March 23, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNIT 1
NRC SPECIAL INSPECTION REPORT 05000373/2006009

Dear Mr. Crane:

On February 27, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a special team inspection at LaSalle County Station, Unit 1. The enclosed report documents the inspection findings, which were discussed with the Plant Manager, Mr. Daniel Enright, and other members of your staff on February 27, 2006.

The special inspection team was established by Region III on February 20, 2006, using the guidance in Management Directive 8.3, “NRC Incident Investigations Procedures.” The special inspection was chartered to evaluate the facts, circumstances, and your actions in response to the events of February 20, 2006, when a perturbation in the main turbine electro-hydraulic control (EHC) system, with the reactor at approximately 6 percent power, unexpectedly caused all five main turbine bypass valves to open. The resulting reactor water level and pressure transients with the reactor in the run mode and in the process of being shut down for a scheduled refueling outage resulted in a reactor scram. Subsequent to the reactor scram, plant operators were unable to immediately verify that all control rods had inserted into the core as designed and declared a Site Area Emergency in accordance with the station’s emergency plan.

Based on the results of this inspection, no findings of significance or violations of NRC requirements were identified. However, several issues remain outstanding regarding control rod performance and control rod position indication. First, the indicated positions for several control rods following the scram, as well as the operation of the rod worth minimizer (RWM) in scram mode, have not yet been fully resolved. Additionally, based on a review of a source range monitor (SRM) nuclear instrumentation count history, it appears that control rod 38-43 did not fully insert into the core in response to the initial scram signal for a significant period of time. Given the potential safety significance of these issues, the NRC is treating these questions as Unresolved Items, pending review and inspection of your completed root cause analyses of these matters and any other relevant documents.

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Sincerely,

/RA by S. West Acting for/

Mark A. Satorius, Director
Division of Reactor Projects, Region III

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

Enclosure: Inspection Report 05000373/2006009
w/Attachments: 1. Supplemental Information
2. Charter for Special Inspection

cc w/encl:
Site Vice President - LaSalle County Station
LaSalle County Station Plant Manager
Regulatory Assurance Manager - LaSalle County Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Senior Vice President - Mid-West Regional Operating Group
Vice President - Mid-West Operations Support
Vice President - Licensing and Regulatory Affairs
Director Licensing - Mid-West Regional Operating Group
Manager Licensing - Clinton and LaSalle
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-373
License No:	NPF-11
Report No:	05000373/2006009
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Unit 1
Location:	2601 N. 21st Road Marseilles, IL 61341
Dates:	February 20 through February 27, 2006
Inspectors:	D. Kimble, Senior Resident Inspector S. Sheldon, Regional Reactor Inspector
Approved by:	Bruce L. Burgess, Chief Branch 2 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000373/2006009; 02/20/2006 - 02/27/2006; LaSalle County Station, Unit 1; Special Inspection for Unit 1 scram with loss of normal heat sink and Site Area Emergency declaration due to multiple unknown control rod positions following the scram on February 20, 2006.

This special inspection examined the facts and circumstances surrounding a scram of LaSalle Unit 1 with complications on February 20, 2006. At approximately 6 percent reactor power and in the process of shutting down the unit to begin a scheduled refueling outage, a fault in the main turbine electro-hydraulic control (EHC) system unexpectedly caused all five main turbine bypass valves to open. The resulting reactor water level and pressure transients with the reactor in the run mode resulted in a reactor scram. Immediately following the scram, plant operators were unable to verify that all control rods had inserted into the core as designed and declared a Site Area Emergency in accordance with the station's emergency plan. No findings were identified in any cornerstones. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealing Findings

No findings of significance were identified.

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Background and Overview

On February 20, 2006, LaSalle County Station, Unit 1, experienced a transient and subsequent scram as a result of a perturbation in the main turbine electro-hydraulic control (EHC) system. For as yet unknown reasons, the EHC system reactor set pressure experienced a temporary step change that caused all five main turbine bypass valves to go fully open. The resulting reactor water level and pressure transients with the reactor in the run mode at approximately 6 percent power and in the process of being shut down for a scheduled refueling outage resulted in a reactor scram and Group 1 (Main Steam Isolation Valves) containment isolation. For several hours following the scram, plant operators were unable to verify that all control rods had inserted into the core as designed and declared a Site Area Emergency in accordance with the station's emergency plan.

As a result of the complications associated with the Unit 1 scram on February 20, 2006, the event was determined to meet the criteria within NRC Management Directive 8.3, "NRC Incident Investigation Program," for a special inspection due to the occurrence of a significant operational event that involved repetitive failures or events involving safety-related equipment or deficiencies in operations. The special inspection was conducted using NRC Inspection Procedure (IP) 93812, "Special Inspection," and IP 71153, "Event Followup." The charter for the special inspection is included as Attachment 2 to this report.

4. OTHER ACTIVITIES

4OA3 Event Follow Up (71153)

.1 Description and Chronology of the Events (93812) (Charter Items 1 and 2)

On February 20, 2006, LaSalle County Station, Unit 1, was in the process of shutting down for a planned refueling outage. At 00:01 a.m., the Unit 1 main generator output breaker was opened. The reactor was operating at approximately 6 percent power with reactor pressure controlled by cycling the turbine bypass valves. At 00:23 a.m., an apparent EHC system failure resulted in bypass valves 1 through 5 opening fully, causing a transient which resulted in high reactor water level (level 8) and low main steam line pressure. With the reactor operating in the Run mode, the low main steam line pressure condition generated an automatic signal to close all main steam isolation valves (MSIVs). An automatic scram signal was subsequently generated by the MSIVs moving shut from their fully open positions. Within seconds, licensed operators had completed all of the required actions for a scram, including arming and initiating a manual scram and placing the mode switch in shutdown.

Immediately following the scram, operators were unable to determine that all control rods had been fully inserted. With the rod worth minimizer (RWM) in scram mode, control rod 38-43 indicated position 24, while control rods 26-15 and 34-47 indicated position unknown. Control rod position displays from the plant process computer (PPC) and on the main reactor control panel similarly indicated to control room operators that the position of these three control rods could not be determined. Control room operators entered emergency operating procedure (EOP) LGA-10, “Failure to Scram,” and declared a Site Area Emergency in accordance with the station’s emergency plan.

Several minutes after the initial scram, control room operators reinitialized the “B” RWM in accordance with their training. Following the reinitialization, the RWM indicated that all control rods were fully inserted to the 00 position. Control room operators exited their EOP for the failure to scram condition and proceeded with scram recovery actions. At 00:37 a.m., operators reset the scram signal in accordance with their procedures and the RWM indicated again that the same three control rods were in unknown positions and not at position 00. Following attempts to reinitialize the RWM and regain position indication for all control rods, operators reentered LGA-10 and inserted a manual scram and an alternate rod insertion (ARI) signal at 00:47 a.m.

At 00:48 a.m., control room operators switched to and reinitialized the “A” RWM. Following reinitialization, the RWM indicated that all control rods were inserted to position 00. At 00:55 a.m., operators noted that all control rods were shown to be at position 00 on the PPC control rod display, except for control rod 26-15, which indicated “bad” on the display for an unknown position. Licensee maintenance and engineering personnel proceeded to investigate and troubleshoot the various control rod position indication anomalies observed by control room operators during the course of the event. At no time during the event did control room operators have any indication from any instrumentation other than control rod position indication that the reactor was in other than a fully shutdown condition.

At 4:27 a.m., with all control rods indicating that they were fully inserted, except for control rod 26-15, which still was shown to be in an unknown position but believed to be inserted, the licensee terminated the Site Area Emergency.

The special inspection team interviewed plant personnel, reviewed plant logs, the sequence of event recorders, plant trend traces, the licensee’s scram reports and other plant information to establish the following detailed sequence of events.

<u>February 20, 2006</u>

00:01 a.m.	Unit 1 main turbine output breaker opened for scheduled refueling outage L1R11. Turbine control valves, stop valves and intermediate valves are fully closed. Reactor pressure control is maintained with a single main turbine bypass valve open.
00:13:07	EHC electrical malfunction alarm.
00:23.03	Both “A” and “B” EHC pressure setpoints drop from approximately 938 to approximately 926 psi. Intercept valve fast closure signal (valves were already closed with main turbine off line) and Group 1 primary containment isolation system (PCIS) signal received. Main turbine load reference changed from 20 Mwe to -20 Mwe, and total main turbine control valve (CV) position indication changed from

-6 percent to approximately -30 percent. Control room operators enter procedures LGP-3-2, “Reactor Scram,” and LGA-001, “Reactor Pressure Vessel Control.”

00:23:04	All five bypass valves reposition to fully open.
00:23:09	Reactor vessel water level 8 reached, with resultant motor driven feed pump trip.
00:23:11	Main steam line pressure low and MSIV isolation signals.
00:23:13	Automatic reactor scram due to MSIVs not fully open.
00:24:09	Both “A” and “B” EHC pressure setpoints inexplicably returned to normal; intercept valve fast closure signal cleared; total CV position indication returned to -6 percent; however, main turbine load reference did not return to its previous value.
00:24:18	EHC permanent magnet generator (PMG) malfunction alarm.
00:25 a.m.	Control room operators declared shutdown safety status on Unit 1 to be RED, due to reactivity control issues associated with control rods being at unknown positions. Control room operators entered EOP LGA-10, “Failure to Scram.” Per procedure, actuation of the automatic depressurization system (ADS) was inhibited and all emergency core cooling system (ECCS) injection was manually prevented. Three control rods did not indicate that they were fully inserted: • 38-43 indicated position 24 • 26-15 indicated an unknown position (“??” on the RWM, “bad” on the PPC) • 34-47 indicated an unknown position (“??” on the RWM, “bad” on the PPC)
00:26 a.m.	Control room operators attempted to execute the steps of LGA-NB-01, “Alternate Rod Insertion,” on control rod 38-43 with no observable impact.
00:28 a.m.	Site Area Emergency declared due to multiple control rods not indicating fully inserted following automatic and manual scram signals. Operators reinitialized the “B” RWM.
00:30 a.m.	Control room operators determined that all control rods were fully inserted based on RWM indication, and exited LGA-10. Procedure LGA-001, “Reactor Pressure Vessel Control,” was entered.
00:32 a.m.	ECCS was restored to standby, and the ADS inhibit removed.

00:37 a.m.	Scram was reset in accordance with LGP-3-2.
00:41 a.m.	RWM “B” was reinitialized. Control rods 38-43, 26-15, and 34-47 indicated an unknown position.
00:45 a.m.	Control room operators reentered LGA-10, inhibited ADS, and prevented all ECCS from automatically injecting into the reactor vessel.
00:46:56	Control room operators inserted a manual scram on Unit 1, and gave the unit an ARI initiation signal.
00:48:37	RWM “A” was selected and reinitialized by control room operators. All rods indicated 00 following the reinitialization.
00:55 a.m.	Control room operators exited LGA-10, and entered LGA-001 based on all control rods other than 26-15 indicating fully inserted. The Unit 1 senior reactor operator (SRO) in command of the event directed control room operators not to reset the scram. Operators maintained the Unit 1 reactor in Mode 3 (hot shutdown) with a pressure band of 450 psig to 650 psig, and a reactor water level band of 20 inches to 50 inches.
04:27 a.m.	Site Area Emergency exited.

.2 Probable Contributing Causes of the Event or Degraded Condition (93812)

a. Equipment Failures

(1) Electro-Hydraulic Control (EHC) System Failure (Charter Item 3)

(a) Inspection Scope

When the special inspection team arrived on February 20, 2006, the licensee had initiated a troubleshooting and a root cause investigation group for the EHC malfunction. The team monitored the licensee’s troubleshooting and root cause investigations and associated activities as they progressed. The team also conducted walkdowns to observe the physical condition of the electronic equipment within the EHC system.

The team performed a detailed review of the licensee’s in-process root cause analysis, including troubleshooting activities. The team reviewed selected computer data captured during the event. Associated work orders and logs were reviewed to assess troubleshooting activities. The team also interviewed licensee personnel overseeing EHC troubleshooting efforts to understand the events and the rationale for the various forms of documentation created during and after the event. Additionally, the licensee’s corrective action database was reviewed to evaluate if the licensee had prior opportunities to identify the associated failure mechanism.

(b) Findings and Observations

Following the event, equipment associated with the EHC system was quarantined, and pictures were taken which facilitated the troubleshooting used to support the root cause investigation.

The team reviewed the troubleshooting plans and associated fault trees to verify that the scope of the equipment investigation was broad enough to capture all potential failure mechanisms which could have caused the responses observed during the event. The licensee reviewed operating experience from similar industry occurrences to identify potential failure mechanisms.

The licensee identified that the common thread impacting all of the anomalous EHC indications was the -22 Vdc power supply to the EHC electronics. An alarm light in the EHC equipment rack indicated that the -22 Vdc supply voltage had dipped by at least 10 percent. There were two redundant -22 Vdc power supplies in the EHC system. One was powered by a PMG attached to the main turbine, while the other was supplied by non-safety related 120 Vac power.

Through bench testing, the licensee identified an anomaly with the output diode on the PMG power supply. Failure of this diode seemed to provide a plausible explanation for the observed voltage drop phenomena, and the licensee performed thorough electrical checks within the EHC cabinets and found no other evidence that would implicate some other cause for the drop in the -22 Vdc power supply voltage.

The suspect diode was sent to an off site laboratory and subjected to destructive diagnostic testing. Electronics experts under contract to the licensee who examined the test data found that while the diode had broken down in the reverse direction, the magnitude of this failure was insufficient to explain all the EHC anomalies encountered from the February 20, 2006, Unit 1 scram. Licensee engineering and electrical maintenance personnel continued troubleshooting on the EHC system, but could identify no other degraded components.

At the time of this report, the licensee's formal root cause analysis had not yet been completed. The team was, thus, unable to review a final extent of condition or formal list of corrective actions for the EHC system failures. However, from discussions with licensee senior management, the team was provided with the following preliminary conclusions for this issue:

- Troubleshooting and diagnostic testing by the licensee's technical staff had determined that the two -22 Vdc EHC power supply units (the PMG power supply and the non-safety related 120 Vac power supply) had to have been the cause of all the EHC anomalies observed on February 20, 2006.
- Both power supply units had been removed and replaced with refurbished units for plant startup coming out of the present Unit 1 L1R11 refueling outage.
- The licensee was planning on hooking up monitoring equipment to the EHC system electronics to monitor power supply performance for some time following unit restart to ensure proper operation.

- The long-term resolution to this, as well as other EHC system vulnerabilities, was the licensee's planned wholesale replacement of the main turbine EHC system with a new digital EHC system during the next refueling outage on each unit.

The team found no issues or performance deficiencies with the licensee's actions. No findings of significance or violations of regulatory requirements were identified.

(2) Control Rod Scram Insertion Failure (Charter Item 4 – Partial)

(a) Inspection Scope

As part of the special inspection charter, the team was tasked with reviewing and inspecting all issues involving rod position indication anomalies associated with the February 20, 2006, Unit 1 scram. At the time of the charter's development, shortly after termination of the licensee's Site Area Emergency declaration, it was the belief of licensee management and technical personnel that all Unit 1 control rods had properly inserted into the core during the initial scram, and that the control rod position anomalies encountered were all due to instrumentation issues. Subsequently, additional analysis revealed that it was highly probable that one control rod, 38-43, did not fully insert on the initial reactor scram, and only completed its scram insertion approximately 24 minutes later after a second scram signal was manually inserted by control room operators.

For this part of the inspection, the special inspection team focused on the mechanical performance of the Unit 1 control rods. The team reviewed control rod scram time testing data required by Technical Specifications, licensee plans and data associated with fuel channel distortion/bowing, and industry guidance on fuel channel distortion monitoring in order to assess licensee compliance with established requirements. As the Unit 1 refueling outage proceeded, the team also observed selected control rod testing and interviewed station nuclear engineers and licensed operators.

(b) Findings and Observations

Following the event, the team focused on the licensee's initial Mode 4 control rod notching tests for the control rods that showed position indication anomalies during the scram. These tests provided several pieces of significant data:

- Each control rod was, in fact, fully inserted beyond position 02.
- The control rod drift alarms were functional.
- The control rods required an inordinate amount of drive water pressure, in some cases the maximum allowed by the normal operating procedure for the system, in order to be moved.
- Control rod settle time was excessively long. In some cases, no control rod settle was observed.

Each of the control cells surrounding control rods 38-43, 26-15, and 34-47 was subsequently disassembled during the licensee's Unit 1 refueling outage. Visual inspections of the fuel bundles and control blades showed evidence of contact between the control blade and fuel channel assemblies. In the case of control rod 38-43, the sections of the control blade that were adjacent to the two fuel channel assemblies that were being monitored for fuel channel distortion showed significant signs of contact.

The team next concentrated on a review of the licensee's actions in response to General Electric (GE) service information letters (SILs) and 10 CFR 21 notifications regarding fuel channel deformation/bowing. The team found that the licensee had established an appropriate program based on the vendor guidance to identify and monitor fuel assemblies that were susceptible to channel deformation. Further, the team identified that all three of the above control rods that had exhibited position indication anomalies on February 20, 2006, were included within the licensee's program for channel deformation monitoring. Each control rod had also been satisfactorily tested in accordance with the monitoring program at least once between September 2005, and January 2006.

An initial assessment by the licensee's technical staff considered it highly likely that all control rods had inserted to at least position 02 on the initial scram, and that all anomalous rod position indications were attributable to instrumentation issues. As the licensee's root cause analysis for this issue progressed, source range monitor (SRM) nuclear instrument count traces yielded questions regarding the position of control rod 38-43 immediately following the initial scram. Further investigation by the licensee's technical staff revealed that it was highly probable that control rod 38-43 had not completed its insertion travel on the initial scram, and that it was at some mid-position, perhaps out as far as position 16, when it was driven to fully insert by the second scram at approximately 00:47 a.m.

The licensee took several actions in response to the discovery that control rod 38-48 did not fully insert on the first scram. The control rod drive mechanism and control blade were both removed and replaced during the refueling outage. Albeit, neither was suspected as a cause for the control rod's failure to fully insert on the scram, the licensee had planned to conduct additional testing or inspection on each. All fuel assemblies that were candidates for fuel channel deformation monitoring, or that may have become candidates for monitoring during the ensuing Unit 1 operating cycle due to their current and predicted burn up values, have been removed from the Unit 1 reload plan and replaced with other fuel assemblies. Finally, the licensee has coordinated with their two fuel vendors, GE and Areva/Framatome, to revisit the present fuel channel deformation monitoring guidance that has been provided to the commercial nuclear industry.

At the time of this report, the licensee's formal root cause analysis had not yet been completed. The team was, therefore, unable to review a final extent of condition or formal list of corrective actions for the failure of control rod 38-43 to fully insert during the first reactor scram. Because of the safety significance of this issue, the NRC is treating the matter as an Unresolved Item, pending the review of the licensee's root cause analysis and any other relevant documentation. (URI 05000373/2006009-01)

.3 Human Factors and Procedural Issues

a. Control Rod Position Indication Issues (Charter Item 4 – Partial)

(1) Inspection Scope

At various times during the events of February 20, 2006, associated with the Unit 1 scram, control room operators were faced with the inability to rapidly and accurately determine the position of all 185 control rods. In some cases, the various instruments available for determining control rod position provided conflicting information to the operators for several control rods.

The special inspection team interviewed control room personnel and reviewed control room logs and records to determine the nature and extent of the Unit 1 control rod position indication problems on February 20, 2006. The team followed the activities of the licensee's technical staff as they performed troubleshooting and diagnostics on the various control rod position indication components.

Finally, as the licensee's troubleshooting efforts narrowed the scope of the source of the control rod position indication problems down to issues associated with the RWM, the team examined the various plausible faults postulated by the licensee's technical staff. A particular focus for the team was a modification performed on the RWM in 1995 specifically intended to correct previously observed problems with the RWM and control rod position indication supplied to control room operators in post-scram scenarios.

(2) Findings and Observations

Very early on, the licensee's technical staff had narrowed the source of the anomalies associated with the various control rod position indication problems down to issues with the RWM. By the end of the day on February 20, 2006, troubleshooting had eliminated control rod position indication system electronics, connectors or instrument cables, and possible scram valve or control rod drive (CRD) system directional control valve (DCV) leakage as plausible explanations for the control rod position indication anomalies. The special inspection team followed the licensee's troubleshooting efforts as the path led to the RWM.

On October 19, 1994, LaSalle County Station, Unit 2, scrammed from full power due to an EHC system malfunction. Nine control rods failed to indicate fully inserted for approximately two minutes following the scram. In the aftermath of the event, the licensee determined that the nine control rods had been inserted slightly beyond the full in reed switch position, and did not indicate fully inserted until each had settled back out to the full in reed switch position.

To correct the problem of control room operators not knowing that all control rods had been inserted into the core beyond position 02 following a scram, the licensee performed a modification to the RWM on each unit. As intended, the modification stopped all erroneous or "bad" data from being displayed for each control rod following a valid scram signal. In theory, following installation of this modification control room operators would have displayed on the RWM the last "good" known positions for every control rod following a scram, and could easily determine that all control rods had been fully inserted even if some had been driven to the insert overtravel position where no reed switch would be picked up. Subsequent “bad” data from a control rod being in an intermediate or indeterminate location along its path of travel would be prevented from over righting the last “good” location.

Unbeknownst to either the licensee engineering staff or operations personnel, the original computer processor source code for the RWM repopulated all 185 control rod positions with a 00 indication following any reinitialization sequence. Control room operators, trained in the use and operation of the RWM for post-scram control rod position analysis, had also been trained to manually reinitialize the RWM if they believed that the RWM computer processor had locked up. Thus, in combination with the 1995 modification, the original computer source code created a condition whereby the RWM could indicate 00 for any control rod following a reactor scram and that control rod could actually be stuck at some intermediate location with an indeterminate position indication signal.

At the time of this report, the licensee’s formal root cause analysis had not yet been completed. The team was, therefore, unable to review a final extent of condition or formal list of corrective actions for the various control rod position indication issues, including issues with the RWM. Because of the safety significance of this issue, the NRC is treating the matter as an Unresolved Item, pending the review of the licensee’s root cause analysis and any other relevant documentation.
(URI 05000373/2006009-02)

b. Site Area Emergency Declaration Issues (Charter Item 5)

(1) Inspection Scope

At 00:28 a.m. on February 20, 2006, an anticipated transient without scram (ATWS) condition was declared along with a Site Area Emergency (SAE). Although all reactor power, pressure, and water level indications were indicative of a shutdown reactor condition, control room operators were faced with anomalous control rod position indications for 3 of the 185 Unit 1 control rods, potentially indicating that they were stuck in some intermediate position and that the reactor may not remain shutdown under all design basis conditions.

The team interviewed licensee control room and emergency response organization (ERO) personnel and reviewed logs and records from the control room and technical support center (TSC) to determine the appropriateness of the licensee’s declaration of the SAE, as well as the timeliness of the declaration. In addition, the team reviewed the licensee’s emergency action level (EAL) criteria and compared it to the generic industry criteria for ATWS events in order to assess the appropriateness of the licensee’s actual EALs.

Throughout this phase of the special inspection, the team maintained close coordination with NRC Region III emergency preparedness (EP) specialists to ensure that any EP issues identified were properly evaluated for significance and generic industry impact.

(2) Findings and Observations

The team determined that current LaSalle County Station EALs were based upon the generic boiling water reactor (BWR) EALs of NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels," and were approved by the NRC staff in December 1993. A review of these generic EALs against the current licensee EALs for ATWS conditions determined that the stated industry guidance for EAL development had been followed. However, the team also found that the licensee's EALs for ATWS conditions contained no additional site specific amplifying or clarifying information either; nothing beyond that called for by the generic industry guidance was included.

The EAL utilized by control room operators to declare the SAE at LaSalle Station on February 20, 2006, was EAL No. MS3. The threshold condition stipulated under this EAL read, "Failure of **BOTH** automatic **AND** manual Scrams to establish shutdown criteria." The team noted that nowhere in the licensee's EAL MS3 basis discussion or elsewhere in the licensee's EP manual or procedures was the term "shutdown criteria" ever defined. The lack of any amplifying or clarifying guidance within the licensee's EP program, procedures, or documents left senior control room operators with no options regarding their actions in emergency plan space.

At LaSalle, whenever a unit undergoes planned significant reactor power maneuvering, a qualified nuclear engineer (QNE) is stationed in the control room as a matter of normal operating procedure. Almost immediately after the scram and following the identification by operators that control rods 38-43, 26-15, and 34-47 did not indicate fully inserted, a licensed SRO on duty in the control room approached the on duty QNE. The QNE, stationed because of the planned Unit 1 shutdown, was asked by the licensee SRO if the QNE could evaluate the reactor as being shutdown under all conditions with the three subject control rods potentially stuck fully withdrawn. Because the 38-43 and 34-47 control rods were diagonally adjacent in the core, the QNE responded that it was not possible to make such an evaluation without first performing detailed computer calculations and simulations. As a result, the training, knowledge, and experience of the licensed SROs on duty in the control room resulted in the declaration of a SAE based on the criteria listed in EAL MS3 and based on the fact that the control rod position indication instrumentation that was providing data that possibly multiple control rods had failed to fully insert in response to a valid scram signal. The team found that the control room operator decision making with respect to the SAE declaration was appropriate and consistent with the emergency plan and site procedures.

The team reviewed all licensee emergency messages that were transmitted to local, county, and state officials to ensure that regulatory requirements regarding content and timeliness were met. Additionally, the team reviewed all emergency and non-emergency official notifications made to the NRC during the event. No findings of significance or violations of regulatory requirements were identified.

At the time of this report, the licensee's formal root cause analysis had not yet been completed. The team was, therefore, unable to review a final extent of condition or formal list of corrective actions for the EP aspects of the event. However, from discussions with licensee senior management, the team was informed that the licensee would be bench marking their EALs against other BWR EALs in the near future, and that an improvement initiative to address the lessons learned from the February 20, 2006, Unit 1 scram and SAE would be forthcoming. Follow up review and assessment by the NRC of the licensee's initiatives in this area will be performed during normal baseline inspection activities.

In conclusion, the team found no issues or performance deficiencies with the licensee's EP actions. No findings of significance or violations of regulatory requirements were identified in the EP area.

4OA6 Meetings

.1 Exit Meeting

The special inspection team presented the inspection results to the Plant Manager, Mr. Daniel Enright, and other members of licensee management on February 27, 2006. The team acknowledged the receipt of certain nuclear fuel vendor proprietary documents from the licensee dealing with fuel assembly control rod channel deformation/bowing. These documents were properly denoted as proprietary in Attachment 1 to this report, and have been controlled in accordance with NRC procedures and policies governing sensitive unclassified information. The licensee was asked whether any other materials examined during the inspection should be considered proprietary. No other proprietary information was identified.

.2 Re-Exit Meeting

Subsequent to the exit meeting conducted on February 27, 2006, additional information became available concerning control rod performance during the scram. A re-exit was conducted with the site's Regulatory Assurance Manager, Mr. Terrence Simpkin, on March 7, 2006, due to this new information.

ATTACHMENTS:
1) SUPPLEMENTAL INFORMATION
2) CHARTER FOR SPECIAL INSPECTION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

S. Landahl, Site Vice President
D. Enright, Plant Manager
J. Bashor, Site Engineering Director
R. Bassett, Emergency Preparedness Manager
T. Connor, Maintenance Director
L. Coyle, Operations Director
R. Ebright, Site Training Director
F. Gogliotti, System Engineering Manager
B. Kapellas, Radiation Protection Manager
S. Marik, Shift Operations Superintendent
J. Rappeport, Nuclear Oversight Manager (Acting)
D. Rhodes, Work Management Director
T. Simpkin, Regulatory Assurance Manager
C. Wilson, Station Security Manager

Nuclear Regulatory Commission

B. Burgess, Chief, Reactor Projects Branch 2

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000373/2006009-01	URI	Failure of Unit 1 Control Rod 38-43 to Fully Insert During an Initial Scram of February 20, 2006. (Section 4OA3.2)
05000373/2006009-02	URI	Inaccurate Control Rod Position Indication Provided to Control Room Operators Following a Unit 1 Reactor Scram on February 20, 2006. (Section 4OA3.3)

Closed

None.

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

Procedures:

- LGA-01; Reactor Pressure Vessel Control; Revision 6
- LGA-10; Failure to Scram; Revision 6
- LGA-NB-01; Alternate Rod Insertion; Revision 7
- LGP-3-2; Reactor Scram; Revision 55
- LOP-RW-01; Rod Worth Minimizer Initialization and Operation; Revision 15
- LOP-RW-02; RWM Error Messages and Corrective Actions; Revision 10
- LOS-RD-SR7; Channel Interference Monitoring; Revision 4
- EP-AA-1000; Exelon Nuclear Standardized Radiological Emergency Plan; Revision 16
- EP-AA-1005; Exelon Nuclear Radiological Emergency Plan Annex for LaSalle Station; Revision 20

Work Orders/Work Requests:

- 0083857501; Control Rod Position Indication Complex Troubleshooting Plan; February 21, 2006

Operations Standing Orders:

- S06-05; Rod Worth Minimizer Operation; 2/21/2006
- S06-06; Rod Worth Minimizer Operation; 2/25/2006

Control Room Logs and Records:

- SRM Count Rate Data for SRMs A-D; 0045 on 2/20/2006 to 0050 on 2/20/2006
- LaSalle Unit 1 Control Room Operator Logs; 0000 to 2359 on 2/20/2006
- LaSalle Unit 1 Wide Range Reactor Pressure; 0000 to 0728 on 2/20/2006
- LaSalle Unit 1 SPDS Reactor Power; 0000 to 0024 on 2/20/2006
- LaSalle Unit 1 Narrow Range Reactor Water Level; 0000 to 0125 on 2/20/2006

Issue Reports:

- 459764; Rod 34-47 Difficult to Move; 2/28/2006
- 461103; Simulator RWM Scram Capture Mode Limitations; 3/2/2006
- 461346; Control Rod Drive 38-43 Needs to be Removed for Analysis; 3/3/2006
- 458939; Signs of Excessive Friction for 4 Rods After Shutdown of L1C11; 2/26/2006
- 462261; Aggregate Review IR Not Written on Control Rod 38-43 Scram Time Degradation; 3/5/2006
- 456066; NOS Identifies ATWS Mitigation Issues; 2/20/2006

- 455968; Three Rods Failed to Indicate Full In Following a Scram; 2/20/2006
- 462570; LGA-NB-01 Actions Would Have No Effect on Rod; 3/6/2006
- 465107; Historical Issue: RWM Scram Capture Modification 50.59 Error; 3/11/2006

Vendor Documents – Proprietary:

- GE SIL 320, Supplement 3; Mitigation of the Effects of Peripheral Core Location on Fuel Channel Bowing; 4/28/2003
- SC 03-08, Revision 1; GE 10 CFR 21 Communication – Interim Surveillance Program for Fuel Channel Bow Monitoring; 4/30/2003
- MFN 03-146; Letter From GE Nuclear Energy to NRC – Final Report Notification – Impact of Fuel Channel Bow on Control Rod Blade Deviations; 11/18/2003
- MFN 05-063; Letter From GE Nuclear Energy to NRC – Surveillance Program for Channel - Control Blade Interference; 7/14/2005

Other Miscellaneous Documents:

- Training Module 048, Revision 2; Rod Worth Minimizer; 2/20/2001
- LaSalle Unit 1 Cycle 11 Fuel Channel Distortion Monitoring Plan, Revision 5; 10/20/2005
- LaSalle Unit 2 Cycle 11 Fuel Channel Distortion Monitoring Plan, Revision 1; 12/2/2005
- Post Transient Review Report for the 2/20/2006 LaSalle Unit 1 Scram; 2/21/2006
- NUMARC/NESP-007; Methodology for Development of Emergency Action Levels; Revision 2

LIST OF ACRONYMS USED

ADS	Automatic Depressurization System
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CV	Control Valve
DC	Direct Current
DCV	Directional Control Valve
DRP	Division of Reactor Projects
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
EHC	Electro-Hydraulic Control
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
ERO	Emergency Response Organization
GE	General Electric
IMC	Inspection Manual Chapter
IP	Inspection Procedure
MSIV	Main Steam Isolation Valve
NRC	U.S. Nuclear Regulatory Commission
PCIS	Primary Containment Isolation System
PMG	Permanent Magnet Generator
PPC	Plant Process Computer
QNE	Qualified Nuclear Engineer
RWM	Rod Worth Minimizer
SAE	Site Area Emergency
SIL	Service Information Letter
SRM	Source Range Monitor
SRO	Senior Reactor Operator
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
Vac	Volts Alternating Current
Vdc	Volts Direct Current

February 20, 2006

MEMORANDUM TO:	Daniel Kimble, Senior Resident Inspector, LaSalle Division of Reactor Projects Stuart Sheldon, Reactor Engineer Division of Reactor Projects
FROM:	Steven West, Deputy Director Division of Reactor Projects
SUBJECT:	LASALLE UNIT 1 SPECIAL INSPECTION CHARTER FOR SITE AREA EMERGENCY DECLARATION SUBSEQUENT TO AUTOMATIC SCRAM WITH ROD POSITION INDICATION ANOMALIES ON FEBRUARY 20, 2006

On February 20, 2006, at about 0023 (CST), LaSalle Unit 1 was in the process of a plant shutdown for a scheduled refueling outage and was operating at 6 percent power when the control room received a turbine intercept valve fast closure alarm. Immediately following this alarm, all 5 turbine bypass valves fully opened, causing reactor vessel level to swell to level 8, initiating a Group 1 isolation that closed the main steam isolation valves, tripped the operating motor driven reactor feedwater pump, and initiated an automatic reactor scram.

Following the automatic scram, the plant operators observed that three control rod position indicators were not indicating appropriately, with one rod at position 24 and 2 other rods with unknown (-99) rod positions. With 3 rod indications not indicating fully inserted, a review of the emergency action levels of the licensee's emergency plan by control room operators resulted in the declaration of a site area emergency at 0028 on February 20, 2006. The licensee reinitialized the rod worth minimizer and reset the scram in an attempt to reset rod position indication and, at one point, had an all rods fully inserted indication. However, three control rod position indicators returned to a status that was unknown. Two returned to normal indication, with one rod (26-15) position indicator remaining indeterminate throughout the duration of the event.

The causes of the failure of the EHC system and the anomalous rod position indication are still under investigation. This event was determined to meet the criteria of Management Directive 8.3, "NRC Incident Investigation Program" to warrant the establishment of a special inspection team.

Based on the criteria specified in Management Directive 8.3 (Part I criterion (g)) and Inspection Procedure 71153, a special inspection was initiated in accordance with Inspection Procedure 93812 and Regional Procedure RP-8.31. The special inspection will commence on February 20, 2006. The special inspection team will consist of Daniel Kimble, Senior Resident Inspector at LaSalle, and Stuart Sheldon, DRP, Reactor Engineer.

The special inspection will evaluate the facts, circumstances, and licensee actions surrounding the February 20, 2006, event. Elements of this inspection should confirm the cause of the automatic reactor trip, the failure of the EHC turbine control system, and the failure of the rod position indication system for the rods that indicated anomalies. In addition, the team should review the licensee's declaration of a site area emergency to confirm the appropriate criteria was met. The team should also focus on assessing the adequacy of the licensee's efforts to resolve the identified equipment problems. A charter was developed and is attached. An entrance meeting will be conducted on Monday, February 20, 2006.

Attachment: As stated

cc w/att: Stuart Sheldon, DRP, Reactor Engineer
Stephen Sand, Project Manager, Project Directorate III, NRR/DLPM
C. Pederson, DRS, Division Director
J. Caldwell, Regional Administrator Region III
G. Grant, Deputy Regional Administrator Region III
Mark Satorius, Division Director, DRP
A. Boland, Deputy Division Director, DRS
Jennifer Dixon Herrity, Region III EDO Coordinator

LASALLE SPECIAL INSPECTION (SI) CHARTER

The special inspection will evaluate the facts, circumstances, and licensee actions surrounding the February 20, 2006, event. Elements of this inspection should confirm the cause of the automatic reactor trip, the failure of the EHC turbine control system, and the failure of the rod position indication system for the rods that indicated anomalies. In addition, the team should review the licensee's declaration of a site area emergency to confirm the criteria was met. The team should also focus on assessing the adequacy of the licensee's efforts to resolve the identified equipment problems. The special inspection will be conducted in accordance with Inspection Procedure 93812, "Special Inspection," and will include, but not be limited to, the following items:

1. Establish a sequence of events of the February 20, 2006, event.

2. Interview plant personnel that were involved in the event to aide in the determination of the technical aspects surrounding the reactor trip, as well as operator actions and the plant response.

3. Evaluate the licensee's root cause determination and corrective actions for the failure of the EHC control system that initiated the event.

4. Review the licensee's root cause evaluation and corrective actions for all of the anomalies associated with the rod position indication system. Focus on the sequence of events regarding which rods were indeterminate for both the initial scram and the licensee's effort to reset the rod position indication system.

5. Assess the licensee's declaration of the site area emergency, including a review of the appropriateness of the EAL criteria (NEI or NUREG 0654), the actual EAL and the timeliness of the declaration.