June 4, 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 EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS, AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000373/2007007(DRS); 05000374/2007007(DRS)

Dear Mr. Crane:

On May 18, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed baseline inspections of Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the LaSalle County Station. The enclosed report documents the results of the inspection which was discussed with Ms. Susan Landahl and other members of your staff at the completion of the inspection on May 18, 2007.

The inspectors 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 the inspection, two NRC-identified findings of very low safety significance were identified. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

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

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

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

Enclosure: Inspection Report 05000373/2007007(DRS); 05000374/2007007(DRS)
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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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
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Illinois Emergency Management Agency
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Letter to Mr. Christopher Crane from Mr. Dave E. Hills dated June 4, 2007

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS, AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000373/2007007(DRS); 05000374/2007007(DRS)

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2007007(DRS); 05000374/2007007(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL 61341
Dates:	April 23, 2007 through May 4, 2007, and May 14 through 18, 2007
Inspectors:	M. Holmberg, Reactor Inspector, Lead R. Langstaff, Reactor Inspector S. Sheldon, Reactor Engineer
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000373/2007002(DRS); 05000374/2007002(DRS); 4/23/2007 - 5/18/2007; LaSalle County Station, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59), and Permanent Plant Modifications.

The inspection covered a 3-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three Region based inspectors. Two Green Non-Cited Violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 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-Revealed Findings

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” involving an inadequate maintenance procedure used to remove drywell head bolts. Specifically, in maintenance procedure MA-AB-756-600 “Reactor Disassembly,” the licensee failed to provide instructions to remove only “every other bolt” to ensure that the drywell head assembly configuration remained within the analyzed configuration for operating Modes 1 through 3. As a corrective action, the licensee intended to provide additional procedure instructions to restrict bolt removal to every other bolt, or delete the procedure option for early bolt removal with the plant in Modes 1 through 3.

  The finding was determined to be greater than minor because absent NRC intervention the inadequate procedure could lead to a more significant problem. Specifically, procedure MA-AB-756-600 would have allowed removal of bolts from adjacent locations on the drywell head assembly which could affect the structural and/or leakage integrity of the containment. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” because it did not represent an actual open pathway for containment, and did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. The primary cause of this finding was related to the cross-cutting area of human performance because the licensee did not provide complete, accurate, and up to date design documentation to plant personnel. (Section 1R17)

- Green. The inspectors identified a finding having very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion XII, “Control of Measuring and Test Equipment,” involving lack of calibrated tools used to establish torque for the drywell head assembly bolts. Specifically, for five air hammer wrenches used to install drywell head assembly bolts on Unit 1 and Unit 2, the licensee failed to ensure these tools were properly calibrated to confirm the accuracy of the torque applied. The licensee entered this issue into the corrective action program, performed an operability evaluation, and concluded that sufficient torque had been applied to the drywell head bolts. The licensee operability conclusion was based upon the vendor advertised torque wrench specifications, torque margins available in the design analysis, and periodic air hammer wrench maintenance.

The finding was determined to be greater than minor because absent NRC intervention the lack of calibration testing for these wrenches could lead to a more significant problem. Specifically, the drywell head assembly bolts may not receive sufficient torque to establish a preload which assures containment leakage and structural integrity. The finding was of very low safety significance based on a Phase 1 screening in accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," because it did not represent an actual open pathway for containment, and did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. The primary cause of this finding was related to the cross-cutting area of human performance because the licensee did not provide adequate and available facilities and equipment (e.g. calibrated equipment) for personnel reassembling the drywell head. (Section 1R17)

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From April 23, 2007 through May 4, 2007, and May 14 through 18, 2007, the inspectors reviewed five safety evaluations performed pursuant to 10 CFR 50.59 to determine if the evaluations were adequate and to determine if prior Nuclear Regulatory Commission (NRC) approval was obtained if applicable. These five safety evaluations were all that the licensee had approved for changes to the plant since completion of the last NRC review of 10 CFR 50.59 safety evaluations. Therefore, the inspectors considered this portion of the procedure completed with less than minimum expected number of safety evaluations (six) as identified in NRC inspection procedure 71111.02. The inspectors also reviewed 18 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation.

The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. To assess the adequacy of these evaluations and screenings, the inspectors used in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments." The list of documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From April 23, 2007 through May 4, 2007, and May 14 through 18, 2007, the inspectors reviewed ten permanent plant modifications that had been installed in the plant during the last two years. The modifications were selected based upon risk significance, safety significance, complexity, and several modifications were chosen that affected the Barrier Integrity Cornerstone.

For the modification and design reviews completed, the inspectors performed physical inspections of the following plant components:

- Drywell head assembly (DHA) tools and equipment laydown areas on the Unit 2 Refueling Floor;
- Unit 2 digital electro-hydraulic control equipment installed in the Auxiliary Electric Equipment Room and Control Room;
- Replacement scram discharge volume vent and drain pilot valves in the Unit 1 Reactor Building;
- Fuel priming pumps installed on the 0 and 1A emergency diesel generators; and
- Unit 1 hydraulic control unit rod control management system transponder cards and branch amplifier card configuration as displayed on a licensee mockup.

The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, the licensing bases, and to confirm that the changes did not adversely affect any systems' safety function. In particular, the inspectors reviewed the post-modification testing to ensure the functionality of associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration. The inspectors applied NRC regulations and applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors are included as an attachment to this report.

b. Findings

b.1 Inadequate Procedure for Removal of Drywell Head Bolts

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” having very low safety significance (Green) involving inadequate maintenance procedure used to remove drywell head bolts.

Description: On February 26, 2007, the licensee removed 30 of 60 DHA bolts with Unit 2 in Mode 3 in accordance with procedure MA-AB-756-600 “Reactor Disassembly.” Specifically, Attachment 2 step 9a of this procedure authorized maintenance staff to remove 50 percent of the drywell head bolts with the Unit in any Mode (e.g., included Modes 1 through 3). However, this procedure step did not provide maintenance staff with instructions to remove only “every other bolt” to ensure the DHA bolted configuration remained within that analyzed in calculation L-002666 “Evaluation of LaSalle Drywell Bolts to Justify Permanent Removal of Bolts.” Fortunately, the licensee maintenance supervisor verbally instructed workers to remove every other bolt to ensure the DHA remained within the analyzed configuration. However, the inspectors identified that the procedure was inadequate because it allowed removal of 50 percent of the DHA bolts without regard to bolt locations. Specifically, if adjacent bolts were removed, it would have invalidated the DHA analysis and potentially affected containment leakage integrity under a design basis loss-of-coolant accident condition.

This inadequate procedure guidance appeared to stem from a lack of on-site engineering reviews when the original procedure was approved by the site staff in September of 2000, based upon a corporate generated maintenance procedure. The licensee’s procedure approval process had not changed since 2000, MA-AB-756-600 had undergone 7 revisions, and this deficient procedure was recently used on the Unit 2 DHA, without licensee staff identifying the deficiency. Therefore, the inspectors concluded that the licensee had opportunities to correct this performance deficiency, and that it represented current plant performance.

Analysis: The inspectors determined that, failure to establish an adequate procedure to ensure that the DHA would remain within analyzed configurations during reactor maintenance activities, was a performance deficiency warranting a significance evaluation. The finding was determined to be greater than minor, because of the absence of NRC intervention, the inadequate procedure could lead to a more significant problem. Specifically, procedure MA-AB-756-600 would allow removal of bolts from adjacent locations on the DHA which could affect the structural and/or leakage integrity of the Unit 1 and Unit 2 containments.

This finding affected the Barrier Integrity Cornerstone because if left uncorrected, it could affected the integrity of the Unit 1 and 2 containment barrier. The inspectors evaluated the finding using inspection manual chapter (IMC) 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations,” Phase 1 screening, and determined that the finding was of Green risk because it did not represent an actual open pathway for containment, and did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. This finding had a cross-cutting aspect in the area of human performance because the licensee did not provide complete, accurate and up to date design documentation to plant personnel.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of May 16, 2007, the licensee had not adequately translated design basis information for the DHA bolted configuration into maintenance procedure MA-AB-756-600 "Reactor Disassembly" Revision 7. Specifically, in Step 9a of Attachment 2 of procedure MA-AB-756-600, the licensee did not limit removal of DHA bolts in Modes 1 through 3 to "every other bolt" as analyzed in Calculation L-002666. This procedural deficiency had existed since September 19, 2000, when Revision 0 of corporate procedure MA-AB-RS-6-00600 "Reactor Disassembly" was approved for site use. The licensee documented this issue in AR 00630337 and intend to provide additional procedure instructions to either restrict bolt removal to every other bolt, or delete the procedure option for early DHA bolt removal. Because this violation was of very low safety significance and was entered into the licensee's corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000373/2007007-01; NCV 05000374/2007007-01).

b.2 Lack of Calibration or Testing for Air Hammer Tools Used for DHA Bolt Installation

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix B, Criterion XII, "Control of Measuring and Test Equipment," having very low safety significance (Green) involving lack of calibrated tools used to establish torque for the DHA bolts.

Description: For the LaSalle Unit 1 and 2 refueling outages the DHA bolts were originally tensioned with hydraulic bolt tensioners. After September of 2000, the licensee changed this process to use air driven impact wrenches for reassembly of the DHA in accordance with procedure MA-AB-756-601"Reactor Reassembly." These air hammer wrenches were not included in the licensee's Maintenance and Test Equipment Control Program and no other calibrated device was used to confirm that adequate torque was applied to the DHA bolts. During reassembly, the 60 DHA bolts are considered tight when the air wrench impacting continues for 10 seconds after rotation of the bolt stops. However, the licensee did not conduct impact wrench torque tests to demonstrate that sufficient torque was produced to establish the bolt preload assumed in the DHA analysis. Specifically, the torque assumed by the licensee in the current DHA analysis (calculation L-002666) was 5000 ft-lbs to ensure 74 kips of bolt preload.

Without calibration tests for these air hammer wrenches, the inspectors were concerned that the DHA bolts may not have been sufficiently torqued to meet the bolt preload assumed in the accident analysis. The inspectors identified the following factors which could result in achieving less than design torque for the DHA bolts: 1) no distinguishing marks existed on air hammer wrench socket, which would challenge maintenance workers' ability to determine when the bolt head enclosed by the socket had stopped rotating; 2) after the end of bolt rotation, the 10-second interval to continue tightening was not measured so actual times could be less; 3) the inlet air supply to the hammer could fall below 90 psig required by the wrench vendor to provide for maximum torque output; and 4) the air hammers could have a factory or service induced defect on internal components adversely affecting torque output. To minimize the effect of service induced defects on torque output, the licensee had rebuilt one of the five air hammer wrenches approximately every two years (eight to ten years between rebuilds for a given air wrench). The licensee staff concluded that sufficient torque had been applied to the drywell head bolts to justify containment operability based upon; the vendor advertized specifications that these wrenches would produce 10,000 ft-lbs within 6 seconds after bolt rotation stops, torque margins available in the design analysis, and periodic air hammer maintenance.

The use of uncalibrated torque wrenches on DHA bolts appeared to stem from a lack of on-site engineering reviews when the original procedure was approved by the site staff in September 2000, based upon a corporate generated maintenance procedure. The licensee's procedure approval process had not changed since 2000, MA-AB-756-601 had undergone 6 revisions, and the licensee continued to use uncalibrated torque wrenches on the DHA for Unit 1 and 2 outages completed since September 2000. Therefore, the inspectors concluded that the licensee had opportunities to correct this performance deficiency, and that it represented current plant performance.

<u>Analysis</u>: The inspectors determined that, failure to establish calibration testing for air hammer wrenches used on the DHA bolting to ensure adequate torque was applied, was a performance deficiency warranting a significance evaluation. The finding was determined to be greater than minor, because of the absence of NRC intervention, the lack of calibration testing could lead to a more significant problem. Specifically, the DHA bolts may not receive sufficient torque to establish a preload which assured containment leakage and structural integrity.

This finding affected the Barrier Integrity Cornerstone because if left uncorrected, it could adversely affect the integrity of the Unit 1 and 2 containment barrier. The inspectors evaluated the finding using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as Green because it did not represent an actual open pathway for containment, did not involve a reduction in defense in depth for the atmospheric control or hydrogen control function of containment. This finding had a cross-cutting aspect in the area of human performance because the licensee did not provide adequate and available facilities and equipment (e.g. calibrated equipment) for personnel reassembling the DHA.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion XII "Control of Measuring and Test Equipment," requires in part that "Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities

affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits."

Contrary to the above, as of May 16, 2007, the licensee had not adequately established measures for five air hammer wrenches used to install DHA bolts on Unit 1 and Unit 2, to ensure these tools were properly calibrated and adjusted at specified periods to maintain the accuracy of the torque applied. This deficiency had existed since September 19, 2000, when the licensee authorized use of air hammer wrenches for bolt installation in accordance with Revision 0 of corporate procedure MA-AB-RS-6-601 "Reactor Reassembly." The licensee documented this issue in AR 00630431 and was evaluating options for use of calibrated equipment for establishing DHA bolt torque including torque tests for the existing air driven impact wrenches. Because this violation was of very low safety significance and was entered into the corrective action program, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000373/2007007-02; NCV 05000374/2007007-02).

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From April 23, 2007 through May 4, 2007, and May 14 through 18, 2007, the inspectors reviewed twenty-six corrective action documents that were related to 10 CFR 50.59 evaluations or permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Ms. Susan Landahl and others of the licensee's staff, on May 18, 2007. The inspectors returned proprietary information reviewed during the inspection and the licensee confirmed that none of the potential report input discussed was considered proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
S. Landahl, Site Vice President
D. Enright, Plant Manager
T. Simpkin, Regulatory Affairs Manager
J. Bashor, Engineering Director
J. Rommel, Design Engineering Manager
B. Hilton, Design Engineering
P. Holland, Regulatory Affairs

Nuclear Regulatory Commission
D. Kimble, Senior Resident Inspector
F. Ramirez, Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000373/2007007-01; 05000374/2007007-01	NCV	Inadequate Procedure for Removal of Drywell Head Bolts (Section 1R17.b.1)
05000373/2007007-02; 05000374/2007007-02	NCV	Lack of Calibrated Air Wrench for Drywell Head Assembly Bolt Installation (Section 1R17.b.2)

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

Applicability Determination

50.59 Applicability Review Form for Revision 5 for LOS-RD-SR7; dated March 13, 2006.

10 CFR 50.59 Screenings

L02-316; EC338718; dated September 6, 2002

L04-154; EC 347065; dated June 23, 2004

L04-162; EC 349848; dated June 30, 2004

L04-197; Rod Control Management System Upgrade; Revision 0

L05-005; LOA-RR-101 Revision 16; LOS-AA-S101 Revision 28; dated January 6, 2005

L05-007; EC 348413; dated January 7, 2005

L05-032; LOP-DO-04; Treating Biological Growth in Diesel Fuel Oil; Revision 0

L05-134: EC 348851; dated May 13, 2005

L05-147; EC 350959; Scram Discharge Volume Vent and Drain Pilot Valve Replacement with AVCO model B7122-145; Revision 0

L05-171; Replace RMCS/RPIS/RWM with RCMS - Transponder Cards/Covers; Revision 0

L05-307; UFSAR Change Package LUCR-66; dated November 10, 2005

L05-311; EC 357748; dated November 22, 2005

L06-009; EC 357031; Installation of Auxiliary Reservoirs and Instrument Isolation Valves on the RR Motors; Revision 1

L06-037; Revised RCIC Room Station Blackout Temperature Transient; Revision 0

L06-074; LGA-003; Primary Containment Control; Revision 0

L07-010; EC 364131; Primary Containment Vent System (VQ) Piping Design Parameters; Revision 0.

L07-037; Unit 1 Compensation for Loss of Jet Pump Flow Signal for Jet Pump 19; Revision 0

L07-049; LOP-RH-17; Alternate Shutdown Cooling; Revision 0

10 CFR 50.59 Evaluations

SE L06-41; Remove Discussion of Loose Parts Monitoring System; dated February 15, 2006

SE L05-073; LaSalle Unit 2 Cycle 11 Reload Package; dated March 3, 2005

SE L07-92; Drywell Head Analysis L-002666 (Input Parameter Change); dated May 3, 2007

SE L07-93: Drywell Head Analysis L-002666 (Method of Evaluation Change); dated May 3, 2007

L06-244; EC 355023; Unit 2 Digital EHC Upgrade Project; Revision 0

Corrective Action Program Documents Generated As a Result of NRC Inspection

AR 00622067; 50.59 Evaluation Reference Error; dated April 26, 2007

AR 00624078; Level of Detail Discrepancy in UFSAR 4.2.3.14; dated May 1, 2007

AR 00624612; UFSAR Change Package Issue Identified During NRC Audit; dated May 2, 2007

AR 00629775; Identified Deficiencies in UFSAR Not Incorporated; dated May 14, 2007.

AR 00629887; 50.59 Documentation of Licensing Basis; dated May 15, 2007

AR 00630872; Incomplete Procedure Review; dated May 17, 2007

AR 00631107; Enhancement to SM-AA-300-1001 for Method 4 Dedication; dated May 18, 2007

Corrective Action Program Documents Reviewed

AR 00309929; Received Loose Parts Monitoring Alarm; dated March 8, 2005

AR 00315906; Loose Parts Monitor Alarm; dated March 22, 2005

AR 00332432; Received Loose Parts Monitoring Alarm; dated May 5, 2005

AR 00335394; Various Alarms Loose Parts Monitor; dated May 16, 2005

AR 00115562; Abnormal Banging and Rattling Noise on LPMS; dated July 15, 2002

AR 00462095; FME in RX Annulus; dated March 4, 2006

AR 00461195; FME Discovered in U1 RX; dated February 23, 2006

AR 00600273; Historical FME at Jet Pump No. 11; dated March 7, 2007

AR 00600756; Historical FME at Jet Pump No. 20; dated March 7, 2007

AR 00600600; Historical FME at Jet Pump No. 15; dated March 7, 2007

AR 00567395; CR 14-35 Classified 30 Day Rod due to Channel Distortion; dated December 10, 2006

AR 00455968; Three Rods Failed to Indicate Full in Following ATWS; dated February 20, 2006

Procedures

LOS-RD-SR7; Channel Interference Monitoring; Revision 5 and Revision 9

LOA-RR-101; Unit 1; Reactor Recirculation System Abnormal; Revision 22

LOS-AA-S101; Unit 1; Shiftly Surveillance; Revision 41

Miscellaneous Documents

RA06-07; Technical Requirements Manual Change - Loose Parts Detection System; dated February 15, 2006

NEDC-32975P-A; BWR Owners Group Licensing Topical Report - Regulatory Relaxation for BWR Loose Parts Monitoring Systems; Revision 0

Global Nuclear Fuel 000-0034-6783-SRLR; Supplemental Reload Licensing Report for LaSalle Unit 2 Reload 10 Cycle 11; Revision 1

GE-NE-0000-0022-8684-R1; LaSalle Units 1 and 2 SAFER/GESTR Loss-of Coolant Accident Analysis for GE14 Fuel; dated December 2004

GE-NE/GNF-000-0013-9020-01 "Surveillance Plan for GNF Thick/Thin Channel-Control Blade Interference Monitoring for BWR/2-5 (C/D- Lattice) Plants"; Revision 2

GE-NE-0000-0036-5084-RO; LaSalle 1 and 2 Off rated Analysis Below the PLU Power Level Power; dated February 2005

SIL No. 320; Recommendations for Mitigation of the Effects of Fuel Channel Bowing; Supplement 1; Supplement 2; and Supplement 3

SC05-06; 10 CFR Part 21 Communication; Updated Surveillance Program for Fuel Channel-Control Blade Interference Monitoring; dated July 14, 2005

SC05-06; 10 CFR Part 21 Communication; Updated Surveillance Program for Fuel Channel-Control Blade Interference Monitoring; dated September 26, 2006

51-9010082001; Control Rod Friction Surveillance Recommendations for AREVA NP Fuel Channels; dated June 27, 2006

NF0400111; Fuel Channel Distortion Monitoring Plan; Revision 1
GNF 0000-0028-6844-SFP GE14-Boraflex Spent Fuel Storage Rack Criticality Analysis for LaSalle Unit 2; dated September 2004
Holtec Report HI-931060; Criticality Safety Evaluation of the Spent Fuel Storage Racks with Postulated Degradation of Boraflex; dated February 1994
NF0400111; LaSalle Fuel Channel Bow Assessment and Monitoring Plans; dated March 21, 2006
AREVA 51-9010082-000; Control Rod Drive Friction Surveillance Recommendations for FANP Fuel Channels; dated January 26, 2006

UFSAR Changes
LUCR-72; dated February 13, 2006
LUCR-66; dated November 9, 2005

IR17 Permanent Plant Modifications (71111.17B)

Calculations
ATD-0351; RCIC Pump Room Temperature Transient Following Station Blackout With Gland Seal Leakage; Revision 2
L-001780; RHR Heat Exchanger - Cooling Water Orifice 1E 12-D304 A/B; Revision 0
LAS-1E22-F012; AC Motor Operated Gate Valve Calculation; Revision 2
NED-M-MSD-66; Seismic Qualification Reevaluation of the Motor Operated valves; Revision 0
L-003022; Seismic Qualification of Linde Pressure Regulator Model CS200/2-20F-250; Revision 1
L-002666; Evaluation of LaSalle Drywell Bolts to Justify Permanent Removal of Bolts; Revision 0
CECO-60Q-303; Evaluation of Drywell Bolt Torque to Justify Elimination of Bolt Tensioner; Revision 0
CECO-64Q-301; Bolt Removal Leakage Integrity of the LaSalle Drywell Head; Revision 0
CECP-60Q-304; Evaluation of Drywell Bolt Torque to Justify Elimination of Use of Bolt Tensioner; Revision 0
CBI Calculation for Refueling Head Assembly, Charge No. 73-6336/7; dated July 1974

Corrective Action Program Documents Generated As a Result of NRC Inspection

AR 00624485; Error - EC 364131 - VQ System Piping Design Parameters; dated May 2, 2007

AR 00629796; Wrong Material Allowable used in Calculation L-002666; dated May 15, 2007

AR 00629822; LOS-RD-SR7 Revised Without Performing a 50.59 Screening; dated May 15, 2007

AR 00630431; MA-AB-756-601 Missing Design Requirements; dated May 16, 2007

AR 00630337; MA-AB-756-600 Inadequate For Drywell Head Bolts; dated May 16, 2007

Commercial Grade Dedications

PE 49943; Pump, Priming, D.C. Fuel, with 115 VDC 3/4 H.P. Motor, for Emergency Diesel Engine; Revision 0

PE 54770; Rebuild/Repair Kit, Cylinder, for Safety Related Relief Valve Air Cylinder; Revision 0

Corrective Action Program Documents Reviewed

AR 00316845; Jet Pump Flow Indications; dated March 31, 2005

AR 00333331; Framatome Principal LOCA Analysis Parameters; dated May 9, 2005

AR 00353163; RCIC Electronic Governor Module Does Not Meet SBC Qualification; dated July 14, 2005

AR 00367856; System Leakage Test Specified by EC 352517 not Performed; dated August 29, 2005

AR 00354047; Error Found in Calculation ATD-0351 Revision 1; dated July 18, 2005

AR 00598803; Jet Pump 16 Main Wedge Wear; dated March 3, 2007

AR 00599100; Jet Pump 5 Main Wedge Wear; dated March 3, 2007

AR 00599119; Jet Pump 16 Rod Wear; dated March 4, 2007

AR 00599133; Jet Pump 18 Set Screw Gap Sizing; dated March 4, 2007

AR 00599169: Jet Pump 15 Main Wedge Movement and Rod Wear; dated March 4, 2007

AR 00599189; Resizing Data for Jet Pump 18 Set Screw Gaps; dated March 4, 2007

AR 00599707; Indication on Jet Pump 10 Restrainer Bracket Pad; dated March 5, 2007

AR 00601319; Indication on Jet pump 14 Restrainer Bracket Pad; dated March 8, 2007

AR 00606601; 10 CFR 50.59 Evaluation for Unit 2 Digital EHC EC 355023; dated March 20, 2007.

Drawings

M-87; P&ID Core Standby Cooling System Equipment Cooling Water System; Revision AK

VPF 3238-836; Weld Ends Carbon Steel Gate Valve; Revision 5

77U-401; 1" Valve Assembly Pressure Regulating; Revision E

M-66; P&ID Drywell Pneumatic System; Revision 0

Procedures

LOS-DG-Q1; 0 Diesel Generator A Cooling Water Pump ASME Section XI Test; completed on March 12, 2006

LOA-FLD-001; Flooding; Revision 7

LMP-GM-20; Flow Orifice Installation; Revision 7

MA-AB-756-601; Reactor Reassembly; Revision 8

MA-AB-756-600; Reactor Disassembly; Revision 7

MA-AB-RS-6-00600; Reactor Disassembly; Revision 0

MA-AB-RS-6-00601; Reactor Reassembly; Revision 0

Miscellaneous Documents

Report E06-255; VT-2 Examination Record 0DG005; dated March 12, 2006

Report E06-253; VT-2 Examination Record 1E12-D304B Orifice; dated March 9, 2006

DBD-LS-M11; Flood Protection; Revision B

Work Order 00661571; Install New Orifice Plate in RH Division 2; dated March 6, 2006

Work Order 00632225; L2R10 - Disassemble and Assemble Reactor Vessel; dated March 14, 2005

Work Order 00708429 09; MPT LTP-1600-11 and LTP-100-13; dated May 4, 2005

Work Order 00790832 01;LOS-AA-S201; dated March 21, 2005

Work Order 00704688; HPCS System Operation PMT; dated March 1, 2006

Work Order 00722095; 1E22-F012 Water Leak Rate Test; dated March 1, 2006

Work Order 00712815 05; Op PMT Perform Functional and Leak Check 1IN35; dated March 14, 2006

Work Order 00348525; Performance Testing Linde Regulator; dated April16, 2004

LTS-600-8: Reactor Vessel Internals In-service Inspection During Reactor Refueling; Revision 15

Specification T-3763; Mechanical and Structural Work Specification; Revision 14

Specification J-2534 Section 2; Primary Containment Steel Liner; dated August 1, 1975
GE 26A6357; Rod Control Management System Requirements Specification; LaSalle 1 and 2; Revision 4
XGEN -2004-12; Assessment of the Root Cause of the Observed Shrinkage of the LaSalle Unit 2 Inlet Mixers in the Region of the Labyrinth Seal and the Proposed Solution; dated December 20, 2004
SIR-04-181; LaSalle Unit 2 Jet Pump Inlet Mixer Repair - Third Party Evaluation; dated January 21, 2005
LUCR-45; UFSAR Changes for EC 348413; dated January 3, 2005
Rising Stem MOV Data Sheet; dated October 15, 1999
EMF-3208(p); LaSalle Units 1 and 2 Principal LOCA Analysis Parameters; dated May 2005
245C9542; MK6 BWR Functional Diagram; Revision D
342A6215CMP; GE D and C Software Configuration Management Plan; Revision 1
342A6215AMP; GE D and C Software Management Plan; Revision 1
342A6215VV; GE D and C Software Verification and Validation Plan; Revision 1
GEH 6126A; HMI for Speedtronic Controls; dated February, 2002
GEK 6421; Mark VI System Manual; Vol. I and II; dated 2006
GEK 111056; General Description of BWR Mark VI Controls; dated October 2004
GEI 100657C; Mark VI Maintenance Procedures for Replacing Circuit Boards on Nuclear Lineups; dated 2005
LST-2006-006; Unit 2 DEHC Power Ascension Test; Revision 2

Modifications
EC 338718; Remove Valve Disc EDG 0 Cooler - Inlet; dated September 6, 2002
EC 347065; RHR Division 2 Service Water Orifice 1E12-D304B Resized; dated October 5, 2005
EC 348413; Install Labyrinth Seal Jet Pump Mixers and Improved Wedges/Hardware; dated February 15, 2005
EC 348851; 1E22-F012 Valve Motor Gearing Change; Revision 0
EC 349848; Nitrogen Bottle Bank Pressure Regulator Valve Design Change; dated July 2, 2004
EC 355023; Unit 2 Digital Electro Hydraulic Control Upgrade; Revision 2
EC 357748; Generic Use of Open Root Welding for ANSI/ASME Piping Systems; dated November 22, 2005
EC 350959; Scram Discharge Volume Vent and Drain Pilot Valve Replacement with AVCO Model B7122-145; Revision 2
EC 357032; Unit 1 Reactor Recirculation Oil Reservoir Upgrade; Revision 1
EC 365060; Design Analysis to Justify Removal of Drywell Head Bolts During Mode 3; Revision 0

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
AR	Assignment Report
CFR	Code of Federal Regulations
DHA	Drywell Head Assembly
ft-lbs	Foot Pounds
IMC	Inspection Manual Chapter
kips	1000 Pounds Force
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
psig	Pounds per Square Inch Gauge
SDP	Significance Determination Process