December 11, 2000

Mr. Oliver D. Kingsley
Chief Nuclear Officer
PECO Energy Company
1400 Opus Place
Downers Grove, IL 60515-5701

SUBJECT: LIMERICK GENERATING STATION - NRC'S INSPECTION REPORT 05000352/2000-008, 05000353/2000-008

Dear Mr. Kingsley:

On November 11, 2000, the NRC completed an inspection at your Limerick 1 and 2 reactor facilities. The enclosed report presents the results of that inspection. The results of this inspection were discussed with Mr. von Suskil and other members of your staff on November 15, 2000.

This inspection was an examination of 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. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC identified one violation that was evaluated under the significance determination process and were determined to be of very low safety significance (Green). This issue have been entered into your corrective action program and is discussed in the summary of findings and in the body of the attached inspection report. This issue was determined to involve a violation of NRC requirements, but because of its very low safety significance the violation is not cited. If you contest this non-cited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with a copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Limerick Generating Station.

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/

Curtis J. Cowgill, Chief
Projects Branch 4
Division of Reactor Projects

Docket Nos.: 05000352, 05000353
License Nos: NPF-39, NPF-85

Enclosure:
Inspection Report 50-352/00-008, 50-353/00-008

Attachments: (1) Supplemental Information
(2) List of Acronyms Used

cc w/encl:
J. J. Hagan, Senior Vice President, Nuclear Operations Station Support
J. Skolds, Chief Operating Officer
G. Hunger, Chairman, Nuclear Review Board
J. A. Hutton, Director of Licensing - PECO Energy Company
J. Benjamin, Vice President of Licensing - Exelon Nuclear
J. D. von Suskil, Vice President - Limerick Generating Station
R.C. Braun, Plant Manager - Limerick Generating Station
K. Gallogly, Manager, Experience Assessment - Limerick Generating Station
Secretary, Nuclear Committee of the Board
Commonwealth of Pennsylvania

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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket Nos:	05000352, 05000353
License Nos:	NPF-39, NPF-85
Report No:	05000352/2000-008, 05000353/2000-008
Licensee:	Exelon Generation Company, LLC (EGC) Nuclear Group Headquarters Correspondence Control P. O. Box 160 Kennett Square, PA 19348
Facility:	Limerick Generating Station, Units 1 & 2
Location:	Evergreen and Sanatoga Roads Sanatoga, PA 19464
Dates:	October 1, 2000 thru November 11, 2000
Inspectors:	A. Burritt, Senior Resident Inspector D. Cullison, Resident Inspector H. Nieh, Visiting Senior Resident Inspector P. Cataldo, Visiting Resident Inspector
Approved by:	Curtis Cowgill, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000352/2000-008; 05000353-00-08; on 10/01-11/11/2000; Exelon Generation Company, LCC; Limerick Generating Station; Units 1 and 2;10CFR50.59; Resident operations report.

This report covered a six-week period of resident inspection conducted per the NRC's Reactor Oversight Process (Attachment 1). The inspection identified one Green finding which was a non-cited violation. The significance of issues is indicated by their color (Green, White, Yellow, Red) and was determined by the Significance Determination Process (SDP) in Inspection Manual Chapter 0609 (See Attachment 1). Findings for which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation.

Cornerstone: Barrier Integrity

- Green - PECO did not properly evaluate the change made to Operational Transient (OT) procedure OT-114, "Inadvertent Opening of a Relief Valve," in May 1996, in accordance with requirements of 10 CFR 50.59. Specifically, PECO did not evaluate whether the delay caused by performing actions to reconfigure electrical busses and reduce recirculation pump flow prior to placing the reactor mode switch to shutdown was consistent with the technical specifications and Updated Final Safety Analysis Report. The issue was considered to be of very low significance because: 1) there was conservatism associated in the design bases analysis and the assumptions for suppression pool heat capacity during this event; 2) the probability of a stuck open SRV with a second event that would challenge containment mitigation capability is low. Failure to perform a safety evaluation for the changes to OT-114 was a violation of 10 CFR 50.59 and is being treated as a non-cited violation (Section 1R11)

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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 brief periods for planned testing and control rod pattern adjustments.

Unit 2 began this inspection period operating at 100% power and remained at or near that power level except for brief periods for planned testing and control rod pattern adjustments.

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

1R04 Equipment Alignment - Walkdowns (71111.04Q)

a. Inspection Scope

The inspectors performed a partial walkdown of the Unit 2 “B,” “C,” and “D” residual heat removal (RHR) trains while the Unit 2 “A” RHR train was out of service for planned maintenance. The inspectors also performed a partial walkdown of the Unit 1 “A” RHR train, while the “B” RHR train was out of service for planned maintenance. These inspections verified critical portions of redundant or backup systems or trains while a system was out of service.

The inspectors performed a complete system walkdown of the Unit 1 reactor core isolation cooling system. The inspectors reviewed valve positions, electrical power availability, component labeling, and equipment deficiencies.

b. Findings

There were no findings identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

.1 Tour Plant Areas Important to Reactor Safety

The inspectors toured high risk areas at both Limerick units to assess EGC’s control of transient combustible material and ignition sources, fire detection and suppression capabilities, fire barriers, and any related compensatory measures. The fire areas included:

- Standby gas treatment area (fire area 28)
- Unit 2 4KV switch gear (fire area 17)
- Service water pipe tunnel (fire area 75)
- Unit 2 4KV switch gear (fire area 16)

b. Findings

There were no findings identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspector reviewed documents and inspected structures, systems, and components relative to the adequacy of flood protection measures for safety-related and risk significant systems and components. The document review included: (1) administrative, emergency operating, special event, surveillance, and maintenance procedures; (2) technical specifications; and (3) the final safety analysis report. The inspector also performed a walkdown of both units that focused on the risk significant emergency safeguards systems, and included relevant components such as flooding instrumentation, watertight doors, and external valve pit inspections. The inspector verified the adequacy of flood mitigation and protection equipment to ensure adequate measures existed to mitigate both the design basis external flood, as well as internal flooding events.

b. Findings

There were no findings identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspector observed licensed operator performance and the evaluator's critique during a simulator training scenario.

The inspector evaluated the adequacy of Operational Transient (OT) procedure OT-114, "Inadvertent Opening of a Relief Valve," a procedure used during the observed simulator training scenario. The inspector's evaluation included a review of the relevant documentation for a revision to OT-114 made in May 1996. Specifically, the immediate operator actions were changed to require performance of GP-4 "Rapid Plant Shutdown to Hot Shutdown" instead of an immediate reactor shutdown.

b. Findings

The inspector identified that PECO did not properly evaluate the change made to procedure OT-114 in May 1996, in accordance with requirements of 10 CFR 50.59. Specifically, PECO did not evaluate whether the delay caused by performing actions to reconfigure electrical busses and reduce recirculation pump flow prior to placing the reactor mode switch to shutdown was consistent with the technical specifications and Updated Final Safety Analysis Report.

PECO did not confirm that the change to OT-114 was consistent with Technical Specification 3.4.2 requirements. Specifically, the technical specification requires that the reactor mode switch be placed in the shutdown position if the stuck open safety relief valve (SRV) can not be shut within 2 minutes or suppression pool temperature reaches or exceeds 110 degrees Fahrenheit. In addition PECO did not confirm that the delay cause by the reconfiguring electrical busses and reducing recirculation pump flow did not exceed the suppression pool temperature assumptions used in the UFSAR analysis for a stuck open SRV.

10 CFR 50.59 allows licensee's to make changes in procedures described in the safety analysis report, without Commission approval, unless the proposed change involves a change in the technical specifications or involves an unreviewed safety question. Records of such changes must include a written safety evaluation that provides the bases for the determination that the change does not involve an unreviewed safety question. In May 1996, the operations staff made changes to OT-114, a procedure described in the updated safety analysis report without determining if the proposed change involved a change in the technical specifications or an unreviewed safety question. As a result, the record of the change did not include a written safety evaluation. Failure to perform a safety evaluation for the changes to OT-114 was a violation of 10 CFR 50.59.

In accordance the NRC's Enforcement Policy violations of 10 CFR 50.59 are dispositioned outside of the significance determination process because violations of 10 CFR 50.59 are considered to Impact the Regulatory Process. The result of this 10CFR50.59 violation, however, was assessed through the significance determination process because the change to OT-114 without appropriate evaluation created a credible impact on safety in that performance of the revised procedure could result in exceeding the suppression pool temperature assumptions for a stuck open SRV event. Nonetheless, the issue was found to be of very low significance (green) because: 1) there was conservatism associated in the design bases analysis and the assumptions for suppression pool heat capacity during this event; 2) the probability of a stuck open SRV with a second event that would challenge containment mitigation capability is low. Due to the overall very low safety significance, this violation of 10CFR50.59 was categorized at Severity Level IV and was treated as a non-cited violation (**NCV 05000352;353/2000-08-01**) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). The issue was entered into the licensee's corrective action process as Performance Enhancement Process (PEP) I0011964.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed EGC's follow-up actions with respect to the Maintenance Rule for the following equipment performance problems:

- Drywell purge exhaust bypass valve (HV 57-211) failed to close (PEP I0011501)
- High pressure coolant injection suppression pool suction valve failed to open (PEP I0011646)
- Reactor core isolation cooling room flood alarm level switch malfunction

b. Findings

There were no findings identified.

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

a. Inspection Scope

The inspectors reviewed EGC's risk management for the following emergent and planned maintenance activities:

- 2A residual heat removal system outage
- 1B residual heat removal system outage
- Unit 2 high pressure coolant injection system high temperature isolation system failure

b. Findings

There were no findings identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Unit 1 reactor water cleanup return valve HV-044-1F042 failure to operate remotely.
- Unit 2 Hydrogen recombiner level switch malfunction
- RCIC steam line isolation instrument failed channel check

b. Findings

There were no findings identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspector evaluated a deficiency with the computerized reactor power versus flow map. The map incorrectly indicated plant operation in the exclusion area during single loop operation with low flow conditions.

b. Findings

There were no findings identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post maintenance tests and reviewed the test data for the following:

- 2A residual heat removal system high point vent installation
- 2A residual heat removal system testable check valve (2032A) internals inspection
- 1B residual heat removal system motor operated valve maintenance
- Unit 2 high pressure coolant injection temperature element replacement

b. Findings

There were no findings identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed or observed the results of several scheduled equipment surveillance tests, including:

- ST-6-051-231-2, A RHR Pump, Valve, and Flow Test
- ST-2-051-107-2, Div. III RHR (LPCI) LSF/SAA - Non-outage
- ST-2-092-321-2, 4KV Emergency D21 Bus Undervoltage Channel Calibration/ Functional Test

b. Findings

There were no findings identified.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

The inspectors presented the final inspection results to Mr. von Suskil and other members of EGC management at the conclusion of the inspection on December 11, 2000.

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

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS

Licensee

M. A. Alderfer	Senior Manager, Plant Engineering
J. M. Armstrong	Director, Site Engineering
R. C. Braun	Plant Manager
K. Gallogly	Manager, Experience Assessment
G. H. Gellrick	Director, Maintenance
J. A. Tucker	Senior Manager, Operations
J. A. Wasong	Director, Training
J. von Suskil	Vice President, Limerick Generating Station

NRC

A. Burritt	Senior Resident Inspector
D. Cullison	Project Engineer
H. Nieh	Visiting Senior Resident Inspector
P. Cataldo	Visiting Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

NCV 05000352,353/2000008-01	Failure to Perform 10CFR50.59 Evaluation of a Change to Operational Transient procedure OT-114, “Inadvertent Opening of a Relief Valve.”

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
EGC	Exelon Generation Company, LLC
HPCI	high pressure core injection
KV	Kilovolt
LER	licensee event report
LPCI	low pressure coolant injection
NCV	non-cited violation
NRC	Nuclear Regulatory Commission
OT	Operational Transient
RHR	residual heat removal
RCIC	reactor core isolation cooling

ATTACHMENT 1

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margins.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margins and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margins but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.