December 7, 2001

EA-01-189
EA-01-293

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

SUBJECT: LIMERICK GENERATING STATION - NRC INSPECTION REPORT 50-352/01-011, 50-353/01-011

Dear Mr. Kingsley:

On November 10, 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 November 16, 2001, with Mr. W. Levis 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 procedures and records, observed activities, and interviewed personnel.

This report discusses a finding that appears to have low to moderate safety significance. As described in Section 1R15 of this report, this finding involves not having adequate measures in place to identify that the 2N Safety/Relief Valve (SRV) was in a degraded condition in which it was vulnerable to a failure to re-close after lifting. This issue was assessed using the Significance Determination Process (SDP) as a potentially safety significant finding that was preliminarily determined to be White, an issue with some increased importance to safety which may require additional inspection. The finding has low to moderate safety significance because the SDP identified two sequences with risk significance. These sequences are: 1) a stuck open SRV with a failure of containment heat removal and a failure to vent the containment; and 2) a stuck open SRV with a subsequent loss of high pressure injection capability and a failure to depressurize the reactor vessel such that low pressure injection sources could be used for inventory makeup.

This finding also appears to be an apparent violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Actions" and is being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy) NUREG-1600.

We believe that we have sufficient information to make our final significance determination for the degraded 2N SRV issue. However, you have the opportunity to either send us your position on the finding's significance and the basis for your position in writing or request a Regulatory Conference to discuss your evaluation and any differences with the NRC evaluation. Please contact Dr. Shanbaky, of my staff, at (610) 337-5209 within 7 days of the date of this letter to inform the NRC of your intentions. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision, and you will be advised by separate correspondence of the results. Since the NRC has not made a final determination in this matter, no Notice of Violation is being issued at this time. In addition, please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

Based on the results of this inspection, the inspectors also identified one issue of very low safety significance (Green). This issue was determined not to involve a violation of NRC requirements.

Since September 11, 2001, Limerick Generating Station has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to Exelon Generating Company. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

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

Sincerely,

/RA/

A. Randolph Blough, Director
Division of Reactor Projects

Docket Nos.: 50-352; 50-353
License Nos: NPF-39; NPF-85

Enclosure: Inspection Report 50-352/01-011, 50-353/01-011

Attachment 1: Supplemental Information

cc w/encl:
J. J. Hagan, Senior Vice President, Mid-Atlantic Regional Operating Group
W. Bohlke, Senior Vice President - Nuclear Services
J. Cotton, Senior Vice President - Operations Support
J. Skolds, Chief Operating Officer
M. Gallagher, Director - Licensing Mid-Atlantic Regional Operating Group
J. Benjamin, Vice President - Licensing and Regulatory Affairs
W. Levis, Vice President - Limerick Generating Station
R. C. Braun, Plant Manager, Limerick Generating Station
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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket Nos:	50-352; 50-353
License Nos:	NPF-39, NPF-85
Report No:	50-352/01-11, 50-353/01-11
Licensee:	Exelon Generation Company, LLC
Facility:	Limerick Generating Station, Units 1 & 2
Location:	Evergreen and Sanatoga Roads Sanatoga, PA 19464
Dates:	September 30, 2001 thru November 10, 2001
Inspectors:	A. Burritt, Senior Resident Inspector B. Welling, Resident Inspector
Approved by:	Mohamed Shanbaky, Chief Projects Branch 4 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000352-01-011, IR 05000353-01-011; on 09/30-11/10/2001; Exelon Generation Company; Limerick Generating Station, Units 1 and 2; Operability Evaluations, Permanent Plant Modifications.

This report was conducted by resident inspectors. The inspection identified one Preliminary White finding, which was an apparent violation, and one Green Finding. 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.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- **GREEN**. The inspector identified that the Unit 2 standby liquid control pump relief valve setpoints were too low such that during some failure to scram scenarios a relief valve could open and divert some standby liquid control flow from the reactor vessel.

  The finding was of very low risk significance since there was no actual loss of safety function because an operability determination supported by a detailed analysis found that the standby liquid control system would still deliver sufficient flow to meet the injection requirements and thereby mitigate all postulated events. (Section 1R17)

Cornerstone: Barrier Integrity

- **PRELIMINARY WHITE**. The inspectors identified an apparent violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Actions," because adequate measures were not in place to identify that the 2N Safety/Relief Valve (SRV) was in a degraded condition in which it was vulnerable to a failure to re-close after lifting. Engineering personnel did not adequately characterize and evaluate the uncertainties in the 2N SRV pilot valve temperature monitoring plan when they recommended that the action temperature be changed from 497°F to 475°F.

  The finding is associated with the actual failure of the 2N SRV to re-close after it lifted as operators were reducing power in preparation for an outage to repair the SRV. The SRV was also in a condition, for approximately 81 days, in which the valve was vulnerable to a failure to re-close if it lifted. The finding has low to moderate safety significance because Phase 2 of the significance determination process identified two sequences with low to moderate risk significance. These sequences are: 1) a stuck open SRV with a failure of containment heat removal and a failure to vent the containment; and 2) a stuck open SRV with a subsequent loss of high pressure injection capability and a failure to depressurize the reactor vessel such that low pressure injection sources could be used for inventory makeup. (Section 1R15)

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Report Details

Summary of Plant Status

Units 1 and 2 began this inspection period operating at 100% power and remained at or near that power level except for brief periods of planned testing and control rod pattern adjustments.

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

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors toured the enclosure for the motor-driven and diesel driven fire pumps and the emergency diesel generator rooms. The inspectors verified the adequacy of cold weather protection for key components and instrument sensing lines. The inspectors referred to the following documents:

- GP-7, Cold Weather Preparation and Operation
- Limerick Winter Readiness Updates
- Limerick Winter Readiness Tracking Sheets

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- Unit 2 “B” residual heat removal (RHR) system, while the Unit 2 “A” system was out of service for planned maintenance.

- Unit 1 “A” RHR system, while the Unit 1 “B” RHR system was out of service for planned maintenance.

The inspectors used piping and instrumentation diagram 8031-M-51, and system procedure S51.9.A, Routine Inspection of the RHR System. The inspectors also reviewed condition report 82445, “1A RHR pump motor oil sightglass markings less than adequate.” The walkdowns included reviews of valve positions, major system components, electrical power availability, and equipment deficiencies.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors toured high risk 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 (UFSAR). The fire areas included:

- 2D core spray room (fire area 59)
- Sump pump passage way (fire area 62)
- 2B and 2D core spray access corridor (fire area 63)
- Radwaste Enclosure (fire area 121)

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On November 8, 2001, the inspector observed the simulator portion of an annual examination to assess licensed operator performance and the evaluator's critique. The inspector discussed the results with operations management, and instructors. The inspector also referred to the simulator scenario document and the following off-normal and emergency operating procedures:

- ON-113, Loss of Reactor Enclosure Cooling Water
- T-101, Reactor Pressure Vessel Control
- T-102, Primary Containment Control

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed Exelon's actions with respect to the Maintenance Rule for equipment performance problems associated with the Unit 2 "A" Safeguard Fill Pump Low Flow and D24 emergency diesel generator jacket water low temperature. The inspectors reviewed associated maintenance action request documents including A1340622 and A1311695.

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 planned maintenance activities. The inspectors reviewed the Sentinel on-line risk assessment results, risk management activities, work control center planning and scheduling, and emergent work-related activities.

- 2A residual heat removal (RHR) system outage
- 1B RHR system outage
- D23 emergency diesel generator overhaul

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

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

- Unit 2 standby liquid control pump relief valve settings low

Documents reviewed included engineering change request ECR 01-00962 and an analysis of standby liquid control system injection rates.

The inspectors also reviewed Exelon's root cause analysis and other documentation related to the actuation of the 2N safety relief valve (SRV) on February 23, 2001. This inspection activity was begun in NRC Inspection Report 50-352;353/2001-003 and was tracked as an unresolved item (URI 50-353/01-03-01). The inspectors interviewed several engineering personnel and reviewed numerous documents related to the event including:

- SRV leakage monitoring plan (RT-6-041-490-2), Revisions 1 through 6
- Operations logs
- Performance Enhancement Program (PEP) I0012314
- SRV testing and trending data
- Maintenance action requests
- Event investigation reports
- LER 2-01-001

b. Findings

TBD - Preliminary White. The inspectors identified an apparent violation of 10 CFR 50 Appendix B, Criterion XVI, "Corrective Actions," because Exelon did not have adequate measures in place to identify that the 2N Safety/Relief Valve (SRV) was in a degraded condition such that it could fail to re-close if it opened. Exelon engineering did not adequately characterize and evaluate uncertainties with the 2N SRV monitoring plan.

On February 23, 2001, the 2N SRV opened and failed to re-close as operators were reducing reactor power in preparation for an outage to repair the 2N SRV. Since June 2000, Exelon's pilot valve temperature monitoring indicated that the first stage pilot valve on the 2N SRV was leaking. Excessive leakage through the first stage pilot valve had the potential to cause the SRV to unexpectedly open and also affect the ability of the SRV to re-close if it had opened. Pilot valve temperature over the eight months (June 2000 - February 2001) continued on a decreasing trend indicating increasing first stage pilot valve leakage over the period.

In April 2000, Exelon began using pilot valve temperature as a means to monitor first stage pilot valve leakage after pressure monitors on the valve failed. The pressure monitors were installed to provide indication of first stage pilot valve leakage. The use of pilot valve temperature may have added uncertainty since it relies on converting temperature to leakage rate and subsequently leakage rate to pressure. At that time Exelon established 497°F, which was 30°F below the baseline temperature of the SRV, as an action temperature limit on the pilot valve temperature for the 2N SRV. This limit was intended to represent first stage pilot valve leakage at a level which would indicate the need to repair or replace the SRV. Exelon planned to shut down the unit and repair/replace the SRV prior to the action temperature being reached,.

In August 2000, Exelon changed the action temperature to 475 °F to remove what was thought to be over-conservatism in the action limit. The pilot valve temperature trend was such that the temperature may reach the 497°F action limit prior to the scheduled outage in April 2001. On February 23, 2001, the 2N SRV pilot valve temperature was at 478°F, well above the new action temperature of 475°F, when the 2N SRV opened and did not immediately re-close with the plant at power as discussed in NRC Inspection Report 50-352;353/2001-003.

Subsequent to February 23, 2001, the inspectors noted that the documentation of the April 2000 change to the SRV monitoring plan (RT-6-041-490-2) indicated that at 492°F (35 °F below the baseline temperature) the SRV can fail to re-close. In addition, Exelon determined that they had not appropriately characterized and evaluated uncertainties in determining the action temperature for the 2N SRV.

The inspectors determined that there was an 81 day period (December 5, 2000 to February 23, 2001) in which the 2N SRV was vulnerable to a failure to re-close if it had opened. The inspectors determined, by a review of the pilot valve temperature history for the 2N SRV, that the pilot valve temperature first dropped below 492°F on December 5, 2000. The pilot valve temperature had dropped to 491°F. The inspectors used the time that the pilot valve temperature first fell below 492°F as the primary basis for determining the time period that the 2N SRV was vulnerable to fail to re-close if it had opened. The inspectors also noted that the 2N SRV lifted well before expected and, after the event, measured pilot valve leakage was higher than predicted. The higher pilot valve leakage, as measured, indicated additional vulnerability of the 2N SRV to fail to re-close if it opened.

Corrective Actions

Following the 2N SRV inadvertent opening event on February 23, 2001, Exelon performed a thorough root cause analysis. Exelon identified a number of deficiencies associated with their performance and initiated corrective action to address them. The inspectors concluded that the completed and planned corrective actions were adequate.

Cross Cutting Aspects - Human Performance

Engineering personnel did not adequately characterize and evaluate the uncertainties in the 2N SRV pilot valve temperature monitoring plan when they recommended that the action temperature be changed from 497°F to 475°F. Management did not adequately challenge the station staff regarding the uncertainties in the action temperature for the 2N SRV or the suitability of the lower action temperature for an extended period of operation. Exelon removed conservatism from the initial action temperature of 497°F without a clear understanding of the basis for the newly adopted action temperature of 475°F. The valve lifting and failure to re-close at 478°F indicated that the conservatism in the initial action temperature of 497°F may have been prudent.

Significance Determination

The lack of adequate measures in place to detect the degraded condition of the 2N SRV, such that it could fail to re-close if it opened, was of more than minor significance because it resulted in both an actual and credible impact on safety. On February 23, 2001, the 2N SRV opened and remained open with the plant at power at a pilot valve temperature that was in the acceptable band of the then existing criteria. Between December 5, 2000 and February 23, 2001, the 2N SRV was in a condition where it was more vulnerable to inadvertently open (and fail to re-close if opened), even though the pilot valve temperature was in the acceptable band of the then existing criteria. The lack of adequate measures in place to detect the degraded condition of the 2N SRV affected the Initiating Events cornerstone (because it allowed the inadvertent opening of a relief valve at power) and the Barrier Integrity cornerstone (because the open relief valve caused a breach of the reactor coolant system).

The significance determination process (SDP) as defined in the Significance Determination of Reactor Inspection Findings for At-Power Situations was applied to determine the risk associated with this finding. Following phase 1 of the SDP, since both the Initiating Event and Barrier Integrity cornerstones were affected, the guidance required that a phase 2 risk evaluation be performed using the Limerick site specific SDP worksheets.

The phase 2 SDP worksheets were used to determine the risk associated with this finding in accordance with the guidance provided in NRC Inspection Manual Chapter (IMC) 0609. Based on pilot valve temperature, the 2N SRV was in a condition for approximately 81 days where it could have failed to re-close if the valve had opened.

Therefore, the phase 2 risk evaluation used a fault exposure time of greater than 30 days. During this period, a transient, such as a turbine trip or a main steam isolation valve closure could have resulted in the 2N SRV opening and remaining open. These transients are listed in IMC 0609, Appendix A, Table 1, Row I. Based on this >30 day fault exposure time and the event frequency in Row I, an estimated likelihood rating of "A" in Table 1 was used in this risk evaluation.

IMC 0609, Table 2, indicates that all the SDP phase 2 worksheets, with the exception of the large LOCA worksheet, should be addressed for findings concerning degraded SRVs. However, since this finding did not adversely affect the ability of the SRV to open, only the Inadvertent/Stuck Open Relief Valve (IORV) worksheet was used for this assessment.

The results of this SDP phase 2 risk assessment identified two low to moderate risk significant sequences (White). These sequences are an 1) IORV followed by a failure of containment heat removal (CHR) and a failure to vent the containment (CV); and 2) IORV followed by a subsequent loss of high pressure injection (HPI) and a failure to depressurize the reactor vessel (DEP) such that low pressure injection sources could be used for inventory makeup. Recovery of the power conversion system (PCS) was credited in the second sequence, since a condensate pump could be used for vessel inventory makeup if other high pressure injection systems were unavailable. In accordance with IMC 0609 guidance, one or more white sequences in a phase 2 SDP risk assessment results in a finding with overall low to moderate risk significance (Preliminary White).

Apparent Violation

10 CFR 50 Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that measures be established to assure that conditions adverse to quality are promptly identified.

Contrary to the above, in the period from August 2000 to February 23, 2001, Exelon did not have adequate measures in place to identify the degrading condition of the 2N SRV when the action temperature was changed to 475° F. This action temperature was not adequate because it did not consider that between 492° F and 475° F, the SRV was vulnerable to a failure to re-close if it had opened. From December 5, 2000 (the first date that the pilot valve temperature dropped below 492° F) to February 23, 2001, the valve was in a condition adverse to quality. This condition increased plant risk. **(AV 50-353/01-11-01)**. **URI 50-353/01-03-01** is closed. This issue is documented in the licensee's corrective action program as PEP I0012314.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the aggregate impact of operator workarounds and equipment deficiencies on both Limerick Units. The inspectors evaluated the cumulative effects of these items on the ability of operators to respond to events in a correct and timely manner. The inspectors also reviewed these deficiencies to determine if there were any items that complicated the operators' ability to implement emergency operating procedures, but were not identified as operator workarounds.

Documents reviewed included the plant health list, operations challenges list and, control room distractions list.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

a. Inspection Scope

The inspector reviewed a recent Exelon assessment to address NRC Information Notice 2001-13 "Inadequate Standby Liquid Control (SLC) System Relief Valve Margin." The Information Notice provided recent NRC inspection insights that indicated there may be a broader issue with SLC pump relief valve margin. Licensees were expected to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. The inspector also reviewed portions of the Power Re-rate and Main Safety Relief Valve Setpoint Tolerance Relaxation permanent plant modifications to assess their impact on the standby liquid control (SLC) system.

Additional documents reviewed by the inspector included:

- Action Request A1331627, Review of NRC Information Notice 2001-13, Inadequate Standby Liquid Control System Relief Valve Margin
- ECR 01-00962, Document Information Associated with SLC Information Notice
- SLC Pump Relief Valve Test Reports
- Limerick Unit 2 Analysis of Standby Liquid Control System Injection Rates

b. Findings

Green. The inspector identified that the Unit 2 standby liquid control (SLC) pump relief valve setpoints were too low such that during some failure to scram scenarios a relief valve could open and divert some SLC flow from the reactor vessel.

The inspector identified that Exelon's August 2001 review of NRC Information Notice 2001-13, did not adequately consider for Unit 2 the cumulative effect of:

- nominal relief valve setpoints (1375 psig);
- allowable relief valve setpoint tolerance;
- potential drift of the relief valve set pressure; and
- the system pressure when all three SLC pumps automatically start.

Exelon performed an operability determination to address these concerns during a failure to scram condition. Exelon's operability determination and subsequent detailed flow balance analysis concluded that one or more SLC pump relief valves might open, but that sufficient system flow would be injected into the reactor vessel to mitigate postulated failure to scram events. Exelon's analysis determined the Unit 2 SLC pump relief valves should be set at 1440 psig.

The low SLC pump relief valve setpoint issue is more than minor in that it had a credible impact on safety since actuation of one or more SLC pump relief valves could divert SLC flow from the reactor vessel. The issue affected the Mitigating System cornerstone since the function of the SLC system, to provide a diverse reactor shutdown by injecting flow into the reactor vessel, was affected. This issue was determined to be of very low risk significance (Green) using the Significance Determination Process for Reactor Inspection Findings for At-Power Situations. There was no actual loss of safety function since an operability determination supported by a detailed analysis found that the SLC system would still deliver sufficient flow to mitigate all postulated events **(FIN 50-353/01-11-02)**. This issue is documented in the licensee's corrective action program as condition report 75653.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- D23 emergency diesel generator (EDG) cylinder liner replacement - jacket water hydrostatic pressure test
- D23 EDG heat exchanger end cover replacements - emergency service water hydrostatic pressure test
- 2A residual heat removal heat exchanger bypass valve (2F048A) motor operator spring pack verification (removal) - motor operated valve static diagnostic test

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

- Work Order R0758470, activities 1 and 58
- Work Order C0198620
- Work order R0732278
- Maintenance procedure M-C-700-256, Limitorque Motor Operator Spring Pack Testing

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- ST-6-092-314-2, D24 Emergency Diesel Generator Slow Start Operability Test
- ST-6-052-232-2, 2B Core Spray Pump Valve and Flow Test
- SP-191, 1B Residual Heat Removal Full Flow Valve Motor Operated Valve Dynamic Flow Test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the temporary changes to the Unit 1 “B” control enclosure chiller relay, Unit 2 “B” drywell chiller bearing high temperature bypass, and the north stack isokinetic flow electronics.

The inspectors verified that the temporary changes did not adversely affect system or support system availability, or adversely affect a function important to plant safety. The inspectors verified that the applicable design and licensing bases were considered and that 10 CFR 50.59 reviews were appropriate. The inspectors compared the actual modification installations of the “B” control enclosure chiller relay and 2B drywell chiller bearing high temperature bypass against the temporary modification documents to verify that the implemented changes were consistent with the approved documents.

Procedures and Documents

- Engineering Change Request (ECR) LG 01-00522
- ECR LG 01-00562
- ECR LG 00-01344

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES [OA]

4OA5 Other

.1 The inspector reviewed the Institute of Nuclear Power Operations (INPO) final report, issued October 2001, that documented the results of the 2 week INPO inspection conducted in February 2001.

.2 In NRC letter dated October 23, 2001, we issued a Severity Level III - Notice of Violation, (EA-01-189). **(VIO 50-352;353/01-11-03)** This violation is considered closed because the NRC has sufficient information on the docket concerning this issue and has documented inspection results directly related to the violation in combined inspection report 50-352/01-013 and 50-353/01-013.

4OA6 Meetings, Including Exit

.1 Exit Meeting

The inspectors presented the inspection results to Mr. Levis and other members of station management on November 16, 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**

a. Key Points of Contact

Exelon Generation Company

R. Braun	Plant Manager
E. Callan	Director - Maintenance
W. Harris	Radiation Protection Manager
W. Levis	Site Vice President
W. O'Malley	Senior Manager - Operations
J. Stone	Director - Outage Management
J. Tucker	Senior Manager - Plant Engineering

b. Items Opened, Closed, and Discussed

Opened

50-353/01-11-01	AV	Failure to have adequate measures to identify a condition adverse to quality for the 2N safety relief valve

Closed

50-353/01-03-01	URI	Review of Exelon's root cause analysis on opening of the 2N safety relief valve

Opened and Closed

50-353/01-11-02	FIN	Unit 2 standby liquid control system pump relief valve setpoints were too low
50-352;353/01-11-03	VIO	Inoperable off-site sirens not identified due to falsified maintenance and testing records and installation of jumpers that bypassed siren failure detection circuitry

c. List of Acronyms

CFR	Code of Federal Regulations
CHR	Containment Heat Removal
ECR	Engineering Change Request
EDG	emergency diesel generator
FIN	finding
HPI	High Pressure Injection
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IORV	Inadvertent/Stuck Relief Valve
LER	licensee event report

NRC	Nuclear Regulatory Commission
PEP	Performance Enhancement Program
RCS	reactor coolant system
RHR	residual heat removal
SDP	significance determination process
SLC	Standby Liquid Control
SRV	safety relief valve
TBD	to be determined
UFSAR	Updated Final Safety Analysis Report
URI	unresolved item