April 27, 2007

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 05000373/2007002;
05000374/2007002

Dear Mr. Crane:

On March 31, 2007, 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 10, 2007, 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, two NRC-identified and three self-revealed findings of very low safety significance were identified. Four of these findings identified 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 NRC's Enforcement Policy. Additionally, two licensee identified violations are 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.

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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 05000373/2007002; 05000374/2007002
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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Vice President - Licensing and Regulatory Affairs
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Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional
Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2007002; 05000374/2007002
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, Illinois
Dates:	January 1, 2007, through March 31, 2007
Inspectors:	D. Kimble, Senior Resident Inspector F. Ramírez, Resident Inspector M. Holmberg, Engineering Inspector M. Mitchell, Health Physicist C. Moore, Operator Licensing Examiner B. Palagi, Senior Operator Licensing Examiner D. Reeser, Operator Licensing Examiner N. Shah, Project Engineer S. Sheldon, Reactor Engineer J. Tapp, 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 05000373/2007002, 05000374/2007002; 01/01/2007 - 03/31/2007; LaSalle County Station, Units 1 & 2; Equipment Alignment, Inservice Inspection Activities, Refueling and Other Outage Activities, and Event Follow-up Report.

The inspection was conducted by resident inspectors and regional inspectors. The report covers a 3-month period of resident inspection, and announced baseline inspections of the inservice inspection program and radiation protection program. Five Green findings and four associated non-cited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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 NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealing finding of very low safety significance was identified following the removal of a safety tag out and valve realignment for the 2A instrument nitrogen (IN) compressor. Specifically, operations personnel were restoring the system valve lineup following maintenance and placed one valve, 2IN073, into the closed position when it should have been left open, which resulted in an unplanned loss of IN system header pressure. A non-cited violation of Technical Specification 5.4.1.a was also identified for failure to follow the required steps for component restoration following the removal of a safety tag out as outlined in the licensee’s procedures.

  The performance deficiency associated with this finding was the failure on the part of plant operators to follow the provisions of their procedure for equipment clearance orders and safety tagging. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Specifically, the inspectors determined that the licensee’s failure to properly realign the Unit 2 IN system following maintenance created an unnecessary challenge to control room personnel, who were forced to use an abnormal operating procedure to maintain Unit 2 IN system header pressure to avoid unplanned and unintended valve actuations. Because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available, the inspectors concluded that the finding was of very low safety significance and within the licensee’s response band. In addition, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance since personnel work practices did not support human performance in that the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel did not follow procedures. Corrective actions planned and completed by the licensee included coaching and counseling of the operators involved and a next shift communication message to all operators on the incident and preliminary cause. (Section 1R04.2)

- Green. A self-revealing finding of very low safety significance was identified following the inadvertent initiation of the Division 1 emergency core cooling system (ECCS) on Unit 2 during reactor vessel nozzle flushing from the refuel floor for radiation dose reduction. Specifically, licensee work planning personnel did not recognize the potential adverse impact on ECCS instrumentation taps from using a high-pressure flushing wand to clean out reactor vessel nozzles, and failed to provide personnel performing the flushing activities with adequate procedural instructions. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances.

  The performance deficiency associated with this finding involved inadequate work planning and written instructions for the reactor vessel nozzle flushing activities. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Because the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since the licensee did not appropriately plan work activities consistent with nuclear safety and failed to incorporate risk insights in accordance with the work activity being performed. Corrective actions planned and completed by the licensee included halting all reactor vessel nozzle flushing operations until an initial investigation into the event was performed and conducting a full root cause analysis for the event. (Section 4OA3.1)

- Green. A self-revealing finding of very low safety significance was identified following the inadvertent initiation of Unit 2 Division 2 ECCS, which occurred when operators started shutdown cooling (SDC) while reactor coolant system was pressurized. Specifically, adequate procedural instructions were not provided and as such, control room personnel did not recognize the potential consequences associated with initiating SDC with a pressurized reactor coolant system. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances.

  The performance deficiency associated with this finding involved Unit 2 control room personnel not properly or thoroughly reviewing actions associated with starting SDC with the reactor vessel water system solid and pressurized prior to their performance. The finding was determined to be of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety. Because the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since the control room personnel did not use conservative assumptions in decision-making and as such, did not identify the possible unintended consequences of their actions. Corrective actions planned and completed by the licensee included performing an initial investigation into the event, performing an engineering analysis of system impact and conducting a full root cause analysis for the event. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated non-cited violation of 10 CFR 50.55a(g)4 for the licensee's failure to perform examinations of the ASME Code Section XI required weld volume for the Unit 1 and 2 'B' residual heat removal (RHR) heat exchanger shell welds. Specifically, the licensee completed only ⅓ of the Code required weld examination volume for four shell welds on each heat exchanger vessel.

  The performance deficiency associated with this finding was the failure of the licensee to complete a full volumetric examination of the 1B and 2B RHR heat exchanger shell welds. This finding was of more than minor significance because it directly affected the Mitigating System Cornerstone objective of equipment performance (reliability). Because the finding did not represent a design or qualification deficiency that resulted in the loss of operability the inspectors concluded that it was of very low safety significance and within the licensee's response band. In addition, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance, since the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported. Corrective actions planned and completed by the licensee included repeating the 'B' RHR heat exchanger shell weld examinations to ensure the required Code volume was covered. (Section 1R08)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance was identified by the inspectors during review of the licensee's activities associated with de-tensioning the drywell head in preparation for scheduled reactor refueling operations. Specifically, the inspectors identified that the licensee had not performed a current Technical Specification required Type 'B' local leak rate test (LLRT) with half of the drywell head closure bolts de-tensioned, such that when they performed the de-tensioning activity in Mode 3 the surveillance requirement was no longer met. Because the licensee took action in response to the inspectors' questions and completed a Type 'B' LLRT on the drywell head with half of the closure bolts de-tensioned within the allowed outage time provided in the Technical

Specifications, no violation of regulatory requirements was identified in conjunction with the finding.

The performance deficiency associated with this finding was the licensee's failure to recognize the impact on the Technical Specifications from this activity until questioned by the inspectors. The finding was determined to be of more than minor significance in that if left uncorrected it would have represented a more significant safety concern. Specifically, absent NRC intervention, the licensee would have not performed a Type ‘B' LLRT within the Technical Specification action statement time limit and a Technical Specification violation would have resulted. Because the finding involved adequate mitigation capability, did not impact primary containment availability, and was not an event that could be characterized as a loss of control, the inspectors concluded that it was of very low safety significance and within the licensee’s response band. In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance since licensee personnel did not use conservative assumptions in decision-making and as such, did not identify the possible unintended consequences their actions. Corrective actions planned and completed by the licensee included the performance of an apparent cause evaluation, and actions for the licensee outage organization to flag any departures from normal practices and discuss these items at weekly pre-outage planning meetings. Other corrective actions included the performance of a 10 CFR 50.59 screening and/or evaluation to support the change to the reactor vessel disassembly procedure allowing the partial de-tensioning of the drywell head in Mode 3, and an action to evaluate potential changes to procedure LTS-100-15. (Section 1R20)

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

Violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee’s corrective action program. These violations and 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. On February 17, 2007, power was reduced to approximately 71 percent to facilitate a control rod sequence exchange and various surveillances. The unit was restored to full power the next day 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 11, 2007, the unit commenced coast down for refueling outage L2R11. The reactor was shut down for this refueling outage on February 26, 2007. Unit 2 Cycle 12 began with initial reactor criticality on March 16, 2007, and full power operation was achieved on March 20, 2007. On March 22, 2007, power was reduced to approximately 72 percent to permit a control rod pattern adjustment, which was required due to initial cycle build-up of fission product poisons. The unit returned to operation at full power later that same day, and continued to operate 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)

.1 Quarterly Partial System Alignment Verifications

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 2 high pressure core spray (HPCS) system; and
- 2A RHR system.

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

These partial equipment alignments constituted two inspection samples.

b. Findings

No findings of significance were identified.

.2 Unit 2 Instrument Nitrogen System Valve Alignment Error

a. Inspection Scope

On January 20, 2007, the licensee responded to various control room alarms that indicated a potential loss of Unit 2 IN system header pressure. Due to the IN system's risk significance as an event initiator, in the aftermath of this event the inspectors performed a partial walkdown of the Unit 2 IN system to verify operability and proper equipment lineup. The inspectors verified the position of critical redundant equipment and looked for any discrepancies between the existing equipment lineup and the required lineup.

In addition, the inspectors reviewed the licensee's issue report (IR) and other corrective action program (CAP) documents related to the event.

This partial equipment alignment constituted a single inspection sample.

b. Findings

Introduction

A self-revealing finding of very low safety significance (Green) was identified following the removal of a safety tag out and valve realignment for the 2A IN compressor. Specifically, operations personnel restoring the system valve lineup following maintenance placed one valve, 2IN073, into the closed position when it should have been left open, which resulted in an unplanned loss of IN system header pressure. A non-cited violation of Technical Specification 5.4.1.a was also identified for failure to follow the required steps for component restoration following the removal of a safety tag out as outlined in the licensee's procedures.

Description

Late on January 19, 2007, licensee maintenance personnel were completing work activities associated with the 2A IN compressor. Plant operators were informed by maintenance personnel that the safety tag out for the 2A IN compressor could be lifted, and the compressor and associated IN system valves returned to a normal line up.

On-Shift plant operators drew up a restoration line up for the 2A IN compressor based solely upon an approved plant piping drawing, which showed the 2IN073 valve (air dryer cross-connect stop) in the closed position. The operators involved did not refer to either the mechanical system checklist in the approved IN system operating procedure, or the position information for the components to be manipulated contained in the licensee's plant equipment computer database. Had they done so, the operators would have found that the normal position for the 2IN073 valve was open, not closed.

Early on the midnight shift on January 20, 2007, plant equipment operators cleared the tag out on the 2A IN compressor. The operators were in the process of restoring the system to a normal line up when the control room received alarms 2PM13J-A404 – R0605, “Instrument Nitrogen System Trouble – Instrument Nitrogen Dryer ‘A’ Switching Failure,” and 2PM13J-A404 – R0604, “Instrument Nitrogen System Trouble – Nitrogen Drywell Inlet Pressure Low.” Control room operators entered the IN system abnormal operating procedure, LOA-IN-201, “Loss of Drywell Pneumatic Air Supply,” as directed by their alarm response procedures, and cross-connected the Unit 2 IN system to the station’s instrument air header. This action stabilized the pressure in the Unit 2 IN system supply header, and prevented any unintended repositioning of valves in the Unit 2 drywell.

Operators investigated the condition and subsequently identified that the 2IN073 valve had been closed in error in accordance with the prior tag out restoration activities. The 2IN073 valve was opened at the direction of on-shift operations supervision and the 2B IN compressor subsequently restored system pressure. Operators then verified the IN system equipment alignment and closed the cross-connects to the station instrument air system to exit the abnormal operating procedure.

Analysis

The inspectors determined that there was a performance deficiency associated with the licensee’s tag out restoration activities. Specifically, the licensee’s procedure for equipment clearance orders and safety tagging, OP-AA-109-101, “Clearance and Tagging,” requires that established system operating procedures be used for tag out and tag out restoration whenever such procedures are available. Contrary to this requirement, the plant operators who drew up the tag out restoration line up for the 2A IN compressor and associated valves on January 19, 2007, used only a plant drawing in which to base the restoration valve positions. This error resulted in the loss of Unit 2 IN system header pressure, associated control room alarms, the entry into an abnormal operating procedure for the Unit 2 IN system, and the need to cross-connect the Unit 2 IN system with the station’s instrument air header.

In accordance with NRC IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” the inspectors determined that the finding was of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety, which is “to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.” Specifically, the inspectors determined that the licensee’s failure to have properly realigned the Unit 2 IN system following maintenance created an unnecessary challenge to control room personnel, who were forced to use an abnormal operating procedure to maintain Unit 2 IN system header pressure to avoid unplanned and unintended valve actuations.

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. Using the criteria for transient initiators, the inspectors determined that the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. This reasoning was based on the fact that the only mitigation equipment supported by the IN system, the 13 pneumatically-operated safety-relief valves located in the drywell, have a dedicated safety-related nitrogen bottle bank supply to provide actuating gas in the event of the loss of the normal IN compressor system. As a result, the inspectors concluded that the finding was of very low safety significance (Green) and within the licensee's response band.

In addition, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, "Operating Reactor Assessment Program," since personnel work practices did not support human performance in that the licensee failed to define and effectively communicate expectations regarding procedural compliance and personnel did not follow procedures.

Enforcement

Technical Specification 5.4.1 states that: "Written procedures shall be established, implemented, and maintained covering the following activities: a) The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Section 1.c of Appendix A to Regulatory Guide 1.33 requires "Equipment control (e.g., locking and tagging)" to be included within the licensee's required set of written procedures. Paragraph 7.1.13 of the licensee's established procedure for equipment tag outs, OP-AA-109-101, "Clearance and Tagging," states, in part: "When a procedure exists to secure or restore equipment to service, then the procedure shall be used to perform the task."

Contrary to this requirement, on January 19, 2007, licensee operators preparing a tag out restoration line up for the 2A IN compressor and associated valves utilized only a plant drawing to establish the restoration valve positions, and specified the incorrect position for valve 2IN073. A checklist denoting normal valve positions was available as part of the IN system normal operating procedure, and it did list the proper normal position for this valve. However, this checklist was not utilized.

The licensee entered this issue into their CAP as IRs 581287 and 581543. Corrective actions planned and completed by the licensee included coaching and counseling of the operators involved and a next shift communication message to all operators on the incident and preliminary cause. Because the licensee has entered the issue into their corrective action program and the finding is of very low safety significance, this violation of Technical Specification 5.4.1.a is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy.
(NCV 05000374/2007002-01)

1R05 Fire Protection (71111.05)

.1 Quarterly Fire Protection Zone Inspections

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 4D1, Unit 1 - cable spreading room, elevation 749'0";
- Fire Zone 4D2, Unit 2 - cable spreading room, elevation 749'0";
- Fire Zone 4D4, Unit 2 - electrical equipment room, elevation 749'0";
- Fire Zone 4E2, Unit 2 - auxiliary equipment room, elevation 731'0";
- Fire Zone 4E3, Unit 1 - Division 2 essential switchgear room, elevation 731'0";
- Fire Zone 4E4, Unit 2 - Division 2 essential switchgear room, elevation 731'0";
- Fire Zone 5B13, balance-of-plant cable zone, elevation 731'0";
- Fire Zone 5D4, heater drain tank zone, elevation 687'0";
- Fire Zone 6D, elevation 687'0";
- Fire Zone 8C3, Unit 2 - HPCS Diesel Pump room, elevation 673'0";
- Fire Zone 8C4, Unit 2 - Division 2 residual heat removal service water (RHRSW) pump room, elevation 674'0"; and
- Fire Zone 8C5, Unit 2 - Division 1 RHRSW pump room, elevation 674'0".

The inspectors reviewed the control of transient combustibles and ignition sources, fire detection equipment, manual suppression capabilities, passive and automatic suppression capabilities, barriers to fire propagation, and any contingency fire watches that were in effect.

These reviews constituted twelve inspection samples.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee’s testing of the Unit 2 RHR heat exchangers 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 identifying and addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee’s results of the heat exchanger’s performance test and compared it against acceptance criteria and observed the licensee’s visual inspection of the redundant heat exchanger. Additionally, the inspectors reviewed the licensee’s test results to verify that the heat exchanger would be capable of performing its safety function.

This heat exchanger performance review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) Activities (IP 71111.08)

a. Inspection Scope

From March 2 through March 6, 2007, the inspectors conducted a review of the implementation of the licensee's ISI program for monitoring degradation of the reactor coolant system boundary, and the risk significant piping system boundaries for Unit 2. 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 03 of the inspection procedure, based upon the ISI activities available for review during the onsite inspection period.

The inspectors observed ultrasonic examination (UT) of the following welds to evaluate compliance with the ASME Code Section XI requirements and to verify that indications and defects (if present) were dispositioned in accordance with the ASME Code Section XI:

- Three 24-inch diameter reactor recirculation loop 'A' austenitic pipe welds, Nos 7, 11, and 12;
- Standby liquid control (SLC) safe-end-to-nozzle (LCS-2-N11) dissimilar metal weld; and
- Reactor vessel nozzle-to-shell weld (LCS-2-N16A).

The inspectors observed dye penetrant examination of three 24-inch diameter reactor recirculation loop 'A' austenitic pipe welds Nos 7, 11, and 12 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.

The inspectors reviewed relevant indications identified during magnetic particle examinations of Class 2 piping supports RI-24-2854X and RH40-2877 and during UT of a dissimilar metal weld (1RH-2004-42A), to determine if the licensee's corrective actions and extent of condition reviews were in accordance with the ASME Code Section XI requirements.

The inspectors reviewed pressure boundary weld records for replacement of a 3-inch diameter Class 1 main steam drain isolation valve 2B21-F019 to determine if the welding acceptance and preservice examinations (e.g., pressure testing, visual, dye penetrant, and weld procedure qualification tensile tests and bend tests) were performed in accordance with ASME Code Sections III, V, IX, and XI requirements.

The inspectors performed a review of ISI related problems that were identified by the licensee and entered into the CAP, conducted interviews with licensee staff, and reviewed licensee CAP records to determine if:

- The licensee had described the scope of the ISI related problems;
- The licensee had established an appropriate threshold for identifying issues;
- The licensee had evaluated industry generic issues related to ISI and pressure boundary integrity; and
- The licensee implemented appropriate corrective actions.

The inspectors performed these reviews to ensure compliance with 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” requirements.

These ISI program reviews constituted a single inspection sample.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) and an associated NCV of 10 CFR 50.55a(g)4 for the licensee’s failure to perform examinations of the ASME Code Section XI required weld volume for the Unit 1 and 2 ‘B’ RHR heat exchanger shell welds. Specifically, the licensee completed only ⅓ of the Code required weld examination volume for four shell welds on each heat exchanger vessel.

Description

On March 4, 2007, the inspectors identified that the licensee had not performed UT of the required weld volume for shell welds on the 1B and 2B RHR heat exchangers.

The ASME Code Section XI required the licensee to perform periodic volumetric examinations of the RHR heat exchanger welds as defined by Figure IWC-2500-1. This figure defined a weld volume which included the full through-wall thickness of the vessel wall to a distance of ½ inch on either side of these welds. During UT of the 1B RHR heat exchanger shell welds in January of 2002 and January of 2004, the licensee’s vendor performed UT of only the inner ⅓ of the shell through-wall thickness. During UT of the 2B RHR heat exchanger shell welds in January of 2003 and February of 2005 the licensee’s vendor performed UT of only the inner ⅓ of the shell through-wall thickness. These examinations were recorded on the licensee’s examination summary sheets, which contained wall thickness profile sheets with weld examination sketches. Each of the weld examination sketches included a dashed line depicting the extent of weld volume covered by the UT transducers, which included only the inner ⅓ of the shell through-wall thickness. The licensee and Authorized Nuclear Inservice Inspector reviewed and approved the wall thickness profile sheets for each of these examinations.

The licensee’s vendor had conducted the RHR heat exchanger vessel shell weld examinations in accordance with Procedure GE-PDI-UT-1, “PDI Generic Procedure For the Ultrasonic Examination of Ferritic Pipe Welds,” which had been developed based upon an industry procedure intended for examination of piping welds. For piping welds, the ASME Code required only the inner ⅓ of the through-wall thickness to be examined. However, the licensee's vendor had applied this procedure to vessel welds, which required full through-wall thickness examinations. Revisions 1 and 2 of GE-PDI-UT-1 contained explicit requirements for examination of the full weld volume for vessel welds as required by the Code, and Revision 3 of GE-PDI-UT-1 omitted the explicit description of the required vessel weld examination volume. Because Revisions 1 through 3 of GE-PDI-UT-1 had been used for these examinations, the inspectors could not attribute the licensee's vendor staff failure to complete the full weld examination volumes to inadequate procedure guidance. Therefore, the inspectors concluded that the examination coverage error was in part, due to insufficient licensee oversight of the vendor staff during UT of these RHR heat exchanger welds.

The licensee entered this issue into their CAP as IR 599201, and concluded that the Unit 1 and Unit 2 systems were operable based on Code compliant examinations of the 'B' RHR heat exchangers completed in 1996. The licensee staff planned to repeat the 'B' RHR heat exchanger shell weld examinations to ensure the required Code volume was covered.

Analysis

The inspectors determined that the failure of the licensee to complete a full volumetric examination of the 1B and 2B RHR heat exchanger shell welds was a performance deficiency that warranted a significance evaluation. This finding was of more than minor significance because absent NRC intervention, the licensee would have relied on an incomplete UT of the 'B' RHR heat exchanger welds for an indefinite period of service, which would have placed this mitigating system at increased risk for undetected cracking, leakage, or component failure. This increased risk of failure for the RHR heat exchangers directly affected the Mitigating System Cornerstone objective of equipment performance (reliability).

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. In this case, the licensee had prior inspection information confirming lack of flaws and based upon industry operating experience, service related degradation would not likely initiate from the uninspected weld locations. As a result, the inspectors determined that the finding did not represent a design or qualification deficiency that resulted in the loss of operability and determined it to be of very low safety significance (Green) and within the licensee's response band.

Because the licensee had numerous opportunities during review and acceptance of the vendor's UT records to identify the incomplete weld examination volumes and failed to do so, the inspectors also determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, "Operating Reactor Assessment Program." Specifically, the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported.

Enforcement

Title 10 CFR 50.55a(g)4 requires, in part, that throughout the service life of a boiling or pressurized water reactor facility, components classified as ASME Code Class 1, 2, and 3 must meet requirements of Section XI. The 1989 Edition, of ASME Code Section XI, Article IWC-2500(a), requires that components be examined and tested as specified in Table IWB-2500-1. Examination Category C-A of this table, "Pressure Retaining Welds in Pressure Vessels," requires, volumetric (e.g., radiographic or UT) examination of the weld volume defined by Figure IWC-2500-1 in the Code, which includes the entire through-wall thickness of the vessel weld.

Contrary to the above, during UT of the 1B and 2B RHR heat exchanger shell welds (Examination Category C-A) that occurred in January of 2002, January of 2003, January of 2004, and February of 2005, the UT did not include the entire through-wall thickness. Specifically, for four Unit 1 RHR heat exchanger shell welds (examination reports Nos. 1R9-133, 1R9-134, 1R10-18, and 1R10-19), and four Unit 2 RHR heat exchanger shell welds (examination reports Nos. 2R9-057, 2R9-058, 2R10-101, and 2R10-102) only the inner ⅓ of the shell through-wall thickness was volumetrically examined. Because of the very low safety significance of this finding and because the issue was entered into the licensee's CAP as IR 599201, this violation of 10 CFR 50.55a(g)4 is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000373/2007002-02; 05000374/2007002-02).

1R11 Licensed Operator Requalification Program (71111.11)

.1 Quarterly Resident Inspector Observation of Licensed Operator Training

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 equipment and instrumentation failures, and the transient resulted in a complex loss of coolant accident. 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, and group dynamics. Additionally, the inspectors observed the post-scenario critiques performed by both the simulator instructor staff evaluating the crew, and the training crew themselves.

This simulator training observation constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.2 Annual Operating Test Results and Biennial Written Examination Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the written, operating and simulator tests, which are required to be given annually per 10 CFR 55.59(a)(2), and were administered by the licensee from September 12 through October 19, 2006. The overall results were compared with the SDP in accordance with NRC Manual Chapter 0609, Appendix I, "Operator Requalification Human Performance Significance Determination Process."

These written examination and operating test results constituted a partial inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the licensee's handling of performance issues and the associated implementation of the Maintenance Rule (10 CFR 50.65) to evaluate maintenance effectiveness for the selected systems. The Unit 1 and Unit 2 circulating water (CW) systems were selected based on being designated as risk significant under the Maintenance Rule and due to recent reliability issues involving the Unit 1 CW pumps.

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.

These reviews 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, and planning for risk significant maintenance activities. The following activities or risk significant systems undergoing scheduled or emergent maintenance were included:

- 2B SLC pump emergent work;
- Unit 2 reactor core isolation cooling (RCIC) work window;
- Unit 1 leading edge ultrasonic feedwater flow meter and correction factor;
- Unit 2 refuel outage L2R11 shutdown safety assessment review; and
- Unit 2 refuel outage L2R11 lost parts evaluations and assessments.

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.

These reviews constituted five inspection samples.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- A revision to the formal operability evaluation for the Unit 1 and Unit 2 RCIC systems' high steam line temperature leak detection instrumentation;
- A formal operability evaluation for increased indicated flow through Unit 1 jet pump no. 19;
- Assessment of operability following the identification of an out-of-position valve associated with the Unit 1 main turbine fire protection deluge system;
- Assessment of operability associated with a planned control room ventilation envelope boundary breech to facilitate a plant modification;
- Assessment of operability associated with an alarming condition on the Unit 1 best estimate power monitor computer point;

- Assessment of operability associated with a high temperature reading noted in the Unit 2 main steam tunnel during surveillance testing; and
- Review of a non-conservative technical specification associated with the Unit 1 and Unit 2 automatic depressurization system safety relief valves and associated instrument nitrogen gas supply.

These 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 seven inspection samples.

b. Findings

No findings of significance were identified.

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 modification. The inspectors also verified that the modifications did not place the plant in an unsafe configuration.

- Unit 2 main turbine digital electro-hydraulic control; and
- Unit 2 core standby cooling system (CSCS) valve replacements.

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.

The inspectors' review of these permanent plant modifications 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.

- 2B SLC pump testing following emergent rebuild work;
- 1B emergency diesel generator (EDG) testing following a scheduled work window;
- 1B SLC pump testing following pump seal replacement;
- Testing following maintenance on valve 2FC-046B;
- Testing following replacement of valve 2DG-117;
- Testing following replacement of valve 2DG-007;
- Testing following work related to valve 2C41-F004B;
- Testing following work related to valve 2C41A-S002;
- Testing following maintenance on valve 2VP-197B; and
- Review of the Unit 2 refuel outage L2R11 reactor system pressure test following reactor vessel reassembly.

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

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

b. <u>Findings</u>

No findings of significance were identified.

1R20 <u>Outage Activities</u> (71111.20)

a. <u>Inspection Scope</u>

The inspectors evaluated outage activities for the Unit 2 L2R11 refueling outage that began on February 26, 2007, and ended on March 17, 2007. 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

Introduction

A finding of very low safety significance (Green) was identified by NRC inspectors during review of the licensee's activities associated with de-tensioning the drywell head in preparation for scheduled reactor refueling operations. Specifically, the inspectors identified that the licensee had not performed a current Technical Specification required Type 'B' LLRT with half of the drywell head closure bolts de-tensioned, such that when they performed the de-tensioning activity in Mode 3 the surveillance requirement was no longer met. Because the licensee took action in response to the inspectors' questions and completed a Type 'B' LLRT on the drywell head with half of the closure bolts de-tensioned within the allowed outage time provided in the Technical Specifications, no violation of regulatory requirements was identified in conjunction with the finding. An unresolved item (URI) was also identified by the inspectors associated with the licensee's 10 CFR 50.59 evaluation performed to support the procedure change to the reactor vessel disassembly procedure that permitted de-tensioning of the drywell head closure bolts with the reactor in Mode 3.

Description

On February 26, 2007, inspectors observed that the licensee had invoked a provision in procedure MA-AB-756-600, "Reactor Disassembly," under Section 4.4, "Drywell Head Removal," which permitted early de-tensioning of half (i.e., every other bolt) of the drywell head closure bolts while the reactor was still in Mode 3 and the vessel pressurized. The inspectors noted that Technical Specification 3.6.1.1 required primary containment integrity in Modes 1, 2, and 3, and that Technical Specification Surveillance Requirement 3.6.1.1.1 called out periodic LLRTs of various containment hatches and penetrations to ensure primary containment operability. As a result, the inspectors questioned the licensee as to what kind of LLRT had been most recently performed on the drywell head, and whether or not the testing had been performed with every other closure bolt de-tensioned. The licensee indicated that no LLRT of the drywell head with any closure bolts de-tensioned had been performed, or was planned.

Following several question and answer sessions with licensee personnel, the inspectors concluded that the licensee had not considered the potential impact on the Technical Specifications from this activity. The licensee had performed an engineering analysis several years earlier that indicated that the drywell head would remain functional with every other closure bolt de-tensioned, and licensee personnel were relying on that analysis as the basis for their de-tensioning activities with the reactor in Mode 3. However, the inspectors pointed out that while the engineering analysis was an acceptable vehicle for demonstrating availability and functionality of the drywell head with half of the closure bolts de-tensioned, it could not be credited in lieu of an actual LLRT to demonstrate compliance with Technical Specification Surveillance Requirement 3.6.1.1.1, and therefore the primary containment should be considered inoperable. As a result, the licensee performed a Type 'B' LLRT on the drywell head with half of the closure bolts de-tensioned on February 27, 2007, which was within the 36 hour action statement time allowed by Technical Specification 3.6.1.1.

As the inspectors continued to review the licensee's activities surrounding the early de-tensioning of half of the drywell head closure bolts, two other potential issues were identified. First, in reviewing the licensee's change to the reactor disassembly procedure that had introduced the ability to partially de-tension the drywell head in Mode 3, the inspectors identified that the licensee's procedure change had not been adequately supported by a 10 CFR 50.59 screening or evaluation. The screening the licensee had performed in accordance with 10 CFR 50.59 was intended to support numerous changes to the reactor vessel disassembly procedure, and was written in a highly generic fashion as a result. Upon closer review, the inspectors identified that the document addressed only the removal of drywell head shield blocks in Mode 3, and that the de-tensioning of drywell head closure bolts in Mode 3 was not addressed at all. At the time of this report, the licensee was still in the process of developing a 10 CFR 50.59 screening and/or evaluation to support the Mode 3 drywell head de-tensioning procedure changes. As a result, this issue is considered unresolved, pending the inspectors' receipt and review of the licensee's 10 CFR 50.59 screening and/or evaluation. (URI 05000374/2007002-03)

The second potential issue identified by the inspectors involved the actual LLRT procedure used by the licensee to conduct the Type 'B' test of the drywell head. Prerequisite B.1.1 of procedure LTS-100-15, "Type 'B' Local Leak Rate Test," required that any containment closure to be tested first be bolted into place. The inspectors challenged the licensee's use of the procedure on the drywell head with half of the closure bolts de-tensioned, and questioned the licensee as to whether or not a procedure change should have been performed prior to conducting the LLRT.

Analysis

The inspectors determined that there was a performance deficiency associated with the licensee's drywell head closure bolt de-tensioning activities with the reactor in Mode 3. Specifically, licensee personnel failed to recognize the impact on the Technical Specifications from this activity until questioned by the inspectors.

In accordance with NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that if left uncorrected it would have represented a more significant safety concern. More precisely, the inspectors determined that absent NRC intervention, the licensee would have not performed a Type 'B' LLRT within the Technical Specification action statement time limit and a Technical Specification violation would have resulted.

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 this issue occurred with shutdown cooling in operation, the inspectors utilized IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." Using Checklist 5, "Boiling Water Reactor Hot Shutdown: Time to Boil less than 2 Hours with RHR in Operation (Reactor Coolant System Pressure less than RHR Cut-in Permissive)," of Appendix G, Attachment 1, "Phase 1 Operational Checklists for Both Pressurized Water Reactors and Boiling Water Reactors," the inspectors qualitatively determined that the finding involved adequate mitigation capability and no change in primary containment availability. In addition, using Table 1, "Losses of Control," of Appendix G, the inspectors qualitatively determined that the finding was not an event that could be characterized as a loss of control. As a result, the inspectors concluded that the finding was of very low safety significance (Green) and within the licensee's response band.

In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, "Operating Reactor Assessment Program," since licensee personnel did not use conservative assumptions in decision-making, did not conduct any effectiveness reviews of their decision to partially de-tension the drywell head in Mode 3, and did not adequately review the decision for unintended consequences.

Enforcement

Because of the inspectors' intervention, the licensee performed a Type 'B' LLRT on the drywell head with half of the closure bolts de-tensioned on February 27, 2007, which was within the 36 hour action statement time allowed by Technical Specification 3.6.1.1. As a result, no violation of regulatory requirements was determined to be associated with the finding.

In addition, upon a more detailed review of LLRT procedure LTS-100-15, "Type 'B' Local Leak Rate Test," and the prerequisite that required that any containment closure to be tested first be bolted into place before testing, the inspectors determined that the licensee's failure to have enacted a procedure change prior to conducting the LLRT on the drywell head on February 27, 2007, had no material affect on the results of the test. Consequently, it was determined to constitute a minor violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," which was not subject to formal enforcement action in accordance with the NRC Enforcement Policy.

The licensee entered these issues into their CAP as IRs 596847, 597525, 601913, 601924, 601925, and 601926. Corrective actions planned and completed by the licensee included the performance of an apparent cause evaluation, actions for the licensee outage organization to flag any departures from normal practices and discuss these items at weekly pre-outage planning meetings, performance of a 10 CFR 50.59 screening and/or evaluation to support the change to the reactor vessel disassembly procedure allowing the partial de-tensioning of the drywell head in Mode 3, and an action to evaluate potential changes to procedure LTS-100-15.
(FIN 05000374/2007002-04)

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

.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 1A EDG;
- Unit 2 periodic traversing incore probe surveillance and local power range monitor calibration;
- Quarterly secondary containment damper operability test; and
- Unit 2, Division 1, EDG response time 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 four inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Based on the relatively high risk significance of the system, the inspectors selected the following Code pump IST activity for review:

- Quarterly IST for the 1A RHR pump.

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 <u>Containment Isolation Valve (CIV) Local Leak Rate Testing</u>

a. <u>Inspection Scope</u>

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 L2R11 refueling outage work:

- Unit 2 hydrogen recombiner CIV LLRTs for HG005A and HG006A; and
- Unit 2 main steam isolation valve LLRTs during L2R11.

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. <u>Findings</u>

No findings of significance were identified.

1R23 <u>Temporary Plant Modifications</u> (71111.23)

a. <u>Inspection Scope</u>

The inspectors selected the following temporary modifications for review. The inspectors reviewed the safety screening, design documents, UFSAR, and applicable Technical Specifications to determine that the temporary modifications were 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.

- Installation of an ultrasonic leading edge flow meter on the Unit 1 feedwater system to permit the calculation and use of a feedwater flow correction factor to the plant process computer thermal power calculation (TCCP 364169);
- Installation of CSCS temporary pipe supports (EC 363303); and
- Defeating All Scram Signals Except Manual (LOP-RP-05).

These reviews constituted three inspection samples.

b. <u>Findings</u>

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

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

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone PIs to determine whether or not the conditions surrounding the PIs had been evaluated and if identified problems had been entered into the CAP for resolution.

These reviews constituted a single inspection sample.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed licensee controls and surveys in the following radiologically significant work areas within radiation areas, high radiation areas (HRAs), 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 areas to determine if the prescribed RWP, procedure, and engineering controls were in place; if licensee surveys and postings were complete and accurate; and if air samplers were properly located.

The inspectors reviewed the RWPs and work packages used to access these three 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.

The inspectors reviewed RWPs for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g., filtered ventilation system operation) and to determine if there was a potential for individual worker internal exposures of greater than 50 millirem committed effective dose equivalent. There were no airborne radioactivity work areas during the inspection period. Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

These reviews constituted four inspection samples.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, licensee event reports, and special reports related to the access control program to determine if identified problems were entered into the CAP for resolution.

The inspectors reviewed 15 corrective action reports related to access controls and one HRA radiological incident (non-PIs identified by the licensee in HRAs less than 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, and 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.

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates greater than 25 R/hr at 30 centimeters or greater than 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures greater than 100 millirem total effective dose equivalent (or greater than 5 rem shallow dose equivalent or greater than 1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure. There were no PI events during the inspection period.

These reviews constituted four inspection samples.

b. Findings

No findings of significance were identified.

.4 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 HRAs for observation of work activities that presented the greatest radiological risk to workers:

- Drywell control rod drive mechanism exchange;
- Refuel floor flood-up and fuel shuffle; 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 as-low-as-reasonably-achievable (ALARA) job briefings.

Job performance was observed with respect to these requirements to determine if 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 reviews constituted three inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

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

The inspectors conducted plant walkdowns to verify the posting and locking of entrances to high dose rate HRA and very high radiation.

These reviews constituted three inspection samples.

b. Findings

No findings of significance were identified.

.6 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 four radiological problem reports which found that the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

These reviews constituted two inspection samples.

b. Findings

No findings of significance were identified.

.7 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 constituted a single inspection sample.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning And Controls (71121.02)

.1 Inspection Planning

a. Inspection Scope

The inspectors reviewed plant collective exposure history, current exposure trends, and 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 control rod drive activities;
- Drywell nozzle inspection activities;
- Drywell scaffold activities; and
- Refuel floor cavity activities.

The inspectors reviewed the site specific trends in collective exposures and source-term measurements.

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

These reviews constituted four 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 five work activities of highest exposure significance:

- Reactor vessel disassembly/reassembly activities;
- Drywell control rod drive activities;
- Drywell scaffold activities;
- Drywell nozzle inspection activities; and
- Drywell support activities.

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

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

These reviews constituted two inspection samples.

b. Findings

No findings of significance were identified.

.4 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to determine the historical trends and current status of tracked plant source terms and to determine if the licensee was making allowances and developing contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant reactor chemistry.

This 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 HRAs that presented the greatest radiological risk to workers. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice by being familiar with the work activity scope and tools to be used, by utilizing ALARA low dose waiting areas and that work activity controls were being complied with. Also, radiation worker training and skill levels were reviewed to determine if they were sufficient relative to the radiological hazards and the work involved.

This review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.6 Declared Pregnant Workers

a. Inspection Scope

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

This review constituted a single inspection sample.

b. Findings

No findings of significance were identified.

2PS2 Radioactive Material Processing and Transportation (71122.02)

.1 Radioactive Waste System

a. Inspection Scope

The inspectors reviewed the liquid and solid radioactive waste system descriptions in the UFSAR, and the 2004 and 2005 annual radioactive effluent release reports for information on the types and amounts of radioactive waste (radwaste) generated and disposed. The inspectors reviewed the scope of the licensee's audit/self-assessment activities, with regard to radioactive material processing and transportation programs to determine if those activities satisfied the requirements of 10 CFR 20.1101(c), and the quality assurance audit requirements of Appendix G to 10 CFR 20 and of 10 CFR 71.137, as applicable.

These reviews constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.2 Radioactive Waste System Walkdowns

a. Inspection Scope

The inspectors walked down portions of the liquid and solid radwaste processing systems to verify that these systems were consistent with the descriptions in the UFSAR and in the process control program, and to assess the material condition and operability of those systems. No changes were made to the radwaste processing systems in the last 2 years. The inspectors reviewed the status of radioactive waste process equipment that was not operational and/or was abandoned in place. These systems included the waste solidification/drumming equipment, the radwaste evaporator system, and the radwaste concentrates system. The inspectors discussed with the licensee the administrative and/or physical controls preventing the inadvertent use of this equipment to ensure that the equipment would not contribute to an unmonitored release path or be a source of unnecessary personnel exposure.

The inspectors reviewed the licensee's processes for transferring waste resin into shipping containers to determine if appropriate waste stream mixing and sampling was performed so as to obtain representative waste stream samples for analysis. The inspectors reviewed the licensee's practices for the collection of area smear surveys to represent the dry-active waste (DAW) stream and the methods used for determining the radionuclide mix of various filter media to ensure they were representative of the intended radwaste stream. Additionally, the inspectors reviewed the methodologies for quantifying gamma emitting radionuclide waste stream content, for determining waste stream tritium concentrations and for waste concentration averaging to ensure that representative samples of the waste products were provided for the purposes of waste classification pursuant to 10 CFR 61.55.

These reviews constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.3 Waste Characterization and Classification

a. Inspection Scope

The inspectors reviewed the licensee's methods and procedures for determining the classification of radioactive waste shipments including the use of scaling factors to quantify difficult-to-measure radionuclides. The inspectors reviewed the licensee's most recent radiochemical sample analysis results for each of the licensee's waste streams, and the associated calculations used to account for difficult-to-measure radionuclides. These waste streams consisted of radwaste demineralizer resins, various filter media, and DAW. The inspectors also reviewed the minimum detectable concentrations achieved for each waste stream as determined by the licensee's contract analytical laboratory compared to the corresponding radionuclide groupings in 10 CFR 61.55 to determine whether the concentration values satisfied the NRC Branch Technical Position on radioactive waste classification. These reviews were conducted to determine if the licensee's program assured compliance with 10 CFR 61.55 and 56, as required by Appendix G of 10 CFR 20. The inspectors also reviewed the licensee's waste characterization and classification program to determine if reactor coolant chemistry data was periodically evaluated to account for changing operational parameters that could potentially affect waste stream classification and thus validate the continued use of existing scaling factors between sample analysis updates.

These reviews constituted a single inspection sample.

b. Findings

No findings of significance were identified.

.4 Shipment Preparation and Records

a. Inspection Scope

The inspectors reviewed the documentation of shipment packaging, surveying, package labeling and marking, vehicle inspections and placarding, emergency instructions, and licensee verification of shipment readiness for six selected non-excepted radioactive material and radwaste shipments, made between March 2006 and January 2007. The shipment documentation reviewed included:

- Control rod drives as low specific activity (LSA) - II;
- Spent resin shipped as LSA-I;
- Waste sludge resin shipped as Type B;
- Fuel pool septa shipped as LSA-II;
- Spent resin shipped as Type B; and
- Tri-nuke filters shipped as Type B radioactive material.

For each shipment, the inspectors determined if the requirements of 10 CFR 20 and 61, and those of the Department of Transportation (DOT) in 49 CFR 170-189 were met. Specifically, records were reviewed, and staff involved in shipment activities were interviewed to determine if packages were labeled and marked properly, if packages and transport vehicle surveys were performed with appropriate instrumentation, and whether survey results satisfied DOT requirements, and if the quantity and type of radionuclides in each shipment were determined accurately. The inspectors also determined whether shipment manifests were completed in accordance with DOT and NRC requirements, if they included the required emergency response information, if the recipient was authorized to receive the shipment, and if shipments were tracked as required by 10 CFR 20.

The inspectors interviewed selected staff involved in shipment activities to determine if they had adequate skills to accomplish shipment related tasks, and to determine if the shippers were knowledgeable of the applicable regulations to satisfy package preparation requirements for public transport with respect to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," and 49 CFR 172, Subpart H. Also, the inspectors observed personnel conduct package preparation, and surveys on a package containing a fuel pool septa in preparation for shipment to a waste processor.

These reviews constituted two inspection samples.

b. Findings

No findings of significance were identified.

.5 Identification and Resolution of Problems for Radwaste Processing and Transportation

a. Inspection Scope

The inspectors reviewed selected condition reports, self-assessment and audit reports, along with field observation reports that addressed the radioactive waste and radioactive materials shipping program, since the last inspection to determine if the licensee had effectively implemented the corrective action program, and that problems were identified, characterized, prioritized, and corrected. The inspectors also verified that the licensee's self-assessment program was capable of identifying repetitive deficiencies, or significant individual deficiencies in problem identification and resolution.

The inspectors also selectively reviewed other CAP reports generated since the previous inspection, that dealt with the radioactive material or radwaste shipping program, and interviewed staff, and reviewed documents to determine if the following activities were being conducted in an effective and timely manner, commensurate with their importance to safety and risk:

- 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 CAP; and
- Implementation/consideration of risk significant operational experience feedback.

These reviews constituted a single inspection sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Data Submission Issue

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the 4th Quarter 2006 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)

.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 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.

4OA3 Event Follow-up (71153)

.1 Unit 2 Division 1 ECCS Actuation on March 3, 2007

a. Inspection Scope

During the Unit 2 L2R11 refueling outage on March 3, 2007, inspectors responded to an inadvertent actuation of the Unit 2 Division 1 ECCS that occurred during efforts to flush reactor vessel nozzles from the refueling floor to reduce radiation levels and personnel dose. The inspectors observed plant parameters and status; evaluated the performance of mitigating systems and licensee actions; and confirmed that the licensee properly addressed event reportability, as required by 10 CFR 50.72 and 50.73.

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

b. Findings

Introduction

A self-revealing finding of very low safety significance (Green) was identified following the inadvertent initiation of the Division 1 ECCS on Unit 2 during reactor vessel nozzle flushing from the refuel floor for radiation dose reduction. Specifically, licensee work planning personnel did not recognize the potential adverse impact on ECCS instrumentation taps from using a high-pressure flushing wand to clean out reactor vessel nozzles, and failed to provide personnel performing the flushing activities with adequate procedural instructions. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances.

Description

On the morning of March 3, 2007, Unit 2 was in refuel outage L2R11. The plant was in the refueling mode of operation with the reactor vessel head removed. Primary coolant temperature was being maintained at approximately 90 degrees Fahrenheit, with the spent fuel pool and reactor cavity gates removed allowing for the spent fuel pool to communicate with the reactor refueling cavity. On the refuel floor, licensee personnel were engaged in the flushing of reactor vessel nozzles. Nozzle flushing was a process whereby a high-pressure wand was inserted between a thermal sleeve and the vessel shell, or aimed directly at nozzles without thermal sleeves, in an attempt to blow out radioactive contamination and reduce the on-contact dose rates in the drywell. The wand delivered roughly 11 gallons per minute of water spray at about 500 psig, and was attached to an underwater handling pole.

At about 8:14 a.m., Unit 2 control room personnel received several alarms related to low reactor vessel water level, along with indications of a Division 1 ECCS initiation. As a result of the Division 1 ECCS signal the low pressure coolant injection (LPCI) discharge valve, 2E12-F042A, opened, the low pressure core spray (LPCS) pump started and began injecting into the Unit 2 reactor vessel, and the Division 1 EDG (common to both Unit 1 and Unit 2) started and began running in an unloaded condition.

Unit 2 operators in the control room responded to the event by verifying that both reactor vessel level and spent fuel pool level were normal, and then securing the ECCS equipment that had actuated. Licensee personnel then began an investigation into the cause of the event, and soon identified that personnel working on the refuel floor had been flushing various reactor penetration nozzles in an effort to reduce radiation levels in the drywell and personnel dose. A review of refuel floor logs showed that at the time of the ECCS initiation refuel floor personnel were flushing the N14A instrument nozzle, which was the reference leg tap for the Division 1 ECCS reactor vessel level instrument. From a reconstruction of the event, it was concluded that a pressure transient occurred in the reference leg sensing line for the reactor level instrument, which resulted in the Division 1 ECCS low reactor vessel water level initiation signal.

Follow-up review of the event revealed that there were no specific procedural limitations or restrictions associated with the nozzle flushing activity. And, although nozzle flushing activities had been performed during previous refueling outages, it was further identified that reactor vessel instrument line nozzles had never been flushed during previous outages. The work package instructions being utilized by personnel on the refuel floor for the flushing activity had no provisions either directing personnel to flush the reactor vessel instrument line nozzles or restricting the flushing of instrument line nozzles.

Analysis

The inspectors determined that there was a performance deficiency associated with the licensee's work planning for the reactor vessel nozzle flushing activities. Specifically, licensee work planners failed to identify potential adverse consequences associated with the high-pressure flushing of the reactor vessel instrument line nozzles, such that the work package instructions being utilized by personnel on the refuel floor for the flushing activities contained neither precautions or limitations nor specific provisions directing personnel as to how to flush the reactor vessel instrument line nozzles. This error resulted in the unnecessary and unintentional actuation of safety-related ECCS equipment.

In accordance with NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on the objective for the Initiating Events Cornerstone for Reactor Safety, which is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." Specifically, the inspectors determined that the licensee's failure to properly plan the flushing activity for the reactor vessel instrument line nozzles and their failure to provide adequate written procedural instructions for this activity created an unnecessary challenge to control room personnel.

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 this finding occurred during the refueling mode of operation, the inspectors utilized IMC 0609, Appendix G, "Shutdown

Operations Significance Determination Process." Using Checklist 7, "Boiling Water Reactor Refueling Operation with Reactor Coolant System Level above 23 Feet," of Appendix G, Attachment 1, "Phase 1 Operational Checklists for Both Pressurized Water Reactors and Boiling Water Reactors," the inspectors qualitatively determined that the finding involved adequate mitigation capability and was not an event that could be characterized as a loss of control. As a result, the inspectors concluded that the finding was of very low safety significance (Green) and within the licensee's response band.

In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, "Operating Reactor Assessment Program," since the licensee did not appropriately plan work activities consistent with nuclear safety and failed to incorporate risk insights in accordance with the work activity being performed.

Enforcement

Criterion V of 10 CFR 50, Appendix B, "Instructions, Procedures, and Drawings," states 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 March 3, 2007, licensee personnel on the Unit 2 refueling floor conducted high-pressure flushing of the reactor vessel instrument line nozzles using inadequate written work package instructions and procedures, such that an unnecessary and unintentional actuation of safety-related Division 1 ECCS equipment occurred and control room operators were subjected to a needless operational challenge.

The licensee entered this issue into their CAP as IR 598883. Corrective actions planned and completed by the licensee included halting all reactor vessel nozzle flushing operations until an initial investigation into the event was performed and conducting a full root cause analysis for the event. 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 a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000374/2007002-05)

.2 Unit 2 Division 2 ECCS Actuation on March 15, 2007

a. Inspection Scope

During the Unit 2 L2R11 refueling outage on March 15, 2007, inspectors responded to an inadvertent actuation of the Unit 2 Division 2 ECCS that occurred when operators were in the process of starting the 2B RHR pump in accordance with LOP-RH-07, "Shutdown Cooling System Startup, Operation, and Transfer," while restoring from performing a reactor coolant system test in accordance with LOS-NB-R2, "Reactor Vessel Leakage Test." The inspectors observed plant parameters and status; evaluated the performance of mitigating systems and licensee actions; and confirmed that the licensee properly addressed event reportability, as required by 10 CFR 50.72 and 50.73.

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

b. Findings

Introduction

A self-revealing finding of very low safety significance (Green) was identified following the inadvertent initiation of Division 2 ECCS on Unit 2 that occurred when operators were in the process of starting the 2B RHR pump in accordance with LOP-RH-07, "Shutdown Cooling System Startup, Operation and Transfer," while restoring from performance of a pressure test in accordance with LOS-NB-R2, "Reactor Vessel Leakage Test." Specifically, adequate procedural instructions were not provided to the operators, who invoked a procedural note that allowed for the performance of procedure steps out of sequence. In the course of performing the procedure steps out of sequence, an inadvertent ECCS actuation resulted. A non-cited violation of 10 CFR 50, Appendix B, Criterion V, was also identified for the failure to adequately prescribe documented instructions or procedures for the work activity that were appropriate to the circumstances.

Description

On the evening of March 15, 2007, Unit 2 was in refuel outage L2R11. The plant was in the cold shutdown mode of operation with the reactor vessel head in place and fully tensioned. The control room personnel had recently performed a reactor coolant system pressure test in accordance with LOS-NB-R2, "Reactor Vessel Leakage Test." This test, which is required by Technical Specifications, raises pressure in the reactor coolant system to 1040 psig for the purpose of detecting leaks in the system that might have developed as a result of refuel outage maintenance activities. After performing LOS-NB-R2, the operators were in the process of restoring the reactor coolant system pressure back to atmospheric pressure.

During restoration from LOS-NB-R2, with pressure in the reactor coolant system still at approximately 50 psig, control room operators decided to invoke a procedural note in LOS-NB-R2 that allowed for the performance of steps out of sequence. As written, the normal sequence of the procedure called for the reactor coolant system to be fully depressurized and the reactor vessel vented before SDC was to be initiated for decay heat removal. Because LOS-NB-R2 had progressed more slowly than expected, control room operators discussed initiating SDC early and eventually concluded that there would be no adverse consequences associated with performing this action out of sequence.

At about 1:02 a.m., Unit 2 control room personnel started the 2B RHR pump in SDC mode in accordance with LOP-RH-07, "Shutdown Cooling System Startup, Operation and Transfer." Almost immediately, several alarms related to low reactor vessel water level were received along with indications of Division 2 and 3 ECCS initiation signals. As a result of the Division 2 and 3 ECCS initiation signals, the 2C RHR pump started and injected water into the reactor vessel and, the Division 2 EDG started and began running in an unloaded condition. Because the Division 3 ECCS components and the associated Division 3 EDG were out-of-service for the current plant conditions, the Division 3 ECCS initiation signal did not result in the actual physical repositioning or actuation of any equipment.

Unit 2 control room personnel responded to the event by verifying that actual reactor vessel water level was normal for current plant conditions and then securing the ECCS equipment that had actuated. A follow-up investigation into the cause of the event by the licensee revealed that when the 2B RHR pump was started in SDC mode, its discharge pressure of approximately 385 psig in the solid reactor coolant system caused perturbations in the reactor vessel water level instrumentation.

<u>Analysis</u>

The inspectors determined that there was a performance deficiency associated with the failure of control room personnel to recognize that starting SDC in a solid condition could result in pressure perturbations in the reactor water level instrumentation. This error resulted in an inadvertent actuation of Division 2 ECCS components, and caused a transient on Unit 2 that presented an unnecessary challenge to shutdown reactor operations.

In accordance with NRC IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had an adverse impact on the objective for the Initiating Events Cornerstone for Reactor Safety, which is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." Specifically, the inspectors determined that: 1) the licensee had not provided the control room operating crew with adequate procedural guidance related to the initiation of SDC following the reactor system pressure test and; 2) control room operating personnel failed to adequately review and assess the potential adverse consequences associated with initiating SDC out of the sequence specified in LOS-NB-R2.

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 this finding occurred during the cold shutdown mode of operation, the inspectors utilized IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." Using Checklist 8, "Boiling Water Reactor Cold Shutdown or Refueling Operation with Time to Boil greater than 2 hours and RCS level less than 23 feet Above Top of Flange," of Appendix G, Attachment 1, "Phase 1 Operational Checklists for Both Pressurized Water Reactors and Boiling Water Reactors," the inspectors qualitatively determined that the finding involved adequate mitigation capability. In addition, using Table 1, "Losses of Control," of Appendix G, the inspectors qualitatively determined that the finding was not an event that could be characterized as a loss of control. As a result, the inspectors concluded that the finding was of very low safety significance (Green) and within the licensee's response band.

In addition, the inspectors determined that the finding was related primarily to the cross-cutting area of Human Performance as defined in NRC IMC 0305, "Operating

Reactor Assessment Program," since the control room personnel did not use conservative assumptions in decision-making and did not adopt a requirement that demonstrated that their actions were safe in order to proceed. Rather, they adopted a requirement to demonstrate that it is unsafe in order to disapprove an action, and in so doing did not identify the possible unintended consequences.

Enforcement

Criterion V of 10 CFR 50, Appendix B, "Instructions, Procedures, and Drawings," states 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 March 15, 2007, the licensee failed to provide the control room operating crew with adequate procedural guidance related to the initiation of SDC following the reactor system pressure test , such that an unnecessary and unintentional actuation of safety-related Division 2 ECCS equipment occurred and a transient on Unit 2 resulted that presented an unwarranted challenge to shutdown reactor operations.

The licensee entered this issue into their CAP as IR 604177. Corrective actions planned and completed by the licensee included performing an initial investigation into the event, performing an engineering analysis of system impact, and conducting a full root cause analysis for the event. Because the licensee has entered the issue into their CAP and the finding is of very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion V, is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000374/2007002-06)

4OA5 Other

.1 (Closed) URI 05000373/2005005-01; 05000374/2005005-01, Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency

Closure of this URI was documented in Section 4OA5.3 of NRC Initial License Examination Report No. 05000373/2006301; 05000374/2006301, dated January 29, 2007 (ADAMS Accession Number ML070320596). It is noted herein for record keeping purposes only.

.2 Compliance with NRC Confirmatory Order Dated November 22, 2005 (EA-04-170)

a. Inspection Scope

The inspectors completed the review of compliance with NRC Confirmatory Order EA-04-170, dated November 22, 2005, and amended by letter dated December 29, 2006. The Order has requirements for specific activities by the licensee during the two refueling outages following November 22, 2005, the initial date of the Order. The inspectors reviewed the licensee actions to determine compliance.

The inspectors reviewed LaSalle County Station procedures and training material and attended a pre-outage dynamic learning activity training session to assure that the licensee:

- Revised initial radiation worker training material to highlight HRA entry requirements and consequences for the radiation worker if requirements are 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 second outage since the initial date of the Confirmatory Order to assure that:

- During the first 10 days, or longer as necessary, of the L2R11 refueling outage, LaSalle had greeters at primary access points to the radiologically controlled area to enhance awareness of radiological controls; and
- For the L2R11 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.

The inspectors reviewed the corrective actions outlined in Exelon's letter dated December 17, 2004, to assure that [the licensee's contractor] Shaw revised its operating procedures, which were applicable fleet-wide, 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 Shaw daily shift brief during L2R11;
- Shaw employees who work in radiation areas read, understand, and sign a pledge to attest to his/her commitment to follow all radiological requirements and that each pledge was co-signed by the Shaw site manager, project superintendent, or site ALARA coordinator and were retained for future audit during a period of at least 1 year;
- Shaw superintendents were present at select pre-job briefs involving HRA entries; and
- Shaw participated in Exelon RP Manager peer group meetings at least once prior to the L2R11 outage and had plans for semiannual evaluation with the resultant commitment to take necessary action on RP 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 Shaw'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 L2R11 outage the plant management clearly established personnel expectations in following radiological work requirements.

These reviews did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors baseline refuel outage and RP inspection activities.

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 10, 2007. 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 exit meetings were conducted for the following inspections:

- A periodic public radiation safety inspection in the area of radioactive material processing and transportation with the Site Vice President, Ms. S. Landahl, and other members of licensee management on January 26, 2007;
- A biennial licensed operator requalification inspection with the Operations Training Manager, Mr. L. Blunk, on February 27, 2007, via telephone;
- A refueling outage ISI program engineering inspection with the Site Vice President, Ms. S. Landahl, and other members of licensee management on March 6, 2007;
- A periodic occupational radiation safety inspection in the areas of refuel outage ALARA and access control with the Site Vice President, Ms. S. Landahl, and other members of licensee management on March 7, 2007.

4OA7 Licensee-Identified Violations

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

- Technical Specification 5.4.1(c) requires that written procedures for the station's fire protection program be established, implemented, and maintained. Contrary to this requirement, on January 17, 2007, the licensee failed to implement an hourly fire watch patrol in accordance with Section 2.6 of OP-MW-201-007, "Fire

Protection System Impairment Control." Specifically, due to a logkeeping error, the hourly fire watch patrol went approximately 82 minutes between rounds.

The objective of the Initiating Events Cornerstone of Reactor Safety is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." In accordance with NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective, one of the key attributes of which is protection against fires. The violation was further determined to be of very low safety significance because an active automatic fire detection system was available in the areas subject to the fire watch patrol. The licensee had entered this issue into their CAP as IR 580108. Corrective actions by the licensee included enhancements to the fire watch patrol logkeeping scheme, a prompt investigation into the issue, and a rapid "same shift" communication on the issue to all personnel.

- Technical Specification 5.4.1(c) requires that written procedures for the station's fire protection program be established, implemented, and maintained. Contrary to this requirement, on March 8, 2007, licensee contractor personnel failed to properly implement a continuous fire watch required for a 30 minute period following hot work in a safety-related service water pump room. Specifically, contrary to the specified responsibilities outlined in Sections 3.4 and 4.3 of OP-MW-201-004, "Fire Prevention For Hot Work," the individual assigned to the fire watch became inattentive to his duties. A licensee contractor supervisor touring the job site noticed the inattentive individual, and directed that he be relieved of his duties.

  The objective of the Initiating Events Cornerstone of Reactor Safety is "to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations." In accordance with NRC Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance in that it had a direct impact on this cornerstone objective, one of the key attributes of which is protection against fires. The violation was further determined to be of very low safety significance because in addition to the inattentive fire watch, there was a second qualified fire watch in the room assigned to another post hot work related job. Upon review of the issue, the inspectors determined that due to the small size of the pump room, the second fire watch had been in a position to detect and take corrective actions for any post hot work ignitions that may have occurred within the room. The licensee had entered this issue into their CAP as IR 600925. Corrective actions by the licensee included disciplinary action against the inattentive fire watch in accordance with company policies, a prompt investigation into the issue, and a rollout of the issue to personnel at the next shift's briefings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee</u>

S. Landahl, Site Vice President
D. Enright, Plant Manager
J. Bashor, Site Engineering Director
R. Bassett, Emergency Preparedness Manager
L. Blunk, Operations Training Manager
R. Chrzanowski, Chemistry Manager
T. Connor, Maintenance Director
H. Do, Exelon Corporate (Cantera) Engineering - ISI
R. Ebright, Site Training Director
B. Ginter, Engineering Programs Manager
F. Gogliotti, System Engineering Manager
B. Kapellas, Radiation Protection Manager
S. Marik, Work Management Director
J. Rappeport, Nuclear Oversight Manager
D. Rhodes, Operations Director
J. Rommel, Design Engineering Manager
T. Simpkin, Regulatory Assurance Manager
H. Vinyard, Shift Operations Superintendent
C. Wilson, Station Security Manager

<u>Nuclear Regulatory Commission</u>

B. Burgess, Chief, Reactor Projects Branch 2

<u>Illinois Department of Emergency Management</u>

B. Metro, ASME Code Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000374/2007002-01	NCV	Failure to Use Valve Alignment Checklist When Clearing Tag Out Results in Mispositioned Valve and Low Instrument Nitrogen System Header Pressure. (Section 1R04.2)
05000373/2007002-02; 05000374/2007002-02	NCV	Incomplete Residual Heat Removal Heat Exchanger Vessel Weld Examinations. (Section 1R08)
05000374/2007002-03	URI	A 10 CFR 50.59 Screening Performed to Support Reactor Vessel Disassembly Procedure Changes Does Not Apply to De-Tensioning the Drywell Head in Mode 3. (Section 1R20)
05000374/2007002-04	FIN	De-Tensioning Drywell Head in Mode 3 Results Has Unanticipated Impact on Technical Specifications. (Section 1R20)
05000374/2007002-05	NCV	Failure to Adequately Plan and Proceduralize Reactor Vessel Nozzle Flushing Activities Results in Inadvertent ECCS Injection into the Reactor Vessel. (Section 4OA3.1)
05000374/2007002-06	NCV	Inadequate Procedural Instructions to Place Shutdown Cooling in Service Results in Inadvertent ECCS Injection into the Reactor Vessel. (Section 4OA3.2)

Closed

05000374/2007002-01	NCV	Failure to Use Valve Alignment Checklist When Clearing Tag Out Results in Mispositioned Valve and Low Instrument Nitrogen System Header Pressure. (Section 1R04.2)
05000373/2007002-02; 05000374/2007002-02	NCV	Incomplete Residual Heat Removal Heat Exchanger Vessel Weld Examinations. (Section 1R08)
05000374/2007002-04	FIN	De-Tensioning Drywell Head in Mode 3 Results Has Unanticipated Impact on Technical Specifications. (Section 1R20)
05000374/2007002-05	NCV	Failure to Adequately Plan and Proceduralize Reactor Vessel Nozzle Flushing Activities Results in Inadvertent ECCS Injection into the Reactor Vessel. (Section 4OA3.1)

05000374/2007002-06	NCV	Inadequate Procedural Instructions to Place Shutdown Cooling in Service Results in Inadvertent ECCS Injection into the Reactor Vessel. (Section 4OA3.2)
05000373/2005005-01; 05000374/2005005-01	URI	Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency. (Section 4OA5.1)

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 the NRC inspector 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.

1R04 Equipment Alignment

Procedures:
- LOP-HP-03; Preparation for Standby Operation of High Pressure Core Spray; Revision 19
- LOP-RH-11; Preparation for Standby Operation of the Low Pressure Coolant Injection System; Revision 24
- LOA-IN-201; Los of Drywell Pneumatic Air Supply; Revision 6
- OP-AA-109-101; Clearance and Tagging; Revision 0
- LOP-IN-02M; Unit 2 Drywell Pneumatic system Mechanical Checklist; Revision 16

Drawings and Prints:
- M-95; High Pressure Core Spray; Revision AM
- M-96; Residual Heat Removal System, Sheet 1; Revision AX
- M-96; Residual Heat Removal System, Sheet 4; Revision AD

Issue Reports:
- 581287; MCR Unexpected Alarms and LOA-IN-201 Entry; 1/20/2007
- 581543; Operations Crew 6 Clock Reset; 1/20/2007

1R05 Fire Protection

LaSalle County Station - Fire Protection Report
LaSalle County Station - Technical Requirements Manual; Plant Systems, Section 3.7.m; Revision 3

Procedures:
- CC-AA-201; Plant Barrier Control Program; Revision 6
- LES-DC-106; Safe Shutdown (Appendix R) DC Emergency Light Inspection Sheets; Revision 37
- OP-MW-201-004; Fire Prevention For Hot Work; Revision 1
- OP-MW-201-007; Fire Watch Inspection Log; Revision 4

Plant Barrier Impairment (PBI) Permits:
- W-Wall-11' 11" S-18-N-744.00r6

Issue Reports:
- 580108; Fire Watch for DEHC PBI Performed 22 Minutes Late; 1/17/2007
- 591765; Fire Watch Tour; 2/15/2007
- 591766; Fire Watch Documentation; 2/15/2007

1R07 Heat Sink Performance

LaSalle Station Generic Letter 89-13 Program Basis Document; Revision 4

Procedures:
- ER-AA-340-1002; Service Water Heat Exchanger and Component Inspection Guide; Revision 3
- ER-AA-340-1002; Attachment 1: Heat Exchanger Inspection Data Sheet; Revision 0

Engineering Changes and Analyses:
- EC 364899; Evaluation of Unit 2A RHR Heat Exchanger Thermal Performance Data Using Alternate (EPRI) Methodology; Revision 0

1R08 Inservice Inspection Activities

Issue Reports:
- 472136; Vendor Conducted NDE Without Proper Certification; 3/29/2006
- 486114; ASME Section XI Relief Request Document Deficiency; 5/3/2006
- 307057; Temporary Attachment Welded to Core Support Leg; 3/1/2005
- 376305; Leak between RHRSW Strainer and 1E12-F336B; 9/21/2005
- 599201; RHR HX Welds Not UT per ASME Code; 3/4/2007
- 599178; NDE Report did not Document the use of an Alternate IQI; 3/4/2007
- 599869; Repair/Replacement of 2B21-F019; 3/6/2007
- 599892; Potential Deficiencies in GE Procedures; 3/5/2007
- 300094; L2R10 - NDE Exhibits Surface Indication; 2/10/2005

NDE Observation Related Documents:
- GE-PDI-UT-1; PDI Generic Procedure for the Ultrasonic Examination of Ferritic Piping Welds; Revision 5
- GE-PDI-UT-2; PDI Generic Procedure for the Ultrasonic Examination of Austenitic Piping Welds; Revision 4
- GE-PDI-UT-10; PDI Generic Procedure for the Ultrasonic Examination of Dissimilar Metal Welds; Revision 2
- GE-UT-705; Procedure for the Examination of Reactor Pressure Vessel Nozzle Inner Radius an Nozzle to Vessel Welds with the Geris 2000 OD in Accordance with Appendix VIII; Revision 5
- GE-PT-100; Procedure for Liquid Penetrant Examination Using Florescent and Visible Dye Penetrant Inspection Methods; Revision 6

Documents Associated with Disposition of Relevant Indications:
- 2R10-100B; Examination Summary Sheet, RI24-2854X; 2/24/2005
- 2R10-109; Examination Summary Sheet, RH40-2877X; 2/9/2005
- 2R8-008; Examination Summary Sheet, 1RH-2004-42A; 11/21/2000

Other Documents:
- Examination Summary Sheet, B-RHR HX; 2/19/1996
- Examination Summary Sheet, 1RH-HX1B-03; 1/18/2002
- Examination Summary Sheet, 1RH-HX1B-04; 1/19/2002
- Examination Summary Sheet, 1RH-HX1B-05; 1/24/2004

- Examination Summary Sheet, 1RH-HX1B-06; 1/25/2004
- Examination Summary Sheet, B-RHR HX; 9/17/1996
- Examination Summary Sheet, 1RH-HX2B-03; 1/23/2003
- Examination Summary Sheet, 1RH-HX2B-04; 1/22/2003
- Examination Summary Sheet, 1RH-HX2B-05; 2/13/2005
- Examination Summary Sheet, 1RH-HX2B-06; 2/13/2005

Pressure Boundary Welding Related Documents:
- Radiographic Film and Reader Sheets for Welds Nos.1 and 2 Fabricated During Replacement of Main Steam Drain Isolation Valve 2B21-F019; 2/21/2005
- Welding Services INC. Non Conformance Report 05-051; 2/22/2005
- Weld Data Sheet, WO 00554321-01, Weld 2R-1; 2/21/2005
- Weld Data Sheet, WO 00554321-01, Weld 1; 2/16/2005
- Weld Data Sheet, WO 00554321-01, Weld 2; 2/16/2005
- Liquid Penetrant Examination Data Sheet, 2/20/2005
- WPS 5B-5B-T-101; 1/27/2005
- PQR 5B-5B-T-102; 1/27/2005

1R11 Licensed Operator Requalification Program

- ESG 73; Licensed Operator Requalification Scenario Guide; Revision 0

1R12 Maintenance Effectiveness

Issue Reports:
- 578368; Circ Water has Met Its Maintenance Rule Reliability Criteria; 1/12/2007
- 466828; 1A CW Pump Tripped on Startup; 3/15/2006
- 465501; 1A CW Pump Tripped; 3/13/2006

1R13 Maintenance Risk Assessments and Emergent Work Control

Issue Reports:
- 566377; 2C41-C001B Discharge Flow Low; 12/7/2006
- 575484; Small Indication on No. 4 Suction Valve During Assembly; 1/4/2007
- 575424; 2B SBLC Low Flow; 1/4/2007

L2R11 Comprehensive Shutdown Safety Assessment Report

Engineering Analyses:
- EC 364168; Methodology to Determine Best Estimate Power Monitor; Revision 0
- EC 364169; Unit 1 – Implementation of Correction Factor for Feedwater Flow; Revision 0
- EC 364955; Generic Evaluation of Lost Parts During L2R11; Revision 0
- EC 364960; 26A Feedwater Heater Lost Carbon Steel Nuts and Lock Washers Evaluation; Revision 0
- EC 364993; Lost Parts Evaluation – Various Lost Parts During L2R11; Revision 0
- EC 365033; Lost Parts Evaluation – HPCS Sparger Nozzle Flow Diverter; Revision 0
- EC 365055; Generic Lost Parts Evaluation for Outage L2R11 for Main Condenser Closeout; Revision 0

Procedures:
- LLP-2007-001; Monitoring Feedwater Flow Correction Factor; Revision 0

1R15 Operability Evaluations

Operability Evaluations:
- OE 05-003; Unit 1 and Unit 2 RCIC Steam Line Tunnel Temperature/Leak Detection Instrumentation; Revision 1
- OE 07-001; Unit 1 Jet Pump No. 19; Revisions 0 and 1
- OE 06-002; Instrument Nitrogen and Safety Relief Valves 1(2)B21-F013C, D, E, S, and U; Revision 2

Issue Reports:
- 554981; Jet Pump No. 19 High D/P; 11/8/2006
- 580175; Jet Pump No. 19 D/P at Upper Limit; 1/18/2007
- 581231; Increased Indicated Flow Through Unit 1 Jet Pump No. 19; 1/19/2007
- 586546; 2FP085 Turbine Bearing Deluge Priming Stop Valve; 2/2/2007
- 591067; Best Estimate Power Monitor Core Thermal Power Alarm Received; 2/13/2007
- 591130; Status of Temperature Monitoring for Steam Tunnels, LOS-CS-Q1; 2/13/2007

Calculations:
- L-000187; Assessment of All 1E Equipment in Zone H5C for Ambient Service Temperature of 200ºF; 2/2/1996

Engineering Changes and Analyses:
- EC 363909; Install Temporary Ventilation Barrier on EHC Panel 2PA01J to Allow Increased AEER Breach Size in Support of DEHC Modification Cable Replacement; Revision 0

1R17 Permanent Plant Modifications

Engineering Changes and Analyses:
- EC 355023; Unit 2 Digital EHC Upgrade Project; Revision 1
- EC 354449; Installation of Valve 2DG117, Installation of Line Stop Fitting Assemblies and Hot Tapping on CSCS Line Nos. 2DG06A-4"-2DG04B-12"; Revision 3

1R19 Post-Maintenance Testing

Issue Reports:
- 566377; 2C41-C001B Discharge Flow Low; 12/7/2006
- 575484; Small Indication on No. 4 Suction Valve During Assembly; 1/4/2007
- 575424; 2B SBLC Low Flow; 1/4/2007

Work Orders:
- 922700-01; Replace 1E22-F363A, 1B EDG ‘A’ Starting Air Receiver Inlet Valve; 1/10/2007
- 922700-02; Replace 1E22-F364A, 1B EDG ‘C’ Starting Air Receiver Inlet Valve; 1/10/2007
- 922700-05; VT-2 Leak Examination for 1E22-F363A, 1B EDG ‘A’ Starting Air Receiver Inlet Valve; 1/10/2007

- 922700-06; VT-2 Leak Examination for 1E22-F364A, 1B EDG ‘C’ Starting Air Receiver Inlet Valve; 1/10/2007
- 982500-01; 2B SLC Pump Low Flow / Repair Pump; 1/4/2007

Procedures:
- LMP-SC-02; Standby Liquid Control Pump Maintenance; Revision 7
- LOS-DG-M3; 1B(2B) Diesel Generator Operability Test; Revision 65
- LOS-SC-Q1; SBLC Pump Operability/Inservice Test and Explosive Valve Continuity Check; Revision 24
- LOP-NB-01; Reactor Vessel Leakage Test; Revision 42

<u>1R20</u> <u>Outage Activities</u>

Issue Reports:
- 596847; Unscheduled LLRT Performed to Support Calculation L002666 Concern; 2/27/2007
- 597525; Drywell Head Bolt De-Tensioning; 2/28/2007
- 601913; Drywell Head De-Tensioning While in Mode 3; 3/10/2007
- 601924; The NRC Identified Potential Finding With 50.59 Evaluation; 3/10/2007
- 601925; The NRC Identified Potential Finding Associated With Drywell Evolution; 3/10/2007
- 601926; The NRC Identified Potential Finding With LTS-100-15; 3/10/2007

Procedures:
- LGP-1-1; Normal Unit Startup; Revision 79
- LGP-1-S1; Master Startup Checklist; Revision 60
- LGP-2-1; Normal Unit Shutdown; Revision 71
- LOP-AA-03; Reactor Mode Changes; Revision 22
- LOP-DW-01; Drywell Close Out (After Outage); Revision 41
- LOP-DW-02; Drywell Entry and Inspection (Shutdown, Startup, or Operation); Revision 14
- LOP-FC-16; Reactor Vessel/Cavity Draindown via RHR Shutdown Cooling; Revision 14
- LOP-RH-07; Shutdown Cooling System Startup, Operation and Transfer; Revision 52
- LOP-RM-01; Reactor Manual Control Operation; Revision 28A
- OP-AB-300-1001; BWR Control Rod Movement Requirements; Revision 3
- OP-AB-300-1003; BWR Reactivity Maneuver Guidance; Revision 3
- MA-AB-756-600; Reactor Disassembly; Revision 7
- LTS-100-15; Type ‘B’ Local Leak Rate Test; Revision 24

Calculation L-002666; Evaluation of the LaSalle Drywell Bolts to Justify Permanent Removal of the Bolts; Revision 0

<u>1R22</u> <u>Surveillance Testing</u>

Procedures:
- LOA-MS-201; Unit 2 Main Steam System Abnormal; Revision 5
- LOP-NR-06; Traversing Incore Probe (TIP) Operation; Revision 24
- LOP-VR-02; Reactor Building Ventilation System Shutdown; Revision 26
- LOS-CS-Q1; Secondary Containment Damper Operability Test; Revision 29
- LOS-DG-M2; 1A(2A) Diesel Generator Operability Test; Revision 67
- LOS-DG-102; 1A Diesel Generator, 1DG01K, Start and Load Acceptance Surveillance; Revision 2

- LOS-DG-209; Unit 2 Integrated Division I Response Time Surveillance; Revision 4
- LOS-RH-Q1; RHR (LPCI) and RHR Service Water Pump and Valve Inservice Test for Modes 1, 2, 3, 4 and 5; Revision 64
- LTS-100-3; Main Steam Isolation Valve Local Leak Rate Test for 1(2)B21-F022A/B/C/D, 1(2)B21-F028A/B/C/D, and 1(2)B21-F067A/B/C/D; Revision 18
- LTS-100-23; Combustible Gas Control Isolation Valves Local Leak Rate Test for 1(2)HG001A/B, 1(2)HG002A/B, 1(2)HG005A/B , and 1(2)HG006A/B; Revision 22
- LTS-300-5; Primary Containment Leak Rate Testing Program; Revision 36
- LTP-1600-7; Data Collection and Evaluation for LPRM Calibration; Revision 19

Work Orders:
- 00834873; LLRT For 2B21-F022A, 2B21-F028A, and 2B21-F067A; 3/1/2007
- 00835751; LLRT For 2B21-F022B, 2B21-F028B, and 2B21-F067B; 3/1/2007
- 00835752; LLRT For 2B21-F022C, 2B21-F028C, and 2B21-F067C; 3/1/2007
- 00835753; LLRT For 2B21-F022D, 2B21-F028D, and 2B21-F067D; 3/1/2007

High Level Activity for LOS-CS-Q1; Secondary Containment Damper Operability Test; 2/5/2007

<u>1R23</u> <u>Temporary Plant Modifications</u>

Temporary Configuration Changes:
- TCCP 364169; Unit 1 – Implementation of correction Factor for Feedwater Flow; Revision 0

Engineering Changes and Analyses:
- EC 364168; Methodology to Determine Best Estimate Power Monitor; Revision 0
- EC 363127; Temporary Covers on EDG Coolers 2DG01A and 2E22-S001 to Facilitate CSCS Valve Replacement; Revision 0
- EC 363303; CSCS Temporary Piping and Supporta Connecting U2 Div 2 to U2 Div 3 During L2R11; Revision 0

Procedures:
- LOP-RP-05; Defeating All Scram Signals Except Manual; Revision 15
- LLP-2007-001; Monitoring Feedwater Flow Correction Factor; Revision 0
- LLP-2006-002; Unit 2 - Division 2 CSCS Draining; Revision 0

<u>2PS2</u> <u>Radioactive Material Processing and Transportation</u>

Issue Reports:
- 303385; General Electric Engineering Documents Have Inadequate Cross References; 2/19/2005
- 306719; Numerous 0DA01 Sump Floor Drains Mislabeled; 3/1/2005
- 309852; Errors and Omissions in Auxiliary Building Floor Drain Drawing; 3/8/2005
- 321356; Former Waste Stream Chosen for Characterization; 4/5/2005
- 322766; Waste Floor Drain Transfer to Waste Equipment Drains Not Per Original Design; 4/8/2005
- 478488; Nuclear Oversight Identified a Perceived Latitude in Scaling Factor Determination; 4/13/2006
- 489291; Radwaste Container Contacted Shield Wall in Interim Radwaste Storage Facility; 5/11/2006

- 576041; 10 CFR Part 61 Data Has Expired; 1/5/2007
- 578084; High Radiation Area Identified in U0 Turbine Building Waste Precoat Aisle; 1/11/2007
- 547812; Self-Assessment – Radioactive Material Shipping; 12/21/2006

Procedures:
- FO-AD-002-46978; Operating Guidelines for Use of Polyethylene High Integrity Containers for Exelon West Stations; Revision 2
- LOP-WX-05; Waste Sludge Tank Decant; Revision 6
- RP-AA-602; Packaging of Radioactive Material Shipments; Revision 11
- RW-AA-100; Process Control Program for Radioactive Wastes; Revision 4

Miscellaneous Reports:
- LM06-038; Radioactive Shipment Package Control Rod Drives; 3/7/2006
- LW06-011; Radioactive Shipment Package Bead Resin; 4/7/2006
- LW06-014; Radioactive Shipment Package Sludge Resin; 5/4/2006
- LW06-015; Radioactive Shipment Package Septa; 6/15/2006
- LW06-023; Radioactive Shipment Package Filters; dated 8/25/2006
- NOSA-COMP-06-04; Chemistry Radwaste, Effluent, and Environmental Monitoring Program Audit Report; 5/31/2006
- NOSA-LAS-06-04; Chemistry Radwaste, Effluent, and Environmental Monitoring Program Audit Report; 4/19/2006
- 2006 Process Waste Scaling Factors; 1/17/2007

<u>2OS1</u> <u>Access Control to Radiologically Significant Areas</u>

Issue Reports:
- 596849; Dose Rates on 2RT Inlet Water Sample From 40 to 550 mR/hour; 2/27/2007
- 596896; 694 Reactor Building 2A RHR is Locked High Radiation Area; 2/27/2007
- 597024; 2A RHR 710, 694, 673 Danger Locked High Radiation; 2/27/2007
- 598713; Issues identified By Radiation Protection Behavior Correction Specialist; 3/2/2007
- 598755; Incorrect Radiation Work Permits Listed for Work Orders In Passport; 3/2/2007
- 599001; Nuclear Oversight Identifies Radiation Worker Practices in Heater Bay; 3/3/2007
- 599088; Nuclear Oversight Identifies Radiation Worker Practices Issues; 3/4/2007
- 599089; Nuclear Oversight Identifies Radiation Protection Procedural Compliance; 3/4/2007
- 460143; Worker Received Electronic Dosimetry Alarm; 2/28/2006
- 528789; Nuclear Oversight Identified: Pre-Job Briefing Checklist Not Used; 9/8/2006
- 522944; Individual Received a Dose Rate Alarm; 8/23/2006
- 524968; Radiation Protection Technician Electronic Dosimeter Rate Alarm While Surveying; 8/28/2006
- 558607; Potential Safety Concern on Radiation Portal Monitors; 11/16/2006

RWP 10006888; L2R11 Noble Metals Application; Revision 0

<u>2OS2</u> <u>As-Low-As-Is-Reasonably-Achievable Planning And Controls (ALARA)</u>

Issue Reports:
- 453272; FASA: La Salle Station Exposure Reduction; 9/8/2006
- 459620; Dose Rate Increase on Scorpion Platform; 2/27/2006
- 464328; 2A Residual Heat Removal Steam Condensing Return Needs Flush; 3/9/2006

- 470925; Scorpion Platform Enhancements to Reduce CRUD Traps; 3/26/2006
- 483032; Cask Well Drain Flush Lessons Learned; 4/25/2006
- 496626; Deficiencies Discovered During Extent of Condition Review; 6/5/2006
- 511388; Scorpion Platform Decontamination dose Over-estimated; 7/20/2006
- 568469; Self-Assessment: Access Control to Radiologically Significant Areas and ALARA Planning and Controls; 2/22/2007

Radiation Work Permits:
- 10006820; L2R11 Scaffold Activities In Drywell; Revision 0
- 10006829; L2R11 Control Rod Drive Pull/Put; Revision 0
- 10006837; L2R11 General Electric Drywell In-Service Inspection; Revision 0
- 10006838; L2R11 Drywell Nozzle In-Service Inspection Support Activities; Revision 0
- 10006863; Reactor Vessel Disassembly/Reassembly for L2R11; Revision 0

Procedures:
- RP-AA-270; Prenatal Radiation Exposure; Revision 3
- RP-AA-401; Operational ALARA Planning and Controls; Revision 7

L1R11 Refueling Outage Report; February 2006

<u>4OA3</u> <u>Event Follow-up</u>

Issue Reports:
- 598883; Unit 2 Division 1 ECCS Unplanned Initiation; 3/3/2007
- 604727; RWP No. 10006863 Reactor Disassembly/Reassembly Post Job Review; 3/16/2007

<u>4OA5</u> <u>Other</u>

EA-04-170; NRC Confirmatory Order; 11/22/2005
Change of Maintenance-Modification Contractor Referenced in NRC Confirmatory Order Dated November 22, 2005; 12/29/2006
Memorandum From John L. Schrage: Second Independent Review of Commitments from Confirmatory Order Concerning High Radiation Area Access Controls; 3/5/2007

Procedures and Instructions:
- Shaw Shift Brief Radiological and High Radiation Discussion Material
- EXN-NMP 101.01; Site Manager Expectations; Revision 0
- EXN-NMP 102.01; Roles and Responsibilities; Revision 0
- EXN-NMP 108.03; Supervision and Non-Manual Staff Expectations and Performance Evaluation Process; Revision 0
- EXN-NMP 108.04; Craft Expectations and Performance Evaluation Process; Revision 0
- EXN-NMP 110.02; Shaw/Stone and Webster Specific Self-Assessment Guidelines; Revision 0
- EXN-NMP 302.01; Outage Meetings and Reports; Revision 0
- RP-LA-210; Issuance of TLD in the Total Exposure Computer System; Revision 0

4OA7 Licensee-Identified Violations

Issue Reports:
- 600925; Fire Watch Discovered Inattentive While on Duty; 3/8/2007
- 580108; Hourly Fire Watch Tour Was Completed 22 Minutes Late; 1/17/2007

Procedures:
- OP-MW-201-007; Fire Protection System Impairment Control; Revision 5
- OP-MW-201-004; Fire Prevention for Hot Work; Revision 1

LIST OF ACRONYMS USED

ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CSCS	Core Standby Cooling System
CW	Circulating Water
DAW	Dry-Active Waste
DEHC	Digital Electro-Hydraulic Control
DC	Direct Current
DG	Diesel Generator
DOT	Department of Transportation
d/p	Differential Pressure
DRP	Division of Reactor Projects
DW	Drywell
ECCS	Emergency Core Cooling System
ED	Electronic Dosimeter
EDG	Emergency Diesel Generator
EHC	Electro-Hydraulic Control
EPRI	Electric Power Research Institute
FASA	Focused Area Self Assessment
FPR	Fire Protection Report
FW	Feedwater
GE	General Electric
HPCS	High Pressure Core Spray
HRA	High Radiation Area
I&C	Instrumentation and Controls
IEEE	Institute of Electrical & Electronic Engineers
IMC	Inspection Manual Chapter
IN	Instrument Nitrogen
IP	Inspection Procedure
IQI	Image Quality Indicator
IR	Inspection Report or Issue Report
ISI	Inservice Inspection
IV	Independent Verification
kV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLRT	Local Leak Rate Testing
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LSA	Low Specific Activity
MG	Motor-Generator
MOV	Motor-Operated Valve

mrem	Millirem
msec	Millesecond
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
OD	Outer Diameter
PBI	Plant Barrier Impairment
PCIS	Primary Containment Isolation System
PDI	Performance Demonstration Initiative
PI	Performance Indicator
PI&R	Problem Identification and Resolution
PM	Planned or Preventative Maintenance
PMT	Post-Maintenance Testing
psid	Pounds Per Square Inch Differential
psig	Pounds Per Square Inch Gauge
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RFO	Refueling Outage
RFP	Reactor Feed Pump
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
SBLC	Standby Liquid Control
scf	Standard Cubic Feet
SDC	Shutdown Cooling
SDP	Significance Determination Process
SIL	Service Information Letter
SLC	Standby Liquid Control
SRA	Senior Reactor Analyst
SSC	Systems, Structures, and Components
SW	Service Water
TIP	Traversing Incore Probe
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Testing or Examination
Vac	Volts Alternating Current
Vdc	Volts Direct Current
VHRA	Very High Radiation Area
WO	Work Order