June 20, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generating Company
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION - NRC INSPECTION REPORT 05000352/2001-004, 05000353/2001-004

Dear Mr. Kingsley:

On May 12, 2001, the NRC completed an inspection at your Limerick Generating Station Units 1 and 2. The enclosed report documents the inspection findings which were discussed on May 25, 2001, with Mr. R. Braun, Plant Manager, and other members of your staff.

This 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, the inspectors identified two issues of very low safety significance (Green). One of these issues was determined to involve a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this letter, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Limerick facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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/NRC/ADAMS/index.html (The Public Electronic Reading Room).

Sincerely,

/RA/

Mohamed Shanbaky, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos.: 05000352; 05000353

License Nos: NPF-39; NPF-85

Enclosure: Inspection Report 05000352/2001-004, 05000353/2001-004
Attachments: (1) Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket Nos:	05000352; 05000353
License Nos:	NPF-39, NPF-85
Report No:	05000352/2001-004, 05000353/2001-004
Licensee:	Exelon Generation Company
Facility:	Limerick Generating Station, Units 1 & 2
Location:	Evergreen and Sanatoga Roads Sanatoga, PA 19464
Dates:	April 1, 2001 thru May 12, 2001
Inspectors:	A. Burritt, Senior Resident Inspector B. Welling, Resident Inspector J. Noggle, Senior Health Physicist L. Scholl, Senior Reactor Inspector
Approved by:	Mohamed Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000352/2001-004, IR 05000353/2001-004, on 04/01-05/12/2001, Exelon Generation Company, Limerick Generating Station, Units 1 and 2. Personnel Performance and Permanent Plant Modifications.

This report was conducted by resident inspectors, a regional reactor inspector and a regional health physics inspector. The inspection identified two Green findings. 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 are indicated by "no color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- **Green.** Operators did not conduct a thorough pre-job briefing prior to a non-routine feedwater control system manipulation. Consequently, the operators were not prepared to respond to an unexpected drop in reactor vessel water level in a manner consistent with training and operational transient procedures.

  The finding was of very low significance because an automatic recirculation pump runback occurred which allowed restoration of proper reactor vessel water level prior to exceeding the low reactor vessel water level reactor scram set point. (Section 1R14)

Cornerstone: Mitigating Systems

- **Green.** Six of the 2N SRV outlet flange studs were missing or loose, and torque values on outlet flange studs of all other Unit 2 SRVs were found to be substantially below the specified range. Exelon's root cause investigation indicated that the safety relief valve outlet flange studs loosened as a result of use of a gasket that was subject to excessive creep, inadequate torque values, and poor torque value determination guidance. The inspectors identified a violation of 10 CFR 50 Appendix B, Criterion III, "Design Control." This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy.

  The finding was of very low significance because the SRV outlet flange joint integrity was maintained. (Section 1R17)

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Report Details

Summary of Plant Status

Unit 1 began this inspection period operating at 100% power and remained at or near that power level except for planned testing and control rod pattern adjustments.

Unit 2 began this inspection period operating at 83% power, in an end-of-cycle coast down. On April 4, 2001, operators shut down the unit for refueling outage 2R06. On April 19, the reactor was taken critical, and on April 23, the unit was restored to 100% power. The unit remained at or near that power level, except for planned testing and control rod pattern adjustments.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed a partial walkdown on the D11, D13, and D14 emergency diesel generators, while the D12 emergency diesel generator was out of service for planned maintenance. The inspectors used procedures' S92.9.N, "Routine Inspection of the Diesel Generators," and S92.1.N, "Diesel Generator Setup for Automatic Operation Following Maintenance."

The inspectors also performed a walkdown of Unit 2 "A" core spray train, while the Unit 2 "B" core spray train was out of service for planned maintenance. The inspectors used piping and instrumentation diagram 8031-M-52.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured fire-areas at both Limerick units to assess Exelon's control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The inspectors reviewed the respective Pre-Fire Action Plan procedures and Section 9A of the Updated Final Safety Analysis Report. The fire-areas included:

- D24 diesel generator cell (fire-area 86)
- Recombiner access area and control structure chiller area (fire-area 1)

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed heat exchanger performance testing per routine test RT-2-12-391-2, for the 2B residual heat removal system heat exchanger. The inspectors reviewed documentation for potential deficiencies which could mask degraded performance and common cause performance problems. The inspector also reviewed the previous records associated with the spray pond sludge inspections to assess whether the licensee was meeting their commitments to Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed Exelon's risk management and risk assessments as required by 10 CFR 50.65 (a)(4) of the following emergent and planned maintenance activities. The inspectors reviewed the Sentinel on-line risk assessments, risk management activities, work control center planning and scheduling, and emergent work-related activities.

- 2B feed pump and 2C condensate pump out of service
- Unit 2 Div. III Emergency Core Cooling System initiation logic fuse failure
- 2B core spray system outage
- D12 emergency diesel generator overhaul

b. Findings

No findings of significance were identified.

1R14 Personnel Performance Related to Nonroutine Plant Evolutions and Events (71111.14)

a. Inspection Scope

The inspectors reviewed personnel performance associated with a non-routine plant evolution of restoring a feedpump to service following maintenance.

The inspectors discussed the event with operations and engineering personnel, and reviewed the following documents:

- Performance Enhancement Program (PEPs) I0012403, I0012420
- Exelon Nuclear Operations Action Notification 2001-04
- Action Requests A1310420, A1310412
- Operational Transient (OT) Procedure OT-100, "Reactor Low Level," and Bases Document
- Operations Lessons Learned
- Plotted plant computer data and operator logs

b. Findings

Operators did not conduct a thorough pre-evolution briefing prior to non-routine feedwater control system manipulations. Consequently, operators were not adequately prepared to respond to an unexpected drop in reactor vessel water level, which led to an automatic recirculation system runback.

On March 22, 2001, the Unit 2 operators experienced an unexpected drop in reactor vessel water level from 35 inches to 30 inches. This occurred during feedwater level control manipulations, while the operators were preparing to restore a feed pump to service following maintenance. The operators entered OT-100, "Reactor Low Level," as required, but were not set up to reduce recirculation flow as required by Step 1 of the OT. The operators took manual control of the feed pumps (Step 2 of OT-100 immediate actions). The operator's manipulations of the feedpump controls in manual caused reactor vessel water level to first go to the high-level alarm set point and then through the low-level alarm set point to the low level recirculation system automatic runback set point. This caused both recirculation pumps to automatically runback. Following the runback, the operators were able to restore proper reactor vessel water level. The original drop in reactor water level was caused by a degraded master feedwater level controller.

The pre-job briefing did not fully address contingency actions during an evolution that had the potential to rapidly change reactor vessel water level. The briefing did not meet department guidance and expectations with respect to assigning roles and responsibilities for potential transients, such as reducing recirculation system flow (Step 1 of OT-100 immediate actions) on low reactor level. It also did not include industry operating experience and discussion of previous experience at Limerick with performing this specific feedwater control evolution.

Operations personnel reviewed operator response to the event and identified two issues. First, the operators were unable to complete the immediate actions of OT-100. Although there were two reactor operators and two senior reactor operators involved in the evolution, they were not able to reduce power using the recirculation pumps. The OT-100 Bases Document states that the fastest way to terminate a reactor vessel water level decrease is to reduce recirculation flow. Exelon attributed the nonperformance of this step to the time constraints of the transient and the poor pre-job brief. Secondly, operations personnel noted that operators did not appear proficient in manual feedwater control manipulations. The review revealed that operators did not use all available parameters and may have been overly focused on reactor level. The review concluded that manual control of the feedpumps caused the recirculation pumps to runback. The inspectors also noted that the operators caused a second high reactor vessel water level condition following the pump runback because for 30 seconds they continued to feed the reactor at a flow rate that was higher than the flow rate prior to the initial transient. Exelon initiated a number of corrective actions for the event and documented them in PEP I0012403.

In summary, the operators did not conduct a thorough pre-job briefing prior to a non-routine feedwater control system manipulation. Consequently, the operators were not prepared to respond to an unexpected drop in reactor vessel water level in a manner consistent with training and operational transient procedures. The less than thorough pre-job briefing prior to performing a non-routine feedwater control system manipulation was more than a minor issue, because it had a credible impact on safety in that if the operator's manipulations of the feedwater control system with a degraded master controller were slightly different, they could have caused a high reactor vessel water level condition which would have led to main turbine and feed pump high level trips and a scram with loss of feedwater. This less than thorough pre-job briefing prior to performing a non-routine feedwater control system manipulation affects the Initiating Events cornerstone because it could cause or increase the frequency of an initiating event. This pre-job briefing issue was determined to be of very low safety significance (Green) by the Reactor Inspection Findings for At-Power Situations Significance Determination Process because an automatic recirculation pump runback occurred which allowed restoration of proper reactor vessel water level prior to exceeding the low reactor vessel water level reactor scram set point. The inspectors concluded that the less than thorough pre-job briefing prior to performing a non-routine feedwater control system manipulation issue did not constitute a violation of NRC requirements. **(FIN 05000353/2001-004-01)**

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the technical adequacy of operability evaluations associated with the following plant equipment conditions:

- 2B residual heat removal heat exchanger missed performance test
- 2H safety relief valve leakage as indicated by high tailpipe temperature

The inspectors reviewed the applicable action request documents and referred to Exelon Nuclear Operations Manual Chapter 11.1, Operability.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspectors reviewed an Exelon root cause investigation into the relaxation and loosening of safety relief valve (SRV) outlet flange studs. Following the Unit 2 SRV actuation in February 2001, Exelon discovered that six of the 2N SRV outlet flange studs were missing or loose, and torque values on outlet flange studs of all other Unit 2 SRVs were substantially below the specified range. This inspection began in the period of Inspection Report 2001-003, and was tracked as an Unresolved Item (URI) 05000353/2001-003-02. The inspectors examined corrective action document PEP I0012312, non-conformance report (NCR) 01-00242, evaluation A1306169, and discussed the issue with the investigation leader and other station personnel.

b. Findings

An Exelon root cause investigation identified the following main causes: the use of an inappropriate gasket subject to excessive creep, inadequate torque values, and poor torque value determination guidance.

Exelon noted that the inappropriate gasket and inadequate torque values resulted from incomplete engineering analysis activities. In 1990, the original gasket was replaced with a gasket of a different filler material without a thorough engineering evaluation of the new material. Recent testing of the gasket showed that it did not fully crush under the torque values applied and was subject to much higher creep. In 1987, the licensee lowered the torque value for the outlet flange based on a vendor-supplied analysis. At that time the licensee's engineering endorsement of the vendor-supplied analysis did not consider the potential effects on the gasket when the stud pre-load was reduced.

Exelon also found that the current specifications for establishing torque values did not provide appropriate controls for accepting torque values provided by vendors. In 2000, maintenance planners lowered the Unit 1 torque value to 230 foot-pounds, based on an SRV vendor drawing. Although permitted by a station specification, the value was unacceptably low for the gasket used at Limerick, and was not evaluated by engineering. As a corrective action, the investigation team recommended that the station develop a methodology for engineers and maintenance planners to assure that torque values are appropriate, including those provided by vendors.

The loosening of SRV outlet flange studs was more than a minor issue because it had a credible impact on safety. If one additional stud had loosened on the 2N SRV while it was open, the SRV outlet flange joint could have lost its structural integrity causing the open SRV to release steam to the dry well rather than directly to the suppression pool water. Containment pressure would have been adversely affected. The loosening of the SRV outlet flange studs affects the Mitigating Systems cornerstone because the function of the SRV, a component in a mitigating system, to transfer steam directly into the suppression pool would not have been performed. The issue was determined to be of very low safety significance (Green) by the Reactor Inspection Findings for At-Power Situations Significance Determination Process because there was no actual loss of safety function since the SRV outlet flange joint integrity was maintained. The inspectors determined that the use of an inappropriate gasket combined with the inadequate torque values constituted a violation of 10 CFR 50 Appendix B, Criterion III, "Design Control." This violation is being treated as a non-cited violation consistent with Section VI.A of the NRC Enforcement Policy. This violation is in Exelon's corrective action program as PEP I0012312. **(NCV 05000353/2001-004-02)**

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed post-maintenance testing and reviewed the test data for the following:

- D12 emergency diesel generator overhaul
- 2A standby liquid control (SLC) pump discharge valve replacement

The inspectors referred to testing procedures and work order documents, including:

- RT-6-092-312-1, “D12 Diesel Generator Run-in”
- ST-6-092-312-1, “D12 Slow Start Operability Test Run”
- ST-6-048-230-2, “SLC Pump, Valve, and Flow Test”

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed and/or reviewed numerous Unit 2 refueling outage activities and controls, including:

- Plant shutdown and cool down activities
- Outage risk management including changes due to emergent work or unexpected conditions
- Outage configuration controls including:
  1) availability and accuracy of reactor coolant system instrumentation
  2) electrical power alignments
  3) decay heat removal system operation
  4) availability of reactor inventory makeup water systems
  5) secondary containment controls and integrity
- Fuel handling operations including fuel movement, fuel assembly tracking, and core verification activities.
- Reactor startup, including system restoration, preparation for reactor mode changes, control rod withdrawal, reactor criticality, reactor coolant system heat up, and reactor power increases.

During the conduct of the refueling inspection activities the inspector reviewed the associated documentation to ensure that the tasks were performed safely and in accordance with plant technical specifications and operating procedures.
The procedures reviewed included the following:

- GP-2, Normal Plant Shutdown
- GP-3, Normal Plant Startup
- GP-6.1, Shutdown Operations - Refueling, Core Alteration and Core Off-Loading

- GP-6.2, Normal Plant Startup
- ON-104, Control Rod Problems
- S51.7.A, Draining Reactor Well and Dryer/Separator Storage Pool With RHR
- S53.4.A, Draining Reactor Well and Dryer/Separator Storage Pool
- Troubleshooting Form 01-0147, Hydraulic Control Unit Air Binding

Prior to the commencement of the reactor startup, the inspector also performed a walkdown of selected Unit 2 structures, systems and components (SSCs) to assess the readiness of the SSCs to support plant restart following the refueling outage.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

During the Unit 2 plant heat up following the refueling outage, the inspectors observed and reviewed the results of a high pressure coolant injection (HPCI) pump operability testing at low steam pressure. The inspectors reviewed test document ST-6-055-321-2, "HPCI Operability Verification."

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector reviewed licensee ALARA performance in accordance with 10 CFR 20.1101(b). Areas reviewed included an evaluation of ALARA planning for the highest exposure outage tasks: drywell Inspection/surveys/blocking, refueling floor work, RHR 50A valve replacement, drywell in service inspection, drywell scaffolding, drywell shielding, drywell under vessel activities, and reactor water cleanup inboard and outboard isolation valve replacements. These reviews included:

- Observations in the drywell of control rod drive replacement activities, RHR 50A valve automatic welding, drywell scaffolding changes over a 4-day period;
- Inboard main steam isolation valve replacement and main SRV replacement activities;
- Observations on the refueling floor of fuel replacement activities;
- Observations in the reactor water cleanup room of outboard isolation valve replacement activities;
- Independent radiation surveys of the drywell and reactor water cleanup rooms;

- Independent shielding evaluations of the drywell and reactor water cleanup outboard isolation valve work location;
- An interview with the drywell scheduler with respect to controlling the drywell access time period;
- An interview with the in-service inspection planner with respect to area coordination to minimize exposure;
- Interviews with scaffold scheduling and carpenter foreman to review the coordination of drywell scaffold platform installation, use and dose minimization planning;
- Observation of closed circuit television equipment use in the drywell and interviews with drywell radiological engineering staff was conducted with respect to drywell remote health physics coverage capability;
- Independent survey of the drywell control point high occupancy area;
- Conducted an interview with the Radiation Protection Manager with respect to hydrolyzing capabilities.

In addition, dosimetry records of four occupational workers that were declared pregnant women during the assessment period were reviewed, and interviews were conducted to establish the adequacy of radiological controls relative to the guidance specified in Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," and 10 CFR 20.1208, "Dose Equivalent to an Embryo/Fetus."

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA6 Meetings, Including Exit

.1 Exit Meetings

The inspectors presented the inspection results to Mr. Braun, Plant Manager, and other members of station management on May 25, 2001.

The regional inspector presented the results of a radiological protection inspection to members of Exelon management at the conclusion of the inspection on April 13, 2001.

The inspectors asked Exelon whether any materials examined during the inspections should be considered proprietary. No proprietary information was identified.

Attachment 1

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Exelon Generation Company

J. Armstrong	Director - Site Engineering
R. Braun	Plant Manager
E. Callan	Director - Maintenance
K. Gallogly	Experience Assessment Manager
C. Gerdes	Manager, Chemistry
W. Harris	Radiation Protection Manager
W. Levis	Site Vice President
J. Tucker	Senior Manager - Plant Engineering

ITEMS OPENED, CLOSED, AND DISCUSSED

Closed

URI 05000353/2001-003-02	Review of Exelon's root cause analysis on safety relief valve outlet flange stud relaxation

Opened and Closed

FIN 05000353/2001-004-01	Operator performance related to feedwater control level system manipulations
NCV 05000353/2001-004-02	Safety relief valve outlet flange stud relaxation

LIST OF DOCUMENTS REVIEWED

ALARA pre job Review No. 01-09, Installation and Removal of drywell scaffolding
ALARA pre job Review No. 01-24, In-Service Inspections including weld preps
ALARA pre job Review No. 01-22, Remove and Replace RWCU PCIVs 01 & 04
ALARA pre job Review No. 02-25, Replace HV50A Valve
Main Steam Isolation Valve Maintenance, Procedure M-041-001, Rev 16 & 17
Personnel exposure records (4) for recently monitored declared pregnant women

LIST OF ACRONYMS

ALARA	As Low As is Reasonably Achievable
FIN	finding
HPCI	high pressure coolant injection
OT	Operational Transient
PEP	performance enhancement program
RCS	reactor coolant system
RHR	residual heat removal
RWCU	reactor water clean up
SDP	significance determination process
SRV	safety relief valve
SSCs	structures, systems and components