UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, IL 60532-4352

December 21, 2012

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2
NRC INITIAL LICENSE EXAMINATION REPORT 05000373/2012301; 05000374/2012301

Dear Mr. Pacilio:

On November 9, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your LaSalle County Station. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on October 30, 2012, with Mr. J. Bauer and other members of your staff. An exit meeting was conducted by telephone on November 9, 2012, between Mr. R. Calvin of your staff, and Mr. D. McNeil, Senior Operations Engineer, to review the proposed final grading of the written examination for the license applicants. During the telephone conversation, NRC resolutions of the station's post-examination comments, initially received by the NRC on November 9, 2012, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of October 22 and October 29, 2012. The written examination was administered by LaSalle County Station Training Department personnel on November 2, 2012. Five Senior Reactor Operator and eight Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on December 4, 2012. All applicants passed all sections of their respective examinations and were issued applicable operator licenses.

The written examination will be withheld from public disclosure for 24 months per your request. However, the examination outline with its associated examiner standards documents relating to the written examination will be available to the public.

In accordance with Title 10 of the Code of Federal Regulations, Section 2.390 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 System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ By Dell R. McNeil Acting For

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

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

Enclosures:
1. Operator Licensing Examination Report 05000373/2012301; 05000374/2012301 w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Written Examination Post-Examination Comment Resolution

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11 and NPF-18
Report No:	05000373/2012301; 05000374/2012301
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	October 22 to November 9, 2012
Examiners:	D. McNeil, Senior Operations Engineer C. Phillips, Senior Resident Inspector, Dresden C. Zoia, Operations Engineer
Approved by:	H. Peterson, Chief Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY OF FINDINGS

ER 05000373/2012301; 05000374/2012301 10/22/2012 – 11/09/2012; Exelon Generation Company, LLC, LaSalle County Station, Units 1 and 2; Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional U.S. Nuclear Regulatory Commission (NRC) examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1.

Examination Summary

All thirteen applicants passed all sections of their respective examinations. Five applicants were issued senior operator licenses and eight applicants were issued operator licenses. (Section 4OA5.1)

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners and members of the facility licensee's staff used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, Supplement 1, to develop, validate, administer, and grade the written examination and operating test. Members of the facility licensee's staff prepared the outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of October 1, 2012, with the assistance of members of the facility licensee's staff. During the on-site validation week, the examiners audited two license applications for accuracy. The NRC examiners, with the assistance of members of the facility licensee's staff, administered the operating test, consisting of job performance measures (JPMs) and dynamic simulator scenarios, during the period of October 22 through October 30, 2012. The facility licensee administered the written examination on November 2, 2012.

b. Findings

(1) Written Examination

The NRC examiners determined that the written examination, as proposed by the licensee, was within the range of acceptability expected for a proposed examination. Less than 20 percent of the proposed examination questions were determined to be unsatisfactory and required modification or replacement.

On November 9, 2012, the licensee submitted documentation noting that there were five post-examination comments for consideration by the NRC examiners when grading the written examination. The post-examination comments and the NRC resolution for the post-examination comments are included in Enclosure 3 of this report. The administered examination and answer key (ADAMS Accession Number ML12355A228) will be available electronically in 24 months in the NRC Public Document Room or from the Agencywide Documents Access and Management System (ADAMS). All changes made to the proposed written examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and are documented on Form ES-401-9, "Written Examination Review Worksheet."

The NRC examiners graded the written examination on November 14, 2012, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

(2) Operating Test

The NRC examiners determined that the operating test, as originally proposed by the licensee, was within the range of acceptability expected for a proposed examination. Changes made to the operating test, documented in a document titled, "Operating Test Comments," as well as the final as-administered dynamic simulator scenarios and JPMs

are available electronically in the NRC Public Document Room or from ADAMS. The NRC examiners completed operating test grading on November 24, 2012.

(3) Examination Results

Five applicants at the Senior Reactor Operator (SRO) level and eight applicants at the Reactor Operator (RO) level were administered written examinations and operating tests. All applicants passed all portions of their examinations and were issued their respective operating licenses.

.2 Examination Security

a. Scope

The NRC examiners reviewed and observed the licensee's implementation of examination security requirements during the examination validation and administration to assure compliance with 10 CFR 55.49, “Integrity of Examinations and Tests.” The examiners used the guidelines provided in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors,” to determine acceptability of the licensee’s examination security activities.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on October 30, 2012, to Mr. J. Bauer, Training Director, and other members of the LaSalle County Station Operations and Training Department staffs.

.2 Exit Meeting

The chief examiner conducted an exit meeting on November 9, 2012, with Mr. R. Calvin, Operations Training Manager, by telephone. The NRC’s final disposition of the LaSalle County Station's post-examination comments were disclosed and discussed with Mr. Calvin during the telephone exit meeting. The examiners asked the licensee if any of the material used to develop or administer the examination should be considered proprietary. One question on the written examination was identified as sensitive information and has been withheld from public disclosure. The licensee also identified that information used to support a second correct answer for question #35 was General Electric Proprietary Information. The proprietary information was returned to the licensee and is not publicly available.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

P. Karaba, Site Vice-President
H. Vinyard, Plant Manager
J. Washko, Operations Director
J. Williams, Shift Operations Supervisor
L. Blunk, Regulatory Assurance
J. Bauer, Training Director
T. Dean, Operations Training Manager
R. Frederes, Initial License Training Lead
D. Wright, Exam Author
R. Calvin, Operations Training Manager
S. Russell, Corporate Training

NRC

D. McNeil, Senior Operations Engineer
C. Phillips, Senior Resident Inspector, Dresden
C. Zoia, Operations Engineer
R. Ruiz, NRC Resident Inspector, LaSalle County Station

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened/Closed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
ARM	Area Radiation Monitor
CFR	Code of Federal Regulations
ER	Examination Report
JPM	Job Performance Measure
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
RBCCW	Reactor Building Closed Cooling Water
RO	Reactor Operator
RWCU	Reactor Water Cleanup
SRO	Senior Reactor Operator
TIP	Traversing In-Core Probe

SIMULATION FACILITY REPORT

Facility Licensee: LaSalle County Station

Facility Docket Nos: 50-373; 50-374

Operating Tests Administered: October 22 – 30, 2012

The following documents observations made by the NRC examination team during the initial operator license examination. These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of non-compliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility other than to provide information which may be used in future evaluations. No licensee action is required in response to these observations.

During the conduct of the simulator portion of the operating tests, the following items were observed:

ITEM	DESCRIPTION
Startup	On October 22, 2012, the simulator had difficulties starting for examination administration.

Post-Examination Comments and NRC Resolution

Question # 10:

Unit 1 is operating at 100% of rated thermal power operating on the 100% Flow Control Line.

- All nuclear instruments are operable
- The C Flow Unit fails to 0%

Based on the above conditions, which of the following correctly states the effect on the unit?

a. A Full Scram.

b. A Rod Block ONLY.

c. A Rod Block and a Half Scram.

d. A Flow Comparator alarm ONLY.

Answer: (c.)

Applicant Contention:

The applicant contends that when a Flow Unit fails to 0% a downscale/INOP condition would cause a Rod Block ONLY, which is distractor (b.).

Facility Licensee Response:

"Based on the Lesson Plan provided and LOP-AA-03 Table 2 Sheet 1, you would not receive an INOP condition without making assumptions that the Flow Unit mode switch is not in operate, module was unplugged, or the power Supply out of specified range. A downscale condition does exist, which would cause a Rod Block and a Half-Scram. The Half-Scram is caused by exceeding the Flow Biased setpoint."

Facility Licensee's Recommendation:

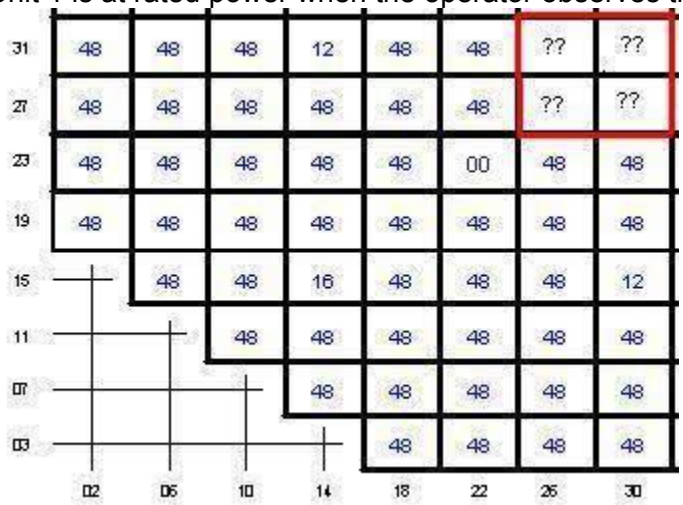
Accept (c.) as the only correct answer.

NRC Resolution:

The NRC agrees with the facility's assessment of the question. A Control Rod Block will be generated if a Comparator Trip (10% difference in output flow signals) exists. Therefore a Rod Block is in effect. However, a Half-Scram will be generated based on APRM Flow Bias Failing such that power is greater than flow. Therefore, distractor (c.) is the correct answer because both a Rod Block and a Half Scram exist. The answer key was not modified for this question. The only correct answer remains as distractor (c.)

Question # 35

Unit 1 is at rated power when the operator observes the following:



Which of the following is the cause of these indication for the four control rods in the red box?

a. Data Faults

b. Rods are at the overtravel position

c. Probe MUX Card not responding

d. No position switches are CLOSED

Answer: (c.)

Applicant Contention:

The applicant contended that when "??" was displayed it represented 4 rods that have Data Faults, which is distractor (a.) and is also correct based on new technical information.

Facility Licensee Response:

The licensee agreed with the applicant and supported two correct answers. The licensee provided new technical information in the form of a General Electric Proprietary chart showing that the "Probe MUX Card not responding" is classified as a "Data Fault" that will cause "??" to be displayed in the manner shown in the question stem.

NRC Resolution:

The NRC reviewed the proprietary information provided by the licensee and agreed with the recommendations from the applicant and the facility licensee. The proprietary information showed that some Data Faults will provide an “XX” display, and in the special case of the Probe MUX Card failure, it will provide a “??” display. The Probe MUX Card failure was shown to be a subset of the system’s “Data Faults.” Since the Probe MUX Card failure is a subset of “Data Faults,” Distractor (a) is a second correct answer. The proprietary information was returned to the licensee. The answer key was amended to accept both distractor (a.) and (c.) as correct answers.

Question # 48

Which of the following is directed to be monitored per LOA-WR-101, "Loss of Reactor Building Closed Cooling Water RBCCW" during reduced cooling capacity conditions in the RBCCW system?

a. RR Pump seal cavity outlet temperature

b. Offgas Refrigeration Machine glycol outlet temperature

c. IN Compressor intercooler discharge temperature

d. RWCU Pump Motor Cooler outlet temperature

Answer: (a.)

Applicant Contention:

The comment made was that a loss of WR also has an impact on RWCU (RT) Pump motor and should be monitored making answer (d.) correct.

Facility Licensee Response:

RWCU (RT) pumps are cooled by WR, but are not required to be monitored in LOA-WR-101, Steps C.2.2, of the discussion section, and B.1.5 of the Reduced Cooling Capacity section, both specify monitoring the RR Pump Seal temperatures. The Licensee recommends retaining only distractor (a.) as a correct answer.

NRC Resolution:

The NRC agrees with the licensee that only distractor (a.) is a correct answer because the procedure is specified. All of the distractors are components that should be monitored during a reduced cooling capacity of RBCCW, but only the RR Pump seal cavity outlet temperature is specified to be monitored in the procedure, making distractor (a.) the only correct answer. Using the applicant's argument, all four of the distractors would be correct answers and the question would be deleted because of the multiple correct answers. However, upon closer review of the procedure, it was determined that the answer is found in a subsequent step of the procedure. Subsequent steps in procedures are not required to be memorized, but an applicant is required to have an awareness of what steps are provided in the procedure, but not necessarily specific steps in the procedure. Because the answer comes from a subsequent action of a procedure and would not necessarily be memorized, the NRC has deleted this question from the examination. The examination key was modified to delete this question from the examination.

Question # 60

With Unit 2 at rated power, a large LOCA occurred in which fuel became temporarily uncovered. LGA-001 and LGA-003 have been entered. 20 Minutes after operators started the Post-LOCA H_2/O_2 monitoring system, the following readings are taken:

- Drywell O_2 concentration is 0.5% by volume and stable.
- Drywell H_2 concentration is 1% by volume and rising slowly.

1) Would these Post-LOCA H_2/O_2 monitoring system readings be RELIABLE or would they still NEED MORE TIME to warm up and stabilize?

2) Based solely on the H_2/O_2 content, for these post-LOCA conditions, would operation of the H_2 Recombiners per LGA-HG-101 be DESIRABLE or NOT DESIRABLE?

a. 1) RELIABLE
   2) NOT DESIRABLE

b. 1) RELIABLE
   2) DESIRABLE

c. 1) NEED MORE TIME
   2) DESIRABLE

d. 1) NEED MORE TIME
   2) NOT DESIRABLE

Answer: (b.)

Applicant Contention:
The applicant contended that the plant is below 2% Hydrogen indication, therefore, the Hydrogen leg of LGA-003 would not be entered making answer (a.) correct as well.

Facility Licensee Response:

LGA-003 lesson plan states if LGA-003 is entered then parallel execution is also required because the symptomatic approach to emergency response precludes the prioritization of any one action path since independence for initiating events and transients must be maintained. Therefore, the Hydrogen leg of LGA-003 is entered because the stem of the question states Hydrogen is 1% and rising slowly, which leads to entering LGA-011 and starting the Hydrogen Recombiner. The licensee recommends that only distractor (b.) be retained as the correct answer to this question.

NRC Resolution:

The NRC agreed with the facility licensee's response. When an LGA is entered, all legs of the LGA are entered and executed simultaneously. Since the stem of the question stated that LGA-003 has been entered, all legs of the LGA would be executed in this postulated event. Since the legs are all being executed, the applicant's contention is incorrect. With hydrogen increasing above 1%, LGA-011 would be entered and the hydrogen recombiner would be started. Because the instrumentation would be reliable at this point, and starting the hydrogen recombiners is desirable, distractor (b.) was retained as the only correct answer to this question. The answer key was not modified, distractor (b.) was retained as the only correct answer.

Question # 74

Unit 1 is at rated conditions with TIP traces in progress.

- TIP area is posted and verified clear.
- 'B' TIP has just completed a trace and has been returned to the shield.
- 'A' TIP is at position 0001.

The 1H13-P601-B211, RB TIP ROOM RAD HI/DOWNSCALE, has come in and is verified to be HI. Which of the following is/are the NEXT expected action(s) for the operator running the TIP trace?

a. close the 'A' TIP ball valve ONLY

b. continue with the next TIP trace

c. close the 'A' TIP ball valve and shutdown the TIP machine

d. withdraw the 'A' TIP to the in-shield position and then close the 'A' TIP ball valve

Answer: (b.)

Applicant Contention:

The applicant contends that distractor (d.) is also correct, based on when the RB TIP ROOM RAD HI/DOWNSCALE alarm comes in. It is possible the alarm was caused by the second TIP run and the TIP run should be placed in a safe condition until the cause of the alarm is verified. This would make distractor (d.) a correct answer.

Facility Licensee Response:

The stem of this question indicated that the “B” TIP has just been completed and has been returned to its shield. The “A” TIP has just been positioned at 0001. At this point, the question stated the TIP Room Area Radiation Monitor (ARM) alarms and it is verified to be “HI.”

To answer the question based only on the data provided without assumptions, the following logical approach is expected: The first operator response to an alarm is to follow the panel alarm procedure, in this case LOR 1H13-P601 B211. This procedure has the operator read the corresponding ARM to determine actual radiation level and also refer to the ARM abnormal response procedure, LOA-AR-101. The magnitude of the dose rate level is not provided and it is not stated whether this is due to known reasons, thus the steps of LOR 1H13-P601 B211 and LOA –AR-101 properly direct actions. These procedural steps direct appropriate actions to place the equipment in a safe condition. To place a TIP in a safe condition is to withdraw it into its shield and close the TIP ball valve.

While execution of a TIP trace set typically does lead to receipt of alarm 1H13-P601 B211, the question does not contain enough details that assure that this is an expected alarm condition. Proper panel alarm and abnormal response procedure actions lead to placing the equipment in a safe condition. This response leads to answer (d.) also being a correct answer. The licensee recommends accepting both (b.) and (d.) as correct answers.

NRC Resolution:

The NRC does not agree with the facility licensee nor with the applicant that distractor (d.) should be included as a correct answer. In the facility's proposed written examination, the following statement was made concerning the validity of the distractors and conflicts with their post-examination comment: "The TIP room HIGH Rad alarm is a **normal occurrence** while operating TIPs. So therefore the operator should continue performing TIP traces." A review of the reference procedures for the question (1H13-P601 B211 and LOA-AR-101) does not show the proposed correct answer {distractor (b.)} as being a correct answer. Furthermore, it does not show distractor (d.) as a correct answer. The procedure requires one of two assumptions in order to answer the question. As the facility licensee discussed, the question stem did not provide information concerning the cause of the high radiation alarm, only that it is verified to be high. Since the cause of the high radiation alarm was not provided, the applicant must make a decision/assumption concerning what action needs to be taken. The next step in the annunciator procedure states that the actual correct answer would be to determine the cause of the alarm. After determining the cause of the alarm, procedure 1H13-P601 B211 allows continued TIP operation only after the room dose rates decay below the alarm setpoint if the alarm is due to TIP operation. If the alarm cause is unknown, the procedure directs RP to check the validity of the alarm and determine the source. It also directs checking various fuses and power supplies. Finally, it directs operator to refer to LOA-AR-101. Procedure LOA-AR-101 directs evacuation of the space, surveys and samples, and to determine the cause of the high radiation. The procedure does not direct placing a TIP in a safe condition, nor does it direct the closure of the TIP ball valve. Neither procedure directs continuation of TIP traces {correct answer (b.)}. The question appears to be a "skill of the craft" response where the operators know what to do if the alarm occurs. Because neither procedure provided a correct answer to this question, the answer key was modified to delete this question from the written examination.

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Sincerely,

/RA/ By Dell R. McNeil Acting For/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

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