January 29, 2007

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, UNITS 1 AND 2
NRC INITIAL LICENSE EXAMINATION REPORT 05000373/2006301(DRS); 05000374/2006301(DRS)

Dear Mr. Crane:

On November 22, 2006, Nuclear Regulatory Commission (NRC) examiners completed initial operator licensing examinations at your LaSalle County Station. The enclosed report documents the results of the examination which were discussed on November 22, 2006, with S. Landahl and other members of your staff. An exit meeting was conducted by telephone on December 18, 2006, between Mr. R. Ebright, Sr. of your staff and Mr. D. McNeil, Senior Operations Engineer, to review the resolution of the station's post examination comments and the proposed final grading of the written examination for the license applicants.

The NRC examiners administered an initial license examination operating test during the week of November 14, and on November 21, 2006. The written examination was administered by NRC examiners and LaSalle County Station training department personnel on November 22, 2006. Eight Senior Reactor Operator applicants were administered license examinations. Two of the applicants were previously licensed Nuclear Station Operators at LaSalle County Station. The results of the examinations were finalized on January 9, 2007. One applicant failed the written examination and was issued a proposed license denial letter. Seven applicants passed all sections of their respective examinations and six were issued senior operator licenses. In accordance with NRC policy, the license for the seventh applicant (who passed the written examination with a score less than 82 percent) was withheld pending the outcome of any written examination appeal that may be initiated.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room, or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at **http://www.nrc.gov/reading-rm/adams.html** (the Public Electronic Reading Room).

We will gladly discuss any questions you have concerning this examination.

Sincerely,

/RA by Bruce Palagi 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 05000373/2006301(DRS); 05000374/2006301(DRS) w/Attachment: Supplemental Information
2. Simulation Facility Report
3. Post Examination Comments w/ NRC Resolution
4. Written Examinations and Answer Keys (SRO)

cc w/encl 1 & 2:
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

cc w/encl 1, 2, 3 & 4: R. Ebright, Sr., Training Director, LaSalle County Station

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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

cc w/encl 1, 2, 3 & 4: R. Ebright, Sr., Training Director, LaSalle County Station

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REGION III

Docket Nos:	50-373; 50-374
License Nos:	NPF-11; NPF-18
Report No:	05000373/2006301(DRS); 05000374/2006301(DRS)
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	November 14 through November 22, 2006
Examiners:	D. McNeil, Senior Operations Engineer M. Bielby, Senior Operations Engineer D. Reeser, Operations Engineer
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000373/2006301(DRS), 05000374/2006301(DRS); 11/14/2006 - 11/22/2006;
Exelon Generation Company, LLC, LaSalle County Station. Initial License Examination Report.

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

Examination Summary:

- One applicant failed the written examination and was issued a proposed license denial. Seven of eight applicants passed all sections of their respective examinations. Six applicants were issued senior operator licenses. The seventh license may be issued pending the outcome of any written examination appeal. (Section 4OA5.1).

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The Nuclear Regulatory Commission's examiners prepared the examination outline and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of October 23, 2006 at the LaSalle County Station Training Building with the assistance of members of the licensee training staff. The NRC examiners conducted the operating portion of the initial license examination during the week of November 14 and on November 21, 2006. The NRC examiners and members of the LaSalle County Station training department staff administered the written examination on November 22, 2006. The NRC examiners used the guidance established in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare, validate, revise, administer, and grade the examination.

b. Findings

Written Examination

During the validation of the written examination several questions were modified or replaced. Changes made to the written examination were documented on Form ES-401-9, "Written Examination Review Worksheet" which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The licensee submitted seven written examination 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 contained in Enclosure 3, "Post Examination Comments and Resolutions." The NRC examiners graded the written examination on January 9, 2007, and conducted a review of each missed question to determine the accuracy and validity of the examination questions.

Operating Test

During the validation of the operating test, several Job Performance Measures (JPMs) were replaced and some modifications made to the dynamic simulator scenarios. The JPMs were replaced for several reasons: 1) the JPM was originally thought to be performed in the control room, when it was actually performed in the auxiliary electric room; 2) the JPM was determined to be too simplistic in nature (inadequate difficulty level); 3) the JPM did not conform to the licensee's ALARA program, and; 4) the JPM was time consuming (>1hour). Changes made to the operating test were documented in a document titled, “Operating Test Comments,” which is available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). The NRC examiners completed operating test grading on January 9, 2007.

Examination Results

Eight applicants were administered written and operating tests at the Senior Reactor Operator level. Two of the applicants were previously licensed Nuclear Station Operators at LaSalle County Station. One applicant failed the written examination and was issued a proposed license denial. Six applicants passed all portions of their examinations and were issued operating licenses. One applicant passed all portions of the license examination, but received a written test grade below 82 percent. In accordance with NRC policy, the applicant’s license will be withheld until any written examination appeal possibilities by other applicants have been resolved. If the applicant’s grade is still equal to or greater than 80 percent after any appeal resolution, the applicant will be issued an operating license. If the applicant’s grade has declined below 80 percent, the applicant will be issued a proposed license denial letter and offered the opportunity to appeal any questions the applicant feels were graded unfairly.

.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

During validation of the examination, the NRC Chief Examiner could not account for all copies of one JPM. It was believed that one copy of the JPM was left in the NRC Region III office, but as a precautionary measure to ensure examination integrity, the JPM was significantly modified. Because it could not be shown that examination material was actually lost and no applicants gained an unfair advantage as a result of the significant revision to the JPM, no violation of 10 CFR 55.49 occurred.

.3 (Closed) URI 05000373/2005005-01; 05000374/2005005-01, Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency

During a Licensed Operator Requalification Program inspection documented in IR 05000373/2005005; 05000374/2005005, NRC inspectors were unable to determine if granting concurrent watchstanding credit for seven control room operators to maintain an active operator license was an acceptable practice when only five watchstanders are required to be on shift by 10 CFR 50.54(m) and by station procedure (OP-LA-101-111-1001, "On-Shift Staffing Requirements"). Unresolved Issue 05000373/2005005-01; 05000374/2005005-01, "Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency" (Section 1R11.7) was opened to document this issue. After discussions with NRC headquarters personnel it was determined that credit for up to seven watchstanders at a two-unit single control room station (one Shift Manager, two Unit Supervisors, and four Nuclear Station Operators) would be allowed if all seven watchstanders were procedurally required to be on shift, and immediate (within 2 hours) action taken to restore minimum watchstander numbers if less than seven watchstanders are available during the shift. To ensure none of the operator's licenses had inadvertently become inactive because of credit being applied for two additional watchstations not required by procedure, additional inspection was completed by the NRC inspectors that verified that if only five watchstations (one Shift Manager, one Unit Supervisor, and three Nuclear Station Operators) were given concurrent credit for watchstanding proficiency, enough watches were stood by the station's licensed operators to complete the minimum number of requisite watches to keep their licenses in an active status and no violation of 10 CFR 55.53(f) occurred. This completes the Region III inspection requirements for this URI. This item is closed.

4OA6 Meetings

Debrief

The chief examiner presented the examination team's preliminary observations and findings on November 22, 2006, to Ms. S. Landahl and other members of the LaSalle County Station Operations and Training Department staff.

Exit Meeting

The chief examiner conducted an exit meeting on December 18, 2006, with Mr. R. Ebright, LaSalle County Station Training Director by telephone. The NRC's final disposition of the station's post-examination comments were disclosed and revised preliminary written examination results were provided to Mr. Ebright during the telephone discussion. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. No proprietary or sensitive information was identified during the examination or debrief/exit meetings.

Interim Exit

On January 17, 2007, the chief examiner conducted an interim exit meeting by telephone with Mr. L. Blunk, Operations Training Manager in which resolution of URI 05000373/2005005-01; 05000374/2005005-01, "Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency (Section 1R11.7)" was discussed. This item was considered closed.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
S. Landahl, Site Vice President
D. Rhoades, Operations Director
R. Ebright, Training Director
T. Simpkin, RA Manager
J. Rappeport, NO Manager
S. DuPont, RA
D. Puckett, Ops Training ILT Lead
L. Blunk, Operations Training Manager
P. Leheney, Operations Trainer

NRC
D. Kimble, LaSalle SRI

IEMA/DNS
J. Yesinowski, RI

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

05000373/2005005-01; 05000374/2005005-01	URI	Credit for More Operators than Described by the Minimum Staffing Specified in 10 CFR 50.54(m) for Watch Standing Proficiency (Section 1R11.7)

LIST OF DOCUMENTS REVIEWED

LAP 0300-03, Operations Shift Staffing, Revision 38

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
DRS	Division of Reactor Safety
NRC	Nuclear Regulatory Commission
ALARA	As Low As Reasonably Achievable
IR	Inspection Report

SIMULATION FACILITY REPORT

Facility Licensee: LaSalle County Station

Facility Docket No: 50-373; 50-374

Operating Tests Administered: November 14-21, 2006

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
None	

<u>**Post Examination Comments and Resolutions**</u>

Examination Question #1

Given the following plant conditions:

- 1B Reactor Recirculation pump tripped
- Power and flow are in the allowable region of Technical Specifications

Thermal Limits:

(1) Average Planer Linear Heat Generations Rate (APLHGR)
(2) Minimum Critical Power Ratio (MCPR)
(3) Linear Heat Generation Rate (LHGR)
(4) Maximum Allowable Power Ratio

Which combination of thermal limits must be adjusted?

a. 1 and 2 only

b. 1, 2, and 3 only

c. 1, 2, and 4 only

d. 3 and 4 only.

Correct Answer: b.

Applicant's Comment:

For question #1, APLHGR and MCPR are the only Thermal limits that are required to be adjusted per the T.S. Bases for 3.2.1, 3.2.2, and 3.2.3.

Per 3.2.1, for SLO (Single Loop Operations), a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operations. (Page B 3.2.1-1). MCPR obviously does get adjusted for SLO based on Safety limit MCPR.

Bases 3.2.3 does NOT state that a SLO multiplier is applied to LHGR for Single Loop Operations. Based on the statement in the Background section of B.3.2.3, Limits on the LHGR are specified to ensure that the fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences, and Per LaSalle UFSAR section 15.3, Decrease in Reactor Coolant System Flow Rate, section 15.3.1 describes a single Recirculation Pump Trip. Hence a single Recirculation Pump Trip is an Anticipated Operational Occurrence and no LHGR multiplier is required.

Therefore, the only correct answer is MCPR and APLHGR. Items #1 and 2 (Answer "A").

References:

TS 3.2.1, 3.2.2, and 3.2.3 bases;
LOA-RR-101 REACTOR RECIRCULATION SYSTEM ABNORMAL; and
Core Operating Limits Report (COLR).

LaSalle Management Response:

A review of the Tech Spec bases for the individual thermal limits indicates that in all cases, APLHGR and MCPR must be adjusted for single loop operations. Per the Tech Spec bases, LHGR should be valid for all anticipated operational occurrences. LOA-RR-101 REACTOR RECIRCULATION SYSTEM ABNORMAL says that LHGR has to be adjusted only if required by the COLR. The students did not have the COLR available for the written examination and should not be required to have the COLR memorized along with all the particulars of the current core load. A detailed review of the COLR indicates that LHGR is only required to be adjusted for single loop operation if GE-14 fuel is installed. A LaSalle Station Qualified Nuclear Engineer has reviewed this question, reviewed the COLR and agrees with this position. Depending on whether GE-14 fuel is installed or not, either answer “A” or “B” could be correct. The question does not indicate if GE-14 fuel is installed. Not enough information is available to differentiate between answer “A” or “B”.

Recommended Disposition:

Recommend accepting both “A” and “B” as correct.

NRC Response:

The applicant argues that since the Technical Specifications (TS) Bases for LHGR do not mention adjustment for single loop operation (SLO) that no adjustment is necessary. The TS Bases is a supporting document and is not a comprehensive reference. In support of the correct answer (distractor b.), Technical Specification LCO 3.4.1 states:

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation within Region III of Figure 3.4.1-1.

or

One recirculation loop shall be in operation within Region III of Figure 3.4.1-1 with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," SLO limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," SLO limits specified in the COLR; and

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," SLO limits specified in the COLR.

When one recirculation loop is out-of-service, per TS 3.4.1, SLO limits for LHGR must be applied when power is greater than or equal to 25 percent Rated Thermal Power (RTP). The applicant also states that, per the TS Bases, limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including Anticipated Operational Occurrences (AOO). The applicant (and Licensee management) argues that since a trip of a reactor recirculation pump is an AOO, no adjustment is necessary. The limits (and subsequent adjustments) are applied so fuel design limits are not exceeded during steady state operations as well as during AOOs. The LHGR limits are power and flow dependant. The adjustments required by TS LCO 3.4.1 are applied post-transient; therefore, the applicant's argument is invalid. Licensee management points out that the COLR was not available. Applicants are instructed prior to the exam, that if they have any questions or concerns about a question that they are to ask the exam administrator for clarification. There was no record that any applicant asked to see the COLR during the examination. Licensee management also states that SLO limits, for LHGR, only apply to GE-14 fuel and therefore the answer is dependant on whether or not GE-14 fuel is installed. In a separate, post-examination conversation between NRC examiners and the station's training department, it was revealed that the current core contains GE-14 fuel. Applicants are instructed to answer questions based on actual plant operation, procedures, and references. Since the current core contains GE-14 fuel, SLO limits for LHGR would be applicable. Because the applicant should have known GE-14 fuel was in the core and LHGR apply to that fuel, answer choice 'b' was retained as the only correct answer.

Examination Question #15

Unit 1 had been operating at 100 percent RTP for 237 days when a Group 1 Isolation occurred. After responding to the event, operators found the following plant conditions:

- RPV pressure is being maintained 800 to 1000 psig using RCIC and SRVs
- RPV water level is being maintained between 11 in. and 59.5 in. with RCIC
- RCIC is taking a suction from the Suppression Pool
- Both loops of RHR are in Suppression Pool Cooling
- Suppression Pool Temperature is 110°F and increasing slowly
- Suppression Pool Level is –5 ft and decreasing slowly

Which of the following will occur first as Suppression Pool Level drops?

a. Damage to the RHR pumps due to inadequate Net Positive Suction Head

b. Damage to the RCIC pump due to air entrainment in the pump suction

c. Damage to the RCIC pump due to inadequate Net Positive Suction Head

d. Pressurization of the suppression chamber due to inadequate condensation of steam by the Suppression Pool

Correct Answer: c.

Applicant's Comment:

This question does not provide rates for Suppression Level decreasing or Suppression Pool Temperature increasing. Answer "C" would only be correct if temperature reached 195°F (with Pool Level at <-7ft). With 2 loops of suppression pool cooling in operation, suppression pool temperature would not be expected to exceed 197°F with pressure being maintained at 800-1000 psig with RI and SRVs. Choice "B" would occur at - 11.4 ft, regardless of suppression pool temperature.

References:

LGA-001 RPV Control; and
LGA-003 Primary Containment Control.

LaSalle Management response:

Answer "C" is correct per the LGA-003 step that states that below -10.4 feet RCIC may not have adequate NPSH. Below -10 feet, the RCIC NPSH curve no longer applies.

Recommended Disposition:

Recommend no change to answer key for question #15.

NRC Response:

As stated by Licensee management, per the low suppression pool level step of LGA-003 and Figure NC (RCIC NPSH LIMIT), adequate NPSH cannot be assured below -10.4 feet. Answer choice 'c' was retained as the only correct answer.

Examination Question #40:

Given the following:

- RPV Blowdown has been initiated due to exceeding the Pressure Suppression Pressure curve; and

- Drywell Pneumatic Bottle Bank header pressures both read approximately 130 psig.

Of the answers given below, which is the highest drywell pressure that will allow the ADS valves to open and remain open?

a. 40

b. 50

c. 60

d. 70

Applicant's Comment:

The question indicates that the RPV has been blowndown due to exceeding the PSP curve. In addition, both ADS bottle banks are at 130 psig. The question does not state what pressure exists in the IN receiver or ADS accumulators. The question implies that because the bottle banks are at 130 psig, the IN system and ADS accumulator pressure has decreased to 130 psig. Although this is a plausible scenario making choice "A" correct, it is also plausible that the IN system and ADS accumulators are still pressurized and that the ADS bottle banks are low. It must be noted that there is a pressure regulator between the bottle banks and the unregulated IN header that opens at 160 psig. Design assumptions (LPGP-CALC-01) states that a minimum differential pressure of 88 PSIG is required to open the ADS valves. It is assumed that the IN system is pressurized to 151 psig (low pressure alarm for the ADS accumulators per LPGP-CALC-01). If the system is pressurized to at least 151 PSIG coupled with 88 psid needed to open the ADS valves, the ADS valves would function up to 63 PSIG in the containment making answer "C" also correct.

References:

LPGP-CALC-01 EOP & SAMG CALCULATION CONTROL -- INSTRUCTIONS AND INPUT DATA

LaSalle Management response:

The student's technical position is correct. Based on conditions listed in the question, the student has to determine the effect on pressure in the ADS accumulators. The question indicates that pressure in the ADS bottle banks are both at 130 psig. For pressure to be that low during the transient, there would have to be a leak in the instrument nitrogen system. Each ADS valve has an ADS accumulator separated from the header by a check valve. If pressure is lost in the ADS bottle banks, pressure can still remain in the ADS accumulators. The bases for the primary containment pressure limit assumes ADS accumulator pressure is at least as high as the ADS accumulator low pressure alarm setpoint. The question does not give information regarding the status of the ADS accumulators. There is not enough information to differentiate between answers "A" and "C".

Recommended Disposition:

Recommend accepting both “A” and “C” as correct.

NRC Response:

The question implies that a high drywell pressure, and therefore a high suppression chamber pressure existed that has required RPV blowdown. Under these conditions the IN compressors are not available (i.e., tripped on low suction pressure due to containment isolation) and once the IN receiver volume is depleted the pneumatic supply is from the bottle banks regulated down to nominal header supply pressure of 160 psig.

The question states that the Bottle Bank <u>header</u> (underline added) (i.e., the pneumatic pressure supplied to the ADS valves) is reading approximately 130 psig. It doesn’t really matter how the pressure got to that value, whether from a leak, bottle depletion due to usage over a long time period, or improper setting of the pressure regulators, this was the given condition. The applicant acknowledged that the question implied that the pneumatic system up through the ADS accumulators was equalized at 130 psig, then postulates a scenario not supported by the question stem. The applicants are instructed, prior to starting the exam, to make no assumptions not supported by the stem of the question and could occur as a consequence of the conditions stated in the question. Because the applicant’s position is not supported by the question stem, answer choice ‘a’ was retained as the only correct answer.

Examination Question #76

The Main Turbine is coasting down following a trip from full power. As speed reduces below 1400 rpm, the unit assist Reactor Operator reports that turbine vibration has increased above 10 mils. You should direct the unit assist Reactor Operator to . . .

a. continue to monitor vibration, as high vibrations are normal as the turbine coasts down and vibration levels should decrease as speed approaches 800 rpm.

b. lower Main Turbine Lube Oil temperature, to increase the oil viscosity, which will slow the turbine down faster.

c. open the condenser vacuum breaker to slow the turbine down to zero speed as quickly as possible to prevent further damage.

d. throttle open the condenser vacuum breaker, to reduce turbine speed more quickly, and when vibrations are less than 10 mils to close the condenser vacuum breaker.

Correct Answer: d.

Applicant's Comment:

I did not select Answer "D" due to the fact that if vibrations do not drop below 10 mils, per Answer D, you would never close the vacuum breaker. LGP-2-1 page 22 says to close vacuum breaker when past critical speeds. In addition, a note also on page 22 of LGP-2-1 states do not let backpressure go greater than 5 inches.

References:

LGP-2-1 NORMAL UNIT SHUTDOWN; AND
LOA-TG-101 UNIT 1 TURBINE GENERATOR Abnormal

LaSalle Management response:

Answer "A" is incorrect based on the requirement to throttle open the condenser vacuum breaker if the turbine exhibits excessive vibrations.

Answer "B" is incorrect based on no direction to change turbine lube oil to reduce turbine speed

Answer "C" is incorrect based on LGP-2-1 step E.3.17.1 which states to close the vacuum breaker when turbine speed is below the critical speed range.

Answer "D" is also incorrect. Although it is true that LGP-2-1 step E.3.17.1 states that if the main turbine exhibits excessive vibration, the vacuum breaker should be throttled open to reduce turbine speed, there is no direction to close the vacuum breaker when vibrations are less than 10 mils. The procedure directs closing the vacuum breaker when turbine speed is below the critical speed range. LOA-TG-101 step C.3 states that the turbine critical speed range is between 800 and 1400 rpm. LOA-TG-101 step B.1.20 states that the vacuum breaker should be throttled open if vibration exceeds 10 mils. LOA-TG-101 also states to close the vacuum breaker when turbine rpm is lowered below the critical speed band. None of the procedures directed the vacuum breaker to be closed when vibrations drop below 10 mils.

Recommended Disposition:

Because there is no correct answer, recommend deleting this question from the exam.

NRC Response:

Turbine-generators typically experience higher than normal vibrations when slowing down through the critical speed range (800-1400 rpm as stated in step C.3 of LOA-TG-101/201). If vibrations are excessive (i.e., greater than 10 mils per LOA-TG-101/201), both LOA-TG-101/201 (step B.1.20) and LGP-2-1 (step E.3.17.1) give direction to throttle open the vacuum breaker, in an attempt to more quickly slow down the turbine-generator. Vibrations usually return to normal values once speed is below the critical speed range.

The LOA and the LGP differ slightly in the direction given for re-shutting the vacuum breaker. The LGP states to close the vacuum breaker when the turbine-generator speed has passed the critical speed. The LOA states to close the vacuum breaker when vacuum has been reduced low enough to slow the turbine-generator below critical speed. The procedures don't specify a particular RPM because, as discussed above, "critical" speed occurs within a range of speeds. There is no direct indication of critical speed and it must be determined by observation of key parameters, one of which is turbine vibration (others include bearing metal temperature and noise levels near the turbine-generator). The only way to determine if the turbine has passed through the critical speed band is by observation of these parameters.

The written exam tests the operator's ability to apply their knowledge and make decisions based on that knowledge. While answer choice 'd' is not a direct quote from the Licensee's procedures, observing that vibration levels have dropped below what is considered an excessive vibration level is an indirect indication that speed has dropped below the critical speed and therefore the vacuum breaker should be shut. Because the station's procedures direct closure of the vacuum breakers, answer choice 'd' was retained as the correct answer.

Examination Question #77

You are the Unit Supervisor for Unit 1. The following alarms (flashing red) are indicated on the Unit 1 Fire Detection Display:

- CONTROL RM ELEV 768' FZ 1-5
- VC RET AIR MON

Select the statement below that best describes your expected actions.

a. Ensure that the reactors in both units are shutdown, dispatch the Fire Brigade, notify plant personnel in both units of the fire location, and evacuate the Main Control Room.

b. Direct Main Control Room personnel to don emergency breathing air apparatus, dispatch the Fire Brigade, and notify plant personnel in both units of the fire location.

c. Ensure that the reactors in both units are shutdown, direct Main Control Room personnel to don emergency breathing air apparatus, direct the unit assist Reactor Operators to locate and extinguish the fire.

d. Verify that the Main Control Room HVAC system has shutdown and isolated, dispatch the Fire Brigade, notify plant personnel in both units of the fire location.

Correct Answer: b.

Applicant’s Comment:

Question 77 describes conditions indicating a fire/smoke in the main control room. In accordance with LOA-FP-101 UNIT 1 FIRE PROTECTION SYSTEM ABNORMAL, if the main control room is not habitable, you are directed to don supplied air. Actions according to LOA-FP-101 agree with answer “B”. LOA-RX-101 UNIT 1 CONTROL ROOM EVACUATION ABNORMAL is required to be entered if its entry conditions are met. The entry condition “Fire, smoke, explosion or other dangerous situation requiring personnel evacuation from the Control Room.” There was not enough information in the question to say whether it was safe for personnel to remain in the control room. It is reasonable to say with indication of fire and smoke in the control room evacuation may be required. If hazardous conditions exist, LOA-RX-101 would direct both units to be shut down and the control room evacuated making answer “A” also correct.

References:

LOA-RX-101 UNIT 1 FIRE PROTECTION SYSTEM ABNORMAL; and
LOA-FP-101 UNIT 1 FIRE PROTECTION SYSTEM ABNORMAL

LaSalle Management response:

Under the conditions listed in the question, LOA-FP-101 would always apply. If conditions were hazardous to control room personnel, at the unit supervisor’s discretion, LOA-RX-101 would be entered, both units scrammed and the control room evacuated. In either case, the fire brigade would be called to deal with the fire. Depending on the unit supervisors opinion of level of hazard to control room personnel, either answer “A” or “B” may be correct.

Recommended Disposition:

Recommend accepting both answers “A” and “B” based on review of LOA-RX-101 UNIT 1 CONTROL ROOM EVACUATION ABNORMAL and LOA-FP-101.

NRC Response:

The question stem contains no information that would lead an applicant to a conclusion that the control room must be evacuated. With only two annunciators flashing on the fire protection panel, and, absent any other indications of smoke, fire, or other hazardous conditions, the applicant cannot make an assumption that control room evacuation may be needed. The purpose of fire detection systems are to detect fires before the progress to the point of causing significant damage or personnel hazard and the question reviewers believed there was adequate information to select the identified correct answer. Since no additional information was provided, the applicant cannot assume that there was fire, smoke, or other extreme condition requiring evacuation from the control room. The applicants are instructed prior to starting the exam not to make assumptions unless they could occur as a consequence of the conditions stated in the question. In choosing answer choice ‘a’, the applicant would have to assume that there is a life threatening condition, which is not supported by the question stem.

Answer choice ‘b’ was retained as the only correct answer.

Examination Question #82

Unit 1 was operating at 100 percent power when a LOCA occurred, concurrent with a leak from the suppression pool. Following a successful reactor scram and isolation of the suppression pool leak, operators observed the following plant conditions:

- RPV Pressure is 25 psig
  Drywell Pressure is 25 psig
- Drywell Temperature is 250°F
- Suppression Chamber Pressure is 25 psig
- LPCS is injecting into the RPV @ 7500 gpm
- RPV water level is –150" FZ and rising very slowly
- RHR/LPCI "A" was recently shifted to Drywell Sprays
- Suppression Pool Level is –14 ft. (Suppression pool Leak)
- Suppression Pool Temperature is 200°F and is expected to remain constant
- NO other injection sources are currently available

Operating RHR/LPCI "A" in the Drywell Spray mode will FIRST cause _____(1)_____ and will require the Unit Supervisor to direct____(2)_____.

a. (1) LGA-001 Figure J limits to be exceeded
   (2) a realignment of RHR/LPCI "A" to inject into the RPV

b. (1) LGA-001 Figure NL limits to be exceeded
   (2) a LPCS pump flow reduction

c. (1) LGA-001 Figure NR limits to be exceeded
   (2) a continuation of current plant lineups

d. (1) LGA-003 Figured D limits to be exceeded
   (2) securing of Drywell Sprays

Correct Answer: c.

Applicant's Comment:

Answer C, LGA-001 "Figure NR limits to be exceeded" would be exceeded at approximately 3 psig in the suppression chamber. Distractor A, "LGA-001 Figure J limits to be exceeded" will be exceeded at approximately 18 psig (assuming RPV pressure is equalized with drywell pressure and tracks down with drywell pressure). Therefore Figure J would be exceeded first. The second part of answer C, "a continuation of current plant lineups" is incorrect: at approximately 3 psig in the suppression chamber/drywell, the proper action would be to secure the RHR pump in drywell sprays to prevent damage due to lack of NPSH. The only justified uses for ECCS with a lack of sufficient NPSH would be for adequate core cooling or to protect the containment. With the containment at 3 psig and the RPV depressurized, the containment is no longer in jeopardy and careful consideration should be given to the use of RHR for other than injection or suppression pool cooling. The LaSalle Shift

Operations Supervisor has given guidance to not realign ECCS for use other than RPV injection until level has been returned to wide range. Based on the question, level would not be on wide range.

References:

LGA-001 RPV CONTROL; and
LGA-003 PRIMARY CONTAINMENT CONTROL.

LaSalle Management response:

Distractor “B” states that Figure NL of EOP LGA-001 would be exceeded (LPCS NPSH curve) and that LPCS pump flow would have to be reduced. This distractor is incorrect since LPCS NPSH margin is not threatened at 200 degrees F.

Distractor “D” states that figure D (Drywell spray limit curve) will be exceeded and that Drywell sprays will have to be secured. This distractor is incorrect since decreasing Drywell pressure resulting from spray operation would not challenge the Drywell spray limit. EOP LGA-003 specifically states when to secure Drywell sprays.

Answer “C” states that Figure NR (RHR NPSH curve) would be exceeded with sprays in operation at the stated suppression chamber pressure. The RHR pump NPSH challenge would not occur until suppression chamber pressure had decreased to approximately 2 psig. The second part of answer “C” would not be prudent in this case; if suppression chamber pressure decreases to 2 PSIG, the correct action would be to secure the RHR pump (not continue operating it). This makes answer “C” an incorrect answer.

Answer “A” states that Figure ‘J’ limits would be exceeded and a realignment of RHR to RPV injection would be required. As the drywell is sprayed, containment pressure will be reduced which narrows the margin to this curve. At approximately a containment pressure of 18 PSIG this curve would be violated (assuming containment temperatures are not significantly reduced). This may potentially cause the loss of level instruments, which would procedurally require alignment of “A” RHR to RPV injection. Level is stated a –150 FZ (rising very slowly) and no mention of Wide Range level indication is provided. With only two injection sources available and level extremely low, prudent action would be to inject with available ECCS. Since the stem of the question asks which response would be FIRST, answer “A” would be the most correct answer to this question.

Recommended Disposition:

Recommend accepting only answer “A” as the correct answer.

NRC Response:

The initial conditions as specified, indicate that the drywell is not at saturated conditions (saturation temperature for 25 psig is about 267°F and initial drywell temperature is 250°F). When drywell spray is initiated the sprayed water droplets absorb heat from the surrounding atmosphere through convective heat transfer (sensible heat from the drywell atmosphere is transferred to the water), reducing drywell ambient temperature and pressure until equilibrium conditions are established. A rapid pressure decrease is not expected. Since the RPV is at saturated conditions, the reactor pressure decrease will lag the drywell pressure decrease. Drywell temperature is expected to decrease such that the Figure J limits are not reached (temperature will be below 212°F before reactor pressure drops below 20 psig). Therefore, the first part of answer choice ‘a’ will not be satisfied.

Additionally, realignment of RHR/LPCI “A” is not mandated simply because Figure J limits are exceeded. Operators are also directed to refer to Table K to evaluate level instrument availability. Given the conditions specified in the stem of the question, Fuel Zone level indicators may be used as long indicated level is reading above – 311 inches. Since RPV water level, as given, is -150 inches and rising, there is no immediate need, or requirement, to realign RHR/LPCI “A” for injection into the RPV.

The applicant’s comment regarding the second part of answer choice ‘c’ is valid. If Figure NR limits are exceeded, and core cooling is assured, continued operation of RHR/LPCI “A” would not be warranted. While operation of RHR/LPCI “A” in the suppression pool cooling mode may seem appropriate, there is no immediate threat to the containment since the RPV is already depressurized.

Based on the above, there was no correct answer provided for this question and the question will be deleted. The answer key was modified to reflect deletion of this question.

Examination Question #83

A unit startup was in progress on Unit 2 in accordance with the Normal Unit Startup LGP. All conditions for entering Mode 1 had been satisfied. When repositioning the Reactor Mode switch, the Reactor Operator inadvertently rotated the Reactor Mode Switch to SHUTDOWN. The following conditions existed after the SCRAM:

- Five control rods, various positions and widely scattered throughout the core, have failed to insert beyond position 04
- Reactor power is decreasing with a –80 second period and is currently indicating on Ranges 2 and 3 of the IRMs
- Reactor water level is being maintained at +20 inches
- Reactor pressure is 900 psig and decreasing slowly

Which of the following identifies the appropriate procedure(s) to be entered?

a. LGP-3-2, Reactor Scram and LGA-NB-01, Alternate Rod Insertion ONLY

b. LGA-001, RPV Control and LGP-3-2, Reactor Scram ONLY

c. LGA-001, RPV Control, LGP-3-2, Reactor Scram, and LGA-NB-01, Alternate Rod Insertion

d. LGA-010, Failure to Scram (entered from LGA-001 and LGA-NB-01, Alternate Rod Insertion

Correct Answer: a.

Applicant’s Comment:

The body of the question stated that reactor water level is being maintained at +20 inches. Prior to the SCRAM level setpoint would have been 36“, with level control in Automatic. Nothing in the question indicates whether level stayed above 20 inches or if level shrank below and has not recovered. If level remained above 20 inches, there would be no entry into LGA-001 so answer 'A' would be correct. It is also likely that with level being controlled at 20 inches, the post scram profile had been activated and a significant level shrink occurred due to the scram. The post scram profile remains activated if a reactor scram occurs and level drops below 20 inches. The function of the post scram profile is to slowly restore level to 20 inches where reactor water level control switches to single element auto at 20 inches. Based on my experience, I would expect level to shrink when the reactor scrams from this power level. Based on indications that the post scram profile has remained activated and expected plant response, I believe level would be expected to drop below 11 inches where LGA-001, RPV Control would be required to be entered. This makes answer “D” also correct.

References:

LGA-001 RPV CONTROL;
LGP-3-2 REACTOR SCRAM;
RWLC System lesson plan for description of the post scram profile; and
LaSalle Simulator.

LaSalle Management Response:

First of all, it is plausible that answer “A” is correct based on information in the stem. The stem indicates that level is being maintained at 20 inches. The stem doesn’t say whether level remained above 20" or not. It is also plausible that answer “D” is correct. Reactor water level may have fallen below 20" and now recovered. The fact that reactor water level is being maintained at 20” instead of the normal 36" implies that a level shrink occurred as a result of the scram. Expected plant response from this plant condition would be for a significant level shrink to occur. This scenario was recreated on the simulator. The reactor was scrammed from the point where the unit met the conditions for entering mode one to observe level response. When the reactor was scrammed, reactor water level shrank to about 0 inches reactor water level. LGA-001 is required to be entered at 11 inches so with 5 rods out, entry into LGA-010 would be required so answer “D” should also be considered correct. There is not enough information in the stem to differentiate between Answer “A” and “D”.

Recommended Disposition:

It is recommended that both “A” and “D” answers be accepted as correct.

NRC Response:

The applicant and Licensee Management make the argument that the condition of RPV water level being maintained at 20” implies that level must have shrunk below the LGA-001 entry condition for level (11”). The Licensee management response discussed a scenario run on the simulator, with initial conditions similar to those specified in the question, in which RPV water level shrank to about 0”. Since distractor “d.” was supported by the simulator response, one <u>must</u> assume that level went below 11" as a consequence of the conditions provided by the question and entry into LGA-001 and subsequently into LGA-010 would be required, making answer choice ‘d’ the correct answer and disqualifying distractor “a.” as a correct answer.

Base on the above, the answer key was modified to accept distractor ‘d’ as the only correct answer.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

SRO Initial Examination ADAMS Accession # ML070230529