December 29, 2008

Mr. Charles G. Pardee
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
Chief Nuclear Officer (CNO)
AmerGen Energy Company, LLC
4300 Winfield Rd.
Warrenville, IL 60555

SUBJECT: LIMERICK GENERATING STATION – EVALUATIONS OF CHANGES, TESTS OR EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000352/2008010 AND 05000353/2008010

Dear Mr. Pardee:

On November 20, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at Limerick Generating Station. The enclosed inspection report documents the inspection results, which were discussed on November 20, 2008 with Mr. C. Mudrick, Site Vice President, 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. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

Based on the results of this inspection, no findings of significance were identified.

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 the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No: 50-352, 50-353
License No: NPF-39, NPF-85

Enclosure: Inspection Report 05000352/2008010 and 05000353/2008010
w/Attachment: Supplemental Information

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DATE	12/23/08		12/24/08		12/29/08			

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REGION I

Docket No:	50-352, 50-353
License No:	NPF-39, NPF-85
Report No:	05000352/2008010 and 05000353/2008010
Licensee:	Exelon Generating Company, LLC
Facility:	Limerick Generating Station, Unit 1 & 2
Location:	Sanatoga, PA 19464
Dates:	November 3 to November 20, 2008
Inspectors:	K. Mangan, Senior Reactor Inspector (Team Leader) A. Ziedonis, Reactor Inspector M. Balazik, Reactor Inspector P. McKenna, Reactor Inspector (in-training)
Approved by:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000352/2008010, 05000353/2008010; 11/03/2008 - 11/20/2008; Limerick Generating Station; Engineering Specialist Plant Modifications Report

The report covers a two week inspection of the evaluations of changes, tests, or experiments and permanent plant modifications. It was conducted by three region-based engineering inspectors. No findings of significance were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

None.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (24 samples)

a. Inspection Scope

The team reviewed five safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance with 10 CFR 50.59. In addition, the team determined whether Exelon had been required to obtain NRC approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information, including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications, and plant drawings, to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Evaluations," as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of nineteen 10 CFR 50.59 screenings, applicability determinations and equivalency evaluations for which Exelon had concluded that no safety evaluation was required. These reviews were performed to assess whether Exelon's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample of issues that were screened out included design changes, temporary alterations, procedure changes, and setpoint changes.

The five safety evaluations that were reviewed were the only safety evaluations Exelon had performed during the time period covered by this inspection (i.e., since the last modifications inspection). The screenings and applicability determinations were selected based on the risk significance of the associated structures, systems, and components (SSCs).

In addition, the team compared Exelon's administrative procedures, used to control the screening, preparation, review, and approval of safety evaluations, to the guidance in NEI 96-07 to determine whether those procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations, screenings, and applicability determinations are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications (10 samples)

.2.1 Feedwater Flow Conversion Constant Adjustment, Unit 1

a. Inspection Scope

The team reviewed a modification that changed the Unit 1 feedwater flow venturi differential pressure to flow conversion constants. The modification installed a new constant for the three feedwater flow orifices. The constants were determined based on a comparison between multiple secondary plant indications, indicated feedwater flow and reactor power. As a result of the modification, indicated feedwater flow was raised to match secondary plant indications. The review was performed to verify that the design bases, licensing bases and performance capability of the feedwater flow indication had not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases related to accuracy of the feedflow indication. The team reviewed the methodology used to establish the feedflow indication accuracy, as well as the inputs used in establishing the basis for the accuracy of the methodology. The resulting constant adjustments for the flow venturi were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team verified whether calculations, analysis, procedures, and the UFSAR were properly updated with revised design information. The team also reviewed the design, testing and calibration programs to determine if the revised calibration techniques had been reviewed by Exelon in accordance with their quality assurance design procedures. Finally, the team conducted interviews with engineering staff to verify whether the indicated feedflow maintained the accuracy assumed in the design and licensing assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.2 Abandoned Loose Parts Monitor System, Unit 2

a. Inspection Scope

The team reviewed a modification that removed the loose parts monitor from service. The modification removed acoustic monitoring devices, changed alarm indications in the control room and left some equipment, such as wiring, retired in place. The review was performed to verify that the design and licensing bases had not been inadvertently changed by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the removal of the monitor was consistent with the design and licensing bases related to the requirements to ensure foreign material in the reactor vessel would not damage structures, systems or components (SSC) such as fuel rods.

The team reviewed the BWR owners group Topical Report and the associated NRC safety evaluation that discussed removal of the equipment to determine if assumptions in the documents were consistent with Unit 2 design and operation. The team also verified that the equipment retired in-place did not affect the operability of other SSCs required to respond to design basis events. The team verified that drawings, calculations, procedures, and the UFSAR were properly updated with revised design information, and verified that applicable operating guidance had been changed. Finally, the team conducted interviews with engineering staff to verify whether other SSCs impacted by the change would continue to function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.3 Installation of HPCI Suppression Pool Level Transmitter Head Chambers, Unit 2

a. Inspection Scope

The team reviewed a modification that installed head chambers on the high pressure coolant injection (HPCI) suppression pool level transmitter indication system. The modification was performed as a corrective action to eliminate the cause of previous calibration problems with the instruments and was designed to prevent water from entering the dry portion of the instrumentation tubing. The chambers are used to aid in the calibration of the suppression pool level indicating equipment. The review was performed to verify that the design bases, licensing bases, and performance capability of the level indication not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed if the modification was consistent with requirements in the design and licensing bases. This review included assessing if the component safety classification had been maintained, specifically the structural adequacy of the design. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team also verified whether drawings, calculations, calibration instrumentation sheets, and calibration procedures were properly updated based on the new equipment configuration. The team reviewed the post-modification testing to verify that the suppression pool level transmitter system was working properly. Additionally, the team conducted interviews with engineering staff to verify whether the affected SSCs functioned in accordance with the design assumptions, and to verify if the modification corrected the previously identified calibration problems. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.4 Replacement of Residual Heat Removal Testable Check Valve HV-51-1F050B, Unit 1

a. Inspection Scope

The team reviewed a modification that replaced the residual heat removal (RHR) system testable check valve with a new valve. The modification was performed as preventive maintenance to address concerns with wearing of the seating surfaces on the installed valve. The new valve also included modifications from the original design that had been previously evaluated under the Exelon 50.59 process. The review was performed to verify that the design bases, licensing bases, and performance capability of the RHR system had not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the replacement valve was consistent with the design and licensing bases, including whether the welds required to install the valve in the RHR system were performed in accordance with the American Society of Mechanical Engineers (ASME) piping classification requirements. Design assumptions used for fabrication were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team verified whether drawings, calculations, analyses, and the UFSAR were properly updated with the revised design information. The team also reviewed the post-modification testing to verify the testing was in accordance with ASME Section XI requirements, and to verify that test results appropriately justified system operability. Finally, the team conducted interviews with engineering staff to determine if the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.5 Replacement of 5-Micron Elements for Control Rod Drive Water Filters, Units 1 and 2

a. Inspection Scope

The team reviewed a modification regarding the replacement of the control rod drive (CRD) system discharge filters AF204 and BF204. The modification replaced the existing 20-micron filters with 5-micron filters to improve reactor recirculation pump seal performance. The review was performed to verify that the design bases, licensing bases and performance capability of the CRD system had not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with the design and licensing bases related to the CRD system. The review included determining if the component safety classification and flow requirements for CRD system were maintained. Supporting mechanical calculations and analyses for the instrumentation setpoints and maximum differential pressure were reviewed to ensure design limits were not exceeded. The design assumptions associated with the analyses were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team verified whether drawings, calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. The team reviewed the post-modification testing and conducted interviews with engineering staff to verify the affected SSCs would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.6 Replacement of RCIC Relief Valve PSV-050-1F018, Unit 1

a. Inspection Scope

The team reviewed a modification regarding the replacement of the barometric condenser/lube oil cooler relief valve PSV-050-1F018, with an alternate relief valve design. The review was performed to verify that the design bases, licensing bases and performance capability of the reactor core isolation cooling (RCIC) system had not been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the replacement relief valve was consistent with assumptions in the design and licensing bases. The team reviewed supporting mechanical calculations and analyses to assess if the component safety classification, relief capacity, seismic analysis, material equivalency were maintained in accordance with the design. Assumptions used in the analyses were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR. The team verified whether drawings, calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. The team reviewed the post-modification testing to verify whether the affected SSCs would function in accordance with the design assumptions. In addition, the team interviewed the responsible design engineers and walked down the relief valve to detect possible abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.7 Revision of Analysis for Post-Accident Heat Sources and Sinks, Units 1 and 2

a. Inspection Scope

The team reviewed changes to the analysis for the emergency service water (ESW) minimum flow required for the residual heat removal (RHR) room coolers. The analysis revised post-accident heat sources and heat sinks which were used as an input to determine the ESW minimum acceptable flow to the RHR room coolers. The review was performed to verify that the design and licensing bases, and performance capability of the ESW or RHR systems had not been degraded by the revised analysis and subsequent modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed selected design inputs within the analysis to ensure that the minimum credited ESW flow to the various RHR room coolers would maintain each of the rooms at an acceptable temperature per the RHR room equipment environmental qualification specifications. Design assumptions were reviewed to evaluate whether they were technically appropriate and consistent with the UFSAR such that minimum required flows were achieved. The team verified whether calculations, analyses, procedures, and the UFSAR were properly updated with revised design information and operating guidance. The team reviewed the post-modification testing to verify whether the affected SSCs would function in accordance with the design bases assumptions and plant operating procedures. Finally, the team conducted interviews with engineering staff to verify whether the affected RHR room coolers would function in accordance with the design assumptions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.8 Replacement of Emergency Diesel Generator K1 Relay, Unit 1

a. Inspection Scope

The team reviewed a modification to replace the K1 relay in emergency diesel generator (EDG) D-14 subsequent to a failure of the relay that was experienced during surveillance testing. The new relay design was installed because the original relay was no longer available. The review was performed to determine whether the design bases, licensing bases, and performance capability of the relay had been degraded by the replacement. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases related to the EDG voltage regulator circuitry. The team reviewed licensing and design documents to determine the component safety classification, auxiliary contactor functional requirements, and seismic requirements for the K1 relay mounting adapter plate. The team evaluated whether the design requirements were maintained in the supporting evaluations and analyses, and whether they were consistent with the UFSAR. The team reviewed selected evaluations, drawings, analyses, procedures, and the UFSAR to determine whether they had been properly updated with revised design information. The team evaluated the post-modification test to determine whether the K1 relay would function in accordance with design requirements. In addition, the team interviewed the responsible design and system engineers to discuss the K1 relay design requirements, surveillance test failure, extent-of-condition evaluations, and evaluation of the subsequent replacement. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.9 Replacement of Emergency Service Water Gate Valve 011-1010, Unit 1

a. Inspection Scope

The team reviewed a modification to replace the ESW gate valve due to excessive seat leakage from the valve that was currently installed. The review was performed to determine whether the design bases, licensing bases, and performance capability of the ESW system had been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases related to the ESW system. This review included assessing if the component safety classification and specific safety function were maintained. Additionally, the technical basis for the allowable leakage was reviewed to evaluate if operability of the ESW system was maintained. The team reviewed design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected drawings, analyses, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification test to verify that the appropriate ASME code acceptance criteria were applied to verify leak-tightness and weld acceptance. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. Finally, the team walked down the ESW valve to detect any potentially abnormal installation conditions. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

.2.10 Replacement of Motor Operator for ‘A’ Reactor Feedwater Inboard Valve, Unit 2

a. Inspection Scope

The team reviewed a modification to replace the motor operator on the ‘A’ reactor feedwater inboard maintenance isolation valve (2F011A). The review was performed to determine whether the design bases, licensing bases, and performance capability of the feedwater system had been degraded by the modification. Additionally, the equivalent 10 CFR 50.59 screen associated with this modification was reviewed as described in section 1R17.1 of this report.

The team assessed whether the modification was consistent with assumptions in the design and licensing bases related to the feedwater system. The team reviewed licensing and design documents to determine the component safety classification, scoping applicability for primary containment isolation function, and adequacy of motor electrical characteristics. The team reviewed design assumptions in the supporting evaluations and analyses to determine whether they were technically appropriate and consistent with the UFSAR. The team reviewed selected drawings, analyses, procedures, and the UFSAR to determine whether they were properly updated with any revised design information. The team evaluated the post-modification test to verify that the motor and valve would successfully perform their functions. In addition, the team interviewed the responsible design and system engineers to discuss the modification and the design requirements. The documents reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of condition reports associated with 10 CFR 50.59 and plant modification issues to determine whether Exelon was appropriately identifying, characterizing, and correcting problems associated with these areas. The team also evaluated whether the planned or completed corrective actions were appropriate to address the deficiencies. In addition, condition reports written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. C. Mudrick, Site Vice President, and other members of Exelon's staff on November 20, 2008. The team verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

C. Mudrick	Site Vice President
E. Callan	Plant Manger
R. Dickinson	Director, Engineering
P. Gardner	Director, Operations
R. Kreider	Manager, Regulatory Assurance
R. Harding	Manager, Regulatory Assurance
S. Bobyock	Manager, Plant Engineering

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

None

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations

LG2006E001, Risk Informed ISI Program Revision 2 with Break Exclusion Regions, Rev. 0
LG2007E001, S71.0.A Defeating the Mode Switch to Shutdown SCRAM, Rev. 0
LG2007E002, Application of TRACG04 for Stability Analysis, Rev. 0
LG2007E003, Fuel Handling Equipment Upgrades, Rev. 0
LG2007E005, Change Inspection Frequency of Low Pressure Turbine to 110,000 Operating Hours, Rev. 0

10 CFR 50.59 Screened-out Evaluations

LG2003S120, Reanalysis of RHR Rooms Post-LOCA and Miscellaneous Corrections, Rev. 0
LG2007S003, Reactor Vessel Thermal Transient Monitor, Rev. 0
LG2008S006, Operability Evaluation OPE-08-004, Rev. 0
LG2007S009, Use of Ultra Low Sulfur Diesel Fuel, Rev. 0
LG2007S015, 2R09 103 Disconnect Switch Jumper Installation, Rev. 0
LG2007S030, Administrative Clearance to Defeat 2C RFP LP Bearing Vibration Alarm, Rev. 0
LG2007S042, Procedural Revisions to Loss of Offsite Power / and Loss of All AC Power, Rev. 1
LG2008S010, Limerick Unit 1 Reload Cycle 13 Design/Management, Rev. 1
LG2008S032, Cut and Plug RHRSW and ESW Drain Lines in Valve Pit 216, Rev. 0

Modification Packages (* denotes Modification sample)

ECR 99-001392, Limitorque Motors – Generic ECR per Target 2000 Criteria, Rev. 2
*ECR 03-00094, RHR Room Coolers-ESW Minimum Flow, Rev. 1
*ECR 03-00432, HPCI Suppression Pool LT Head Chambers, Unit 2, Rev. 0

*ECR 04-00168, EDG K1 Contactor, Rev. 0
ECR 06-00287, Limerick Unit 2 Cycle 10 Core Reload Design, Rev. 0 & 1
*ECR 06-00295, ESW Gate Valve has Excessive Leakage, Rev. 0
*ECR 06-00463, Replace HV-51-1F050B, Rev. 2
*ECR 06-00499, Unit 1 Feedwater Flow Conversion Constant Adjustment, Rev. 0
ECR 06-00510, Request Changes to UFSAR Section 10.2.3.6 (1R12/20-LIST), Rev. 0
*ECR 07-00077, 2A Reactor Feedwater Isolation Valve Motor Changeout, Rev. 0
ECR 07-00105, 2R09 103 Disconnect switch Jumper Installation, Rev. 0
ECR 07-00115, Evaluate Limitorque Motor for HV-009-212-OP, Rev. 0
ECR 07-00334, CDBI FASA – E-1412 (Missing/Broken Cable Ties), Rev. 0
ECR 08-0228, Piping Mods in Valve Pit at Manhole 216, Rev. 0
*ECR 07-00242, Alternate Replacement for Relief Valve PSV-050-1F018 (RCIC), Rev. 0
*ECR 08-00046, 5-Micron Replacement Elements for CRD Water Filters, Rev. 0
*ECR 08-00157, Abandon Loose Parts Monitor system for LGS U2, Rev. 0

Calculations & Analysis

22A6249AA Control Rod Drive System Design Specification Data Sheet, Rev. 12
EC-2619, Seismic Report- AG-Crosby 1 x 1-1/2 Style 972 Relief Valve, Rev. 1
EC-364586, Ultra Low Sulfur Fuel Technical Evaluation, dated 02/16/07
EE-94LGS, Proper Calibration of Feedwater Flow Elements (FE-006-1(2) N001A,B,C (GE SIL 452), Rev. 14
ER-9605, Missile Probability Analysis Methodology for Limerick Generating Station, Units 1 & 2 with Siemens Retrofit Turbines, Rev. 2
Operability Evaluation 06-006, Emergency Diesel Generators and Auxiliaries/EDG Fuel Oil Storage and Transfer, Rev. 5
LEAM-0016, Risk Informed In-service Inspection Evaluation Final Report, Rev. 0
LM-0007, Diesel Generator Fuel Oil Storage, Rev. 1 & Rev. 1A
LM-0414, RHR Rooms Analysis, Rev. 1
LM-0562, CRD Flow-rates and System Pressures, Rev. 2
LM-0663, Diesel Generator Day Tank Minimum Level, Rev. 0 & Rev. 1
LS-0266, Structural Evaluation of Refueling Platform Upgrade, Rev. 1
M-20-08, Diesel Generator Fuel Oil Storage, Rev. 3
NEDC-32975P-A, BWR Owners' Group Licensing Topical Report - Regulatory Relaxation for Loose Parts Monitoring Systems, Rev. 0
NEDE-21821-A, Boiling Water Reactor Feedwater Nozzle/Sparger Final Report, February 1980
SIR-00-021, Assessment of Feedwater Nozzle Thermal Sleeve Seal Refurbishment Intervals for Limerick Units 1 and 2, Rev. 0
WT-36219, Wall Thickness Procedure, Rev. 1

Condition Reports (* denotes NRC identified during this inspection)

AR 00074579	AR 00583915	AR 00832600	AR 01484602
AR 00142874	AR 00660875	AR 00842765*	AR 01504969
AR 00253342	AR 00741893	AR 00842990*	AR 01565145
AR 00501705	AR 00751491	AR 00843005*	AR 01618065
AR 00511802	AR 00761650	AR 00845148*	AR 01662955
AR 00516562	AR 00783306	AR 00847138*	AR 01665645
AR 00554242	AR 00793332	AR 00847286	

Drawings

8031-M-10, Sht. 1 and 2, Service Water, Rev. 62 and 56
8031-M-11, Sht. 2 and 3, Emergency Service Water, Rev. 81 and 52
8031-M-12, Sht. 1 and 6, Residual Heat Removal Service Water, Rev. 64 and 2
8031-M-41, Sht. 4, Nuclear Boiler, Rev. 42
8031-M-46, Control Rod Drive Hydraulic – Part A & B, Rev. 51 & 45
8031-M-50, Sht. 1, P&ID RCIC Pump Turbine, Rev. 36
8031-M-71-65, Sht. 1A, D11 Diesel Generator Control Schematic, Rev. 18
9 1198 01 910, Nuclear (EDG) Exciter/Regulator, Rev. E
C99973 Sht. 1, Pressure Relief Valve, Rev. A
HBB-149-E1, Sht. 1, Isometric-RCIC Lube Oil Cooler, Rev. 6
SIM-M-0012, Sht. 1, Emergency Service Water/RHR Service Water Overview, Rev. 9

Procedures and Completed Surveillance Tests

ER-AA-330-001, Section XI Pressure Testing, Rev. 8
LA-AA-104-1000, Exelon 50.59 Resource Manual, Rev. 4
LS-AA-104, Exelon 50.59 Review Process, Rev. 5
LT-055-2N062B, Instrumentation Calibration Sheet, Rev. 2
LT-055-2N062F, Instrumentation Calibration Sheet, Rev. 2
M-006-005, Feedwater Flow Elements Inspection, Rev. 1
M-046-004, Cleaning or Replacement of CRD Water Filter Elements, Rev. 0
M-097-044, New Fuel Receipt and Inspection, Rev. 20
MA-AA-716-040, Control of Portable Measurement and Test Equipment Program, Rev. 5
MA-AA-716-230-1002, Vibration Analysis/Acceptance Guideline, Rev. 1
NE-C-210, Establishing and Maintaining Component Classification, Rev. 6
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698567-03, FASA Report – Permanent Plant Modifications and 10 CFR 50.59s

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C0214847, FE-006-1N001B – Clean and Inspect, dated 03/28/06
C0215638, 'B' Feedwater Flow Transmitter Cal's, dated 04/01/06
C0217785, Replaced 8" gate valve 011-1010 and Adjacent Piping, dated 04/23/08
C0219553, Unit 1 Feedwater Flow Element Coefficient Change, Rev. 1
C0220492, Calibrate Inst. Magnetic Breaker, dated 06/04/07
C0221892, Replace HV-51-1F050B/ ECR 06-00463, dated 08/23/07
C0223950, Replace CRD Drive Water Filters per ECR 08-00046, dated 04/22/08
C0224084, Rework/Replace Section of 30" HBC-091-03 Line, dated 11/07/08
C0225127, ESW/RHRSW NDE & Repairs, dated 10/13/08
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R0777908, PSV-050-1F018 Remove/Test/Replace/ 1" x 1.5" Relief Valve, dated 11/26/07

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ILD INC Limerick Unit 1 Best Estimate Power Monitor, Rev. 3
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LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRD	Control Rod Drive
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
HPCI	High Pressure Coolant Injection
IP	Inspection Procedure
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SSC	Structures, Systems and Components
UFSAR	Updated Final Safety Analysis