1 Operability Assessment; Revision 0

Work Orders:
- 811471-01; Disassemble and Reassemble the Reactor Vessel; 2/20/2006

1R17 Permanent Plant Modifications

Engineering Documents:
- EC 347065; RHR Division 2 Service Water Orifice 1E12-D304B Resizing Due to EC 341950; Revision 1
- EC 347739; Upgrade Wide Range Reactor Level Indication to Mitigate Ringing; Revision 1

1R19 Post-Maintenance Testing

Issue Reports:
- 440093; Terminal 38 of U1 Div 3 Battery Has a Low Voltage; 1/09/2006
- 442958; NRC Identified Loose Nuts on Seismic Tie Rod on Battery Rack; 1/17/2006
- 445618; Control Room Indication on Div 1 DC Battery Charger; ½5/2006

- 446018; Unit 1 Div 1 Battery Charger Low Voltage Alarm on 1DC09E; ½6/2006
- 447775; Diff Pressure Sw 1E-31-N008 Failed to Reset; 1/20/2006

Work Orders:
- 485731-07; 1E12-F330B U-1 RHR SW Pmp 1B Valve Replacement; 3/10/2006
- 485732-07; 1E12-F330A U-1 RHR SW Pmp 1A Valve Replacement; 3/10/2006
- 659171-01; Tshoot Div 1 Batt Charger 1DC09E Percent Current Unbal Out of Tol; ½6/2006
- 704591-01; OP LOP_NB-01 Reactor Vessel Leakage Test; 3/17/2006
- 721423-01; ES DG Start and Load Acceptance - Unit 0; 3/13/2006
- 722098-01; EM Perform Div 1 Battery Charger Capacity Test Per LES-DC-1; ½4/2006
- 849144-02; OP PMT LOS-RH-Q2 Cycle, Check Local Indication and Leakage; ½3/2006
- 886460-01; U1 Div 1 Battery Charger 1DC09E Tripped During Load Test; ½6/2006

Procedures:
- LOS-DG-101; 0 Diesel Generator, 0DG01K, Start and Load Acceptance Surveillance; Revision 3
- LOS-NB-R1; Reactor Vessel Leakage Test; Revision 0
- LTS-1100-4; Scram Insertion Times; Revision 26

1R20 Outage Activities

Issue Reports:
- 455968; Three Rods Failed to Indicate Full in Following a Scram; 2/20/2006
- 456561; Unit 1 Div 1 ARI Failure Following Initiation During ATWS; 2/21/2006
- 457593; U1 Control Rod 26-15 Settled to Odd Notch 01 During ARI Test; 2/23/2006
- 457625; Braided Hose From 1B33-F372B Has Hydraulic Leak; 2/23/2006
- 458484; NRC Identified Cables on DW Floor in Contact with Fryquel; 2/24/2006
- 460068; Initial Conditions of Drywell Under Grating from Leak; 2/22/2006
- 466476; NR Cables Cleaned with Soapy Solution; 3/14/2006
- 467422; L1R11 Drywell Close Out Punch List from NRC Walkdown; 3/16/2006
- 470496; NRC Raised Questions Regarding Fryquel & Coatings in Drywell; 3/24/2006

Engineering Documents:
- EC 343284; Evaluation of Reactor Recirculation Oil Leak on Drywell Equipment During Normal and Accident Conditions; Revision 0
- EC 360162; Technical Evaluation Regarding the Long-Term Drywell Coating Effects from Fryquel EHC Fluid Contamination; Revision 0

Procedures:
- LOP-AA-03; Reactor Mode Changes; Revision 20
- LGP-1-1; Normal Unit Startup; Revision 75
- LGP-1-S1; Master Startup Checklist; Revision 56
- LOP-RM-01; Reactor Manual Control Operation; Revision 28
- LOP-DW-02; Drywell Entry and Inspection (Shutdown, Startup, or Operation); Revision 13
- OP-AB-300-1001; BWR Control Rod Movement Requirements; Revision 3

1R22 Surveillance Testing

Drawings:
- 1E-1-4008AK; Division 2 125V DC Battery Main Charger 1BA (1DC17E)
- 1E-1-4000DB; 125 Volt DC Distribution System

Issue Reports:
- 440584; HPCS Water Leg Pump Discharge Pressure Below Required Amount; 1/10/2006
- 87096; Wrong Instrument Used for Performance of LOS-HP-Q1; 12/12/2001

Procedures:
- LES-DC-103B; Division II Battery Charger Capacity Test; Revision 16
- LOS-HP-Q1; Unit 2 HPCS System Operability and Inservice Test; Revision 54
- LOS-DG-M2; 2A Diesel Generator Idle Start; Revision 63
- LTS-900-12; RHR Pressure Isolation Valve Water Leak Rate Test; Revision 21
- LTS-300-5; Primary Containment Leak Rate Testing Program; Revision 35
- LTS-100-2; LLRT Mass Makeup Method; Revision 30

Work Orders:
- 857097-01; OP LOS-HP-Q1 U2 HPCS Pump Run Att 2A; 1/10/2006
- 820863-03; 1E51-F028 Check Valve As-Left LLRT; 3/8/2006

1R23 Temporary Plant Modifications

Engineering Documents:
- EC 355284; Temporary Power Evaluation for Procedure LOP-AP-142X, LOP-AP-142Y, LEP-AP-102 to Support Refuel Outage L1R11; Revision 0
- EC-EVAL 359092; Evaluate Temporary Power to Support L1R11; Revision 1

Issue Reports:
- 454844; Temporary Power Cable Routing in Reactor Building; 2/12/2006

1EP6 Drill Evaluation

LaSalle EP Drill Scenario LS06-01

Procedures:
- EP-AA-1005; LaSalle Annex, Hazard Recognition Category Hazards and Other Conditions; Revision 20

2OS1 Access Control to Radiologically Significant Areas

Issue Reports:
- 376560; No Dose Rate Information on Laboratory Waste Containers; 9/22/2005
- 441923; Nuclear Oversight Observed Dynamic Learning Activity Sessions with Varied Radiation Protection Standards; 2/2/2006
- 440293; Question if Credit for Completing Dynamic Learning Activity Should be Given; 1/14/2006

- 446416; Document Prompt Investigation for Human Performance Error While Selecting RWP; 2/1/2006
- 448702; Results of Electronic Dosimeter Station Monitoring; 1/31/2006
- 455122; High Radiation Area Swing Gate Not in Accordance with RP-LA-460-1004; 12/6/2006
- 459796; Issues Identified By Radiation Protection Behavior Correction Specialist; 3/5/2006
- 458781; Unplanned Airborne Radioactivity Area; 2/25/2006
- 458984; Insulator Received Electronic Dosimeter Dose Rate Alarm; 2/26/2006
- 281312; Assess the Current Plan to Reduce Low Level Personnel Contamination Events to Within Industry Standards Focused Area Self Assessment; 12/15/2004
- 299238-05; Check Self-Assessment Report: Contractor Radiation Protection Technician Radworker Performance During L2R10; 6/30/2005
- 434170-05; Access Control to Radiologically Significant Areas and ALARA Planning and Controls Focused Area Self-Assessment; 1/13/2006
- 378502; Nuclear Oversight Identified Three Way Communication Weaknesses in an ALARA Brief; 9/27/2005
- 458466; Nuclear Oversight Identified Incomplete Review of RWPs by Workers; 2/24/2006

Procedures:
- RP-AA-403; Administration of the Radiation Work Permit Program; Revision 1
- RP-AA-460; Controls for High and Very High Radiation Areas; Revision 10
- RP-AA-460-1001; Additional High Radiation Exposure Control; Revision 0
- RP-AA-600-1005; Radioactive Material Shipment Checklist; Revision 5
- RP-LA-2101; Operation and Use of the Siemens ED-MK2 Electronic Dosimeter; Revision 5

DLA ID 5099; Radiation Worker Dynamic Learning Activity; 1/9/2006

Radiation Work Permits:
- 1000-5618; Unit 1 Insulation Activities, Excluding Drywell; Revision 0
- 1000-5626; Unit 1 Laborer Routines, Excluding Drywell; Revision 0

2OS2 As Low As Is Reasonably Achievable Planning And Controls

Procedures:
- RP-AA-401; Operational ALARA Planning and Controls; Revision 5
- RP-LA-401-1003; Radiation Protection Briefing Requirements; Revision 1
- RP-AA-401; Work-In-Progress Review; Revision 5

Radiation Work Permits:
- RWP 1000-5654; L1R11 Tendon Inspection /Repair Refuel Floor, Cavity, and Tendon Tunnel; Revision 1
- RWP 1000-5577; L1R11 Drywell Insulation Activities; Revision 0
- RWP 1000-5581; L1R11 Drywell Reactor Recirculation Pump Seal Activities; Revision 0
- RWP 1000-5592; L1R11 Drywell Under Vessel Sump Activities; Revision 0

4OA2 Identification and Resolution of Problems

Issue Reports:
- 280803; RHR Pump Seal Cooling Water Maximum Temperature; 12/8/2004
- 280881; RHRSW Flow to RHR Seal Cooler Calculation Needs Revision; 12/9/2004
- 134065; 2A RHR Pump Seal Cooler Flow Rate Lower Than Acceptance Criteria; 12/4/2002

Memorandums:
- LaSalle Heat Sink Inspection RHR Pump Seal Temperature; from Daniel Schmit; Attached John Crane Memo from R. Gabriel; 12/20/2004
- John Crane 3-1/2" Type 8B1 Seal CFSP-38327-8; 12/17/2004

4OA3 Event Follow-up

Procedures:
- LGA-01; Reactor Pressure Vessel Control; Revision 6
- LGA-10; Failure to Scram; Revision 6
- LGA-NB-01; Alternate Rod Insertion; Revision 7
- LGP-3-2; Reactor Scram; Revision 55
- LOP-RW-01; Rod Worth Minimizer Initialization and Operation; Revision 15
- LOP-RW-02; RWM Error Messages and Corrective Actions; Revision 10
- LOS-RD-SR7; Channel Interference Monitoring; Revision 4
- EP-AA-1000; Exelon Nuclear Standardized Radiological Emergency Plan; Revision 16
- EP-AA-1005; Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station; Revision 20

Control Room Logs and Records:
- SRM Count Rate Data for SRMs A-D; 0045 on 2/20/2006 to 0050 on 2/20/2006
- LaSalle Unit 1 Control Room Operator Logs; 0000 to 2359 on 2/20/2006
- LaSalle Unit 1 Wide Range Reactor Pressure; 0000 to 0728 on 2/20/2006
- LaSalle Unit 1 SPDS Reactor Power; 0000 to 0024 on 2/20/2006
- LaSalle Unit 1 Narrow Range Reactor Water Level; 0000 to 0125 on 2/20/2006

Issue Reports:
- 459764; Rod 34-47 Difficult to Move; 2/28/2006
- 461103; Simulator RWM Scram Capture Mode Limitations; 3/2/2006
- 461346; Control Rod Drive 38-43 Needs to be Removed for Analysis; 3/3/2006
- 458939; Signs of Excessive Friction for 4 Rods After Shutdown of L1C11; 2/26/2006
- 462261; Aggregate Review IR Not Written on Control Rod 38-43 Scram Time Degradation; 3/5/2006
- 456066; NOS Identifies ATWS Mitigation Issues; 2/20/2006
- 455968; Three Rods Failed to Indicate Full In Following a Scram; 2/20/2006
- 462570; LGA-NB-01 Actions Would Have No Effect on Rod; 3/6/2006
- 465107; Historical Issue: RWM Scram Capture Modification 50.59 Error; 3/11/2006

Other Miscellaneous Documents:
- Post Transient Review Report for the 2/20/2006 LaSalle Unit 1 Scram; 2/21/2006

- NUMARC/NESP-007; Methodology for Development of Emergency Action Levels; Revision 2

4OA5 Other

Procedures:
- LOA-AP-101; Unit 1, AC Power System Abnormal; Revision 22
- LOA-GRID-001; Low Grid Voltage; Revision 6
- LOP-AP-43; Emergency Load Conservation; Revision 1
- OP-AA-108-107-1001; Station Response to Drig Capacity Conditions; Revision 1
- OP-AA-108-107-1002; Interface Agreement Between Exelon Energy Delivery and Exelon Generation for Switchyard Operations; Revision 2
- WC-AA-8000; Interface Procedure Between Exelon Energy Delivery (COMED/PECO) and Exelon Generation (Nuclear/Power) for Construction and Maintenance Activities; Revision 0

NRC Confirmatory Order EA-04-170; 11/22/2005

Memorandum From John L. Schrage: Independent Review of Commitments from Confirmatory Order Concerning High Radiation Area Access Controls; 2/24/2006

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CR	Condition Report
CSCS	Core Standby Cooling System
CY	Calendar Year
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EHC	Electric-Hydraulic Control
ENS	Event Notification System
F	Fahrenheit
FW	Feedwater
GL	Generic Letter
gpm	Gallons Per Minute
HCU	Hydraulic Control Unit
HPCS	High Pressure Core Spray
HRA	High Radiation Area
HRSS	High Radiation Sample System
ILT	Initial License Training
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report or Issue Report
ISI	Inservice Inspection
IST	Inservice Test
JPM	Job Performance Measure
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LORT	Licensed Operator Requalification Training
LSRO	Limited Senior Reactor Operator
MDRIR	Minimum Debris Retention Injection Rate
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
OM	Operations and Maintenance
PCIV	Primary Containment Isolation Valve
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water

RP	Radiation Protection
RPT	Radiation Protection Technician
RPV	Reactor Pressure Vessel
RWP	Radiation Work Permit
SBGT	Standby Gas Treatment
SDP	Significance Determination Process
SPV	Single Point Vulnerability
TCCP	Temporary Configuration Change Procedure
TCV	Turbine Control Valve
TI	Temporary Instruction
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Examination
Vac	Volts Alternating Current
Vdc	Volts Direct Current