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

December 22, 2016

Mr. Bryan C. Hanson
Senior VP, Exelon Generation Company, LLC
President and CNO, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 – NRC INITIAL LICENSE EXAMINATION REPORT 05000373/2016301; 05000374/2016301

Dear Mr. Hanson:

On November 16, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your LaSalle County Station, Units 1 and 2. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on November 4, 2016, with Mr. W. Trafton and other members of your staff. An exit meeting was conducted by telephone on December 1, 2016, between Mr. Trafton of your staff, and Mr. Zoia, Chief Operator Licensing Examiner, 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 16, 2016, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of October 17, October 24, and October 31, 2016. The written examination was administered by training department personnel on November 4, 2016. Thirteen Senior Reactor Operator and eleven Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on December 13, 2016. Three applicants failed one or more sections of the administered examination and were issued proposed license denial letters. Twenty-one applicants passed all sections of their respective examinations and twelve were issued senior operator licenses and seven were issued operator licenses. In accordance with NRC policy, the licenses for the remaining two applicants are being withheld pending the outcome of any written examination appeal that may be initiated.

The written examination and other related written examination documentation will be withheld from public disclosure for 24 months per your request. However, if an applicant received a proposed license denial letter, because of a written examination grade that is less than 80.0 percent, the applicant will be provided a copy of the written examination. For examination security purposes, your staff should consider that written examination uncontrolled and exposed to the public.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, “Public Inspections, Exemptions, Requests for Withholding,” of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert J. Orlikowski, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 05000373; 05000374
License Nos. NPF–11; NPF–18

Enclosures:
1. OL Examination Report 05000373/2016301; 05000374/2016301
2. Simulation Facility Fidelity Report

cc: Distribution via LISTSERV®
J. Lindsey, Training Director,
LaSalle County Station

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos:	05000373; 05000374
License Nos:	NPF–11; NPF–18
Report No:	05000373/2016301; 05000374/2016301
Licensee:	Exelon Generation Company, LLC
Facility:	LaSalle County Station, Units 1 and 2
Location:	Marseilles, IL
Dates:	October 17 – November 16, 2016
Inspectors:	C. Zoia, Chief Examiner J. Seymour, Examiner/ Chief Examiner-in-Training M. Bielby, Examiner
Approved by:	R. Orlikowski, Chief Operations Branch Division of Reactor Safety

SUMMARY

Examination Report 05000373/2016301; 05000374/2016301; 10/17/2016 – 11/16/2016; 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 Nuclear Regulatory Commission examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 10, Supplement 1.

Examination Summary

Twenty-one of twenty-four applicants passed all sections of their respective examinations. Twelve applicants were issued senior operator licenses and seven applicants were issued operator licenses. Three applicants failed one or more sections of the administered examination and were issued proposed license denials. The licenses for the remaining two applicants are being held and may be issued pending the outcome of any written examination appeal. (Section 4OA5.1).

REPORT DETAILS

4OA5 Other Activities

.1 Initial Licensing Examinations

a. Examination Scope

The U.S. Nuclear Regulatory Commission (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 10, 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 September 19, 2016, with the assistance of members of the facility licensee's staff. During the on-site validation week, the examiners audited three 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 17, 2016, through November 3, 2016. The facility licensee administered the written examination on November 4, 2016.

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.

All changes made to the proposed written examination, were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and documented on Form ES-401-9, "Written Examination Review Worksheet." On November 16, 2016, the licensee submitted documentation noting that there were eight 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 with this report. The Form ES-401-9, the written examination outlines (ES-401-2 and ES-401-3), and both the proposed and final written examinations will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS) in 24 months. (ADAMS Accession Numbers ML15274A405 and ML15274A403).

The NRC examiners graded the written examination on November 29, 2016, 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 the Publicly Available Records component of NRC's ADAMS.

The NRC examiners completed operating test grading on December 13, 2016.

(3) Examination Results

Thirteen applicants at the Senior Reactor Operator level and eleven applicants at the Reactor Operator level were administered written examinations and operating tests. Nineteen applicants passed all portions of their examinations and were issued their respective operating licenses on December 13, 2016.

Three applicants failed one or more sections of the administered examination and were issued proposed license denials. Two applicants passed all portions of the license examination, but received a written test grade below 82 percent. In accordance with NRC policy, the applicants' licenses 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 incorrectly.

.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 Title10 of the *Code of Federal Regulations*, Section 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 November 4, 2016, to W. Trafton, Site Vice-President, and other members of the LaSalle County Station Operations and Training Department staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on December 1, 2016, with Mr. W. Trafton, Site Vice President, by telephone. The NRC's final disposition of the station's post-examination comments were disclosed and discussed with Mr. Trafton 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

W. Trafton, Site Vice President
H. Vinyard, Plant Manager
G. Ford, Regulatory Assurance Manager
T. Lanc, Regulatory Assurance
J. Keenan, Operations Director
M. Smith, Operations Shift Manager
J. Lindsey, Training Director
D. Wright, Operations Training Manager
D. Fuson, Training Specialist
C. Betken, Operations Instructor
J. Fiesel, Maintenance Director
M. Fakhreddine, Chemistry
B. Roy, Fleet Assessment

U.S. Nuclear Regulatory Commission

R. Ruiz, Senior Resident Inspector
C. Hunt, Acting Resident Inspector
C. Zoia, Chief Examiner
J. Seymour, Examiner/ Chief Examiner-in-Training
M. Bielby, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
JPM	Job Performance Measure
LPRM	Local Power Range Monitor
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
RCMS	Rod Control Management System
SWR	Simulator Work Request

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: LaSalle County Station, Units 1 and 2

Facility Docket Nos: 50-373; 50-374

Operating Tests Administered: October 17, 2016, – November 3, 2016

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 Title 10 of the *Code of Federal Regulations* 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
Exelon Nuclear Issue #02733595	While performing a simulator reset during NRC exam simulator scenarios, a simulator operator failed to correctly position an LPRM bypass switch according to the lineup that was required for the scenario guide. This human performance error was not detected by the simulator operator due to the use of an override function in the simulator software that permitted the simulator to be reset for the next scenario with the switch remaining out of position. As a result, a malfunction inserted during the subsequent scenario did not present itself to the applicants. The NRC examiners removed the crew of applicants from the simulator and sequestered them while facility staff investigated and corrected the issue. Following a delay of approximately 45 minutes, the applicants were returned to the simulator and the scenario was resumed. The scenario was then completed without further simulator issues.
SWR #0132672	During a JPM that required synchronizing an emergency diesel generator to a bus, there was one instance in which the associated output breaker failed to close. The applicant was removed from the simulator and sequestered by an NRC examiner during facility investigation of the issue. Facility simulator staff were unable to reproduce the issue. Following a brief delay, the applicant was returned to the simulator and the JPM was resumed. The JPM was then completed without further simulator issues.

SWR #0132673	During a JPM that required synchronizing an emergency diesel generator to a bus, there was one instance in which the associated sync selector switch failed to turn on the synchroscope. The applicant was removed from the simulator and sequestered by an NRC examiner during facility investigation of the issue. Facility simulator staff were unable to reproduce the issue. Following a brief delay, the applicant was returned to the simulator and the JPM was resumed. The JPM was then completed without further simulator issues.
SWR #0132673	During the operating portion of the NRC exam, there were multiple instances of the 1H13-P601 silence and test buttons sticking. This resulted in applicants being unable to silence annunciators from this location and necessitated the silencing of annunciators from another location in the simulator.
N/A	During multiple simulator scenarios, a persistent RCMS-related annunciator was present. This was not an expected alarm for the conditions established by the scenario guides. The presence of this alarm did not interfere with the execution of the simulator scenarios or cause distraction to the applicants. Due to the nature of the underlying simulator issue, and the potential impacts of repairs on simulator availability, facility simulator staff deferred repair of the issue until after the operating portion of the NRC exam had been completed.

POST EXAM COMMENTS AND RESOLUTIONS

RO Question 18 (Post-Exam Comment #1)

Original Question:

Unit 1 is at 100% power.

The NSO starts 1A CD/CB pump and secures 1B CD/CB pump.

One minute later, Annunciator 1H13-P601-F402, MSL A/B Radiation Monitor HI alarms.

(1) Is the MSL A/B Radiation Monitor High alarm expected or unexpected?

(2) What is the correct operator action, if any?

A. (1) Unexpected.
(2) Commence power reduction per LGP 3-1.

B. (1) Unexpected.
(2) Direct all nonessential personal to stay clear of Turbine Building Elevation 768.

C. (1) Expected.
(2) No additional action required. Monitor parameters and trends; annunciator 1N62-P600-B502, OFF GAS PRE-TREATMENT RADIATION HI may alarm.

D. (1) Expected.
(2) No additional action required. Monitor parameters and trends; annunciator 1N62-P600-B304, STATION VENT STACK RAD HI may alarm.

Answer: C

Applicant Feedback:

Answers A and C are both correct and supported by procedures and engineering changes. Additionally, per Exelon standards per OP-AA-103-102, Section 4.5.5, alarms that are not previously flagged with LOR's reviewed are not expected which eliminate answers (C) and (D).

Facility Response:

A review of OP-AA-103-102 does state that if an alarm is not previously discussed then the alarm should be considered unexpected. Also a recent plant modification per EC 364641 to relocate the hydrogen injection points in the condensate header has been successful at reducing the frequency of the MSL Rad Monitor HI alarms following CD/CB pump lineup changes. The question provides the candidate with having to determine if the alarm was an expected alarm based on the definition from OP-AA-103-102 without clarifying information in the stem. Without this information and the engineering change reducing the frequency of the alarm then the correct answer is no longer valid since the 1st part of the two parts states EXPECTED. Since both C and D state EXPECTED, they can be considered incorrect per procedural process. In addition, the student could make the assumption that the alarm was caused by a fuel failure, due to the engineering change, and therefore the alarm would be unexpected and require a power reduction as stated in answer A.

STATION RECOMMENDATION: There is NO correct answer

References:

OP-AA-103-102, Watch-Standing Practices, Revision 16

(reference withheld from public disclosure due to proprietary content)

EC 364639, Mitigate MSL Rad Monitor Spikes, Revision 2

This reference states, in part, that:

LaSalle Station, in the mid 1990's, installed plant modifications to inject hydrogen into the reactor feedwater, the Hydrogen Water Chemistry (HW) system. The purpose of the modifications was to protect the Reactor Internals and Reactor Recirculation Piping by reduction of Intergranular Stress Corrosion Cracking (IGSCC).

This modification (EC 364639) will relocate the current point of hydrogen injection into line 1CD07A-30" suction piping, from between 1CBO1PB (1B) and 1CBO1PC (1C) Condensate Booster (CB) pumps to a point in line 1CD07A-30" upstream of all four Condensate Booster pumps. This relocation of hydrogen injection will reduce the MSL radiation spikes by providing a more uniform mixing of hydrogen in the Condensate Booster system.

EC 364641, Mitigate MSL Rad Monitor Spikes, Revision 0

This reference states, in part, that:

LaSalle Station, in the mid 1990's, installed plant modifications to inject hydrogen into the reactor feedwater, the Hydrogen Water Chemistry (HW) system. The purpose of the modifications was to protect the Reactor Internals and Reactor Recirculation Piping by reduction of Intergranular Stress Corrosion Cracking (IGSCC).

This modification (EC 364641) will relocate the current point of hydrogen injection into line 2CD07 A-30" suction piping, from between 2CB01PB (2B) and 2CB01PC (2C) Condensate Booster (CB) pumps to a point in line 2CD07 A-30" upstream of all four Condensate Booster pumps. This relocation of hydrogen injection will reduce the MSL radiation spikes by providing a more uniform mixing of hydrogen in the Condensate Booster system.

LOR-2H13-P601-F402, MSL A/B Radiation Monitor Downscale/INOP/HI, Revision 3

This reference states in section C.2 that:

During CP changes and CD/CB Pump swaps MSL Rad monitor HI alarms have spuriously annunciated in the past when HWC is online. This phenomenon is an actual radiation level change induced by N-16 production, which is a normal by-product of H2 gas injection into the reactor. The suspect cause is a release of H2 gas within the CD/CB piping from a pocketed location. When this finite amount of gas reaches the reactor, it results in the formation of N-16 and is detected as a spike on the MSL and OG Pretreatment Rad monitors.

<u>LOR-1N62-P600-B502, Off Gas Pre-Treatment Radiation Monitor High Radiation, Revision 8</u>

This reference states in section C that:

During CP changes and CD/CB pump swaps, OG Pretreatment Rad Monitor HI alarms have spuriously annunciated in the past when HWC is on-line. This phenomenon is an actual radiation level change induced by N-16 production, which is a normal by product of H2 gas injection into the reactor. The suspect cause is a release of H2 gas within the CD/CB piping from a pocketed location. When this finite amount of gas reaches the reactor, it results in the formation of N-16 and is detected as a spike on the MSL and OG Pretreatment Rad Monitors.

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. It was noted that the LOR procedures still discuss a possible high radiation alarm condition. It was also noted that an engineering change was made to prevent such high radiation alarms. The definition of ‘expected alarm’ as it relates to the question was determined to add further ambiguity. Based upon these considerations, the NRC concluded that no correct answer exists for this question. Therefore, RO question #18 has been deleted from the exam.

<u>RO Question 27 (Post-Exam Comment #2)</u>

Original Question:

Unit 1 is in a LOCA

(3) Drywell pressure is 10 psig and rising slowly
(4) Drywell and Suppression Chamber Hydrogen is 1%
(5) Drywell and Suppression Chamber Oxygen is 2%

What action is required?

The Hydrogen Recombiner must be.....

A. STOPPED manually

B. STARTED manually

C. Verified to have AUTO-TRIPPED

D. Verified to have AUTO-STARTED

Answer: B

Applicant Feedback:

The comment made was that drywell pressure of 10 psig and rising slowly does not adequately describe the time to reach 15.3 psig which is the point at which the Hydrogen Recombiner will trip. The Hydrogen Recombiner should be stopped manually prior to reaching 15.3 psig which makes (A) also correct.

Facility Response:

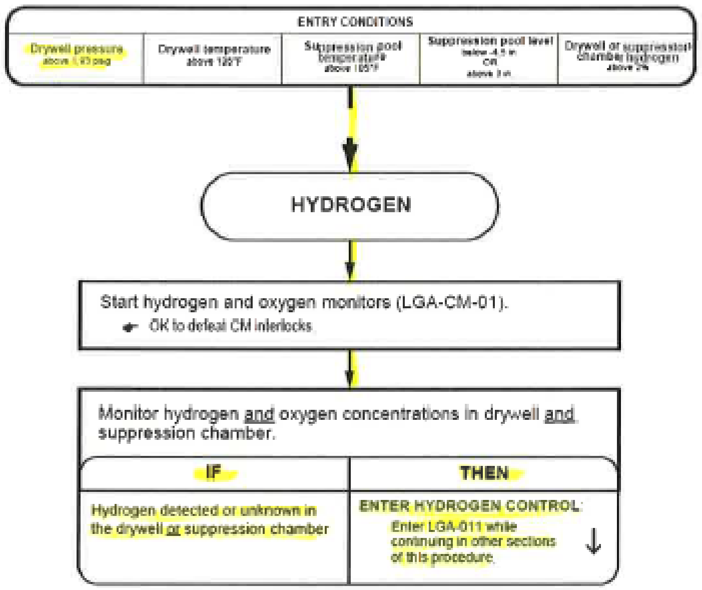
LGA-003 lesson plan states if LGA-003 is entered then parallel execution is also required because of the symptomatic approach to emergency response precludes the prioritization of any one action path since independence for initiating events and transients must be maintained.

Therefore, the Hydrogen leg of LGA-003 is entered because the stem of the question states that Hydrogen is 1%, which leads to entering LGA-011 and starting the Hydrogen Recombiner. There is procedure guidance in LGA-HG-101 to shutdown the Hydrogen Recombiner when drywell pressure exceeds 15.3 psig, but the Hydrogen Recombiner does not have an auto start feature, and therefore would not be running to require STOPPING Manually. The LGA-HG-101 indicates that the Hydrogen Recombiner will trip at 15.3 psig of drywell pressure and direct the operator to use containment sprays to reduce drywell pressure before restarting the Hydrogen Recombiner.

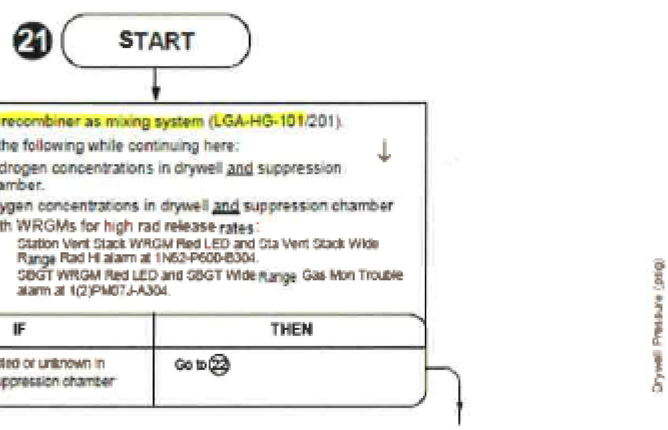
STATION RECOMMENDATION: ACCEPT ONLY (B) as the correct answer

References:

LGA-003 PRIMARY CONTAINMENT CONTROL



LGA-011 HYDROGEN CONTROL



<u>LGA-HG-101, Operation of the Hydrogen Recombiner as a Mixing System, Revision 1</u>

This reference states in section E.1.b that:

___ b. IF at any time during the performance of this procedure, Drywell pressure exceeds 15.3 psig,

___ THEN GO TO Step E.4 to shutdown the Hydrogen Recombiner.

This reference states in section F.2 that:

2. The Hydrogen Recombiner blower will trip and the inlet valve will close on High Inlet Pressure (greater than 30 psia/15.3 psig), on High HG Return Gas Temperature (250 °F) or on High HG Blower Inlet Temperature (250 °F). It may be necessary to use containment spray to reduce the pressure before the blower can be restarted again.

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. The stem of the question is clear that drywell pressure is currently 10 psig with a slowly rising trend. Per LGA-HG-101, the recombiner is manually stopped after 15.3 psig is exceeded, which also corresponds to the automatic trip setpoint. Based upon these considerations, the NRC concluded that there should be no change to key for RO question #27.

RO Question 40 (Post-Exam Comment #3)

Original Question:

RCIC is operating in the PRESSURE CONTROL MODE with the RCIC Pump Discharge Flow Controller in AUTO set to 600 GPM.

Which of the following set of RCIC system control manipulations would result in the FASTEST RATE of RISE in Suppression Pool water temperature?

Throttle 1E51-F022, Full Flow Test Upstream Valve, (1)___, in order to (2)___.

A. (1) Open
- Maximize pump flowrate

B. (1) Closed
- Maximize pump flowrate

C. (1) Open
- Maximize pump discharge pressure

D. (1) Closed
- Maximize pump discharge pressure

Answer: D

Applicant Feedback:

Closing the 1E51-F022 causes the turbine to spin faster and output more heat to the suppression pool. Flow will remain at the maximum of 600 GPM. In automatic 600 GPM is the maximum and flow will go no higher which makes (B) also correct.

Facility Response:

Station's Response: The RCIC operating procedure for the pressure control mode per LOP-RI-09 indicates that the parameter controlled when throttling 1E51-F022 is RCIC discharge pressure and RCIC flowrate will be automatically maintained at the flow controller setpoint when operated in AUTO. The RCIC system lesson plan states LE51-F022 is throttled (to increase pressure and the turbine (pump) speed is automatically adjusted to achieve the flow specified.

STATION RECOMMENDATION: ACCEPT ONLY (D) as the correct answer

References:

LOP-RI-09, Operating the Reactor Core Isolation Cooling System for Pressure Control, Revision 11

This reference states in step E.1.10 that:

E.1.10 THROTTLE 1(2)E51-F022, RCIC Full Flow Test Upstrm Valve, as needed to maintain desired Rx pressure and/or cooldown rate.

This reference states in Attachment A (Hardcard – RCIC Operations) step 3 that:

3. To change Cooldown Rate:

- THROTTLE 1(2)E51-F022, RCIC PMP TEST TO CY UPSTREAM VLV, as needed, to control Reactor Pressure.

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. It was determined that system operation and procedural direction support the answer key. Furthermore, it was determined that the selection of distractor 'B' would require an incorrect understanding of system operation. Throttling FO-22 with the controller in "AUTO" would not be expected to cause a flow change; if a max flow rate was wanted, the flow controller would need to be adjusted. Based upon these considerations, the NRC concluded that there should be no change to key for RO question #40.

RO Question 53 (Post-Exam Comment #4)

Original Question:

Unit 1 is operating at rated power

(6) A trip of the 1A Service Water Pump results in a Service Water low pressure alarm.
(7) Shortly thereafter, 1H13-P601-B301, SERV WTR EFFLUENT RAD HI alarms.
(8) NO other alarms have been received on 1H13-P601.

What is the source of the rising radiation levels?

A. RBCCW Heat Exchangers

B. TBCCW Heat Exchangers

C. Fuel Pool Cooling Heat Exchangers

D. Primary Containment Ventilation Chiller Condensers

Answer: C

Applicant Feedback:

RBCCW is cooled by service water and with reduced service water pressure RBCCW can leak into service water. RBCCW is a potentially contaminated system. The applicant believes that both 'A' and 'C' are correct answers.

Facility Response:

RBCCW is filled with makeup condensate and is chemically treated. Leakage of contaminated water into the RBCCW system is possible. The RBCCW system has a process radiation monitor which also alarms on 1H13-P601. The stem of the question indicates that no other alarms have been received on 1H13-P60L and indicates that the leak is not into the RBCCW system.

STATION RECOMMENDATION: ACCEPT ONLY (C) as the correct answer

References:

LOR-1H13-P601-B301, Service Water Effluent Radiation High, Revision 2

This reference states in section B.4 that:

4. Loads on service water which are monitored by Service Water Effluent Radiation Monitor are:

   a. RBCCW Heat Exchangers

   b. Fuel Pool Cooling Heat Exchangers.

   c. Primary Containment Chiller Heat Exchangers and Pump Out Coolers.

   d. Aux Bldg Lab Air Conditioner.

   e. Aux Bldg Office Air Conditioner.

   f. Laundry Reverse Osmosis Unit.

   g. Unit 1 Floor Drain and Chemical Waste Concentrator Units (Concentrate After Cooler, Concentrate Receiver, Condenser, and Distillate After Cooler).

<u>LOR-1H13-P601-B401, Reactor Building Closed Cooling Water Radiation High, Revision 2</u>

This reference states in section B that:

B. <u>OPERATOR ACTIONS</u>

1. VERIFY RBCCW Monitor Mode Selector Switch is in OPERATE.

2. CHECK RBCCW Radiation Monitor at Panel 1H13-P604 or on Recorder 1D18-R610 at 1H13-P600 indicates a current or recent upward trend.

3. NOTIFY Chemistry Department to sample RBCCW System for radioactivity concentration.

4. DETERMINE source of Radiation and ISOLATE source from RBCCW System if possible.

5. Contact Rad Protection for any setpoint/calibration information and to troubleshoot, if necessary.

62 INITIATE Action Request if applicable.

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. It was determined that the stem conditions of the question clearly rule out the RBCCW activity necessary to make distractor “A” correct. Specifically, no high RBCCW activity alarm is present, and therefore, a leak from RBCCW will not cause high activity in Service Water. Based upon these considerations, the NRC concluded that there should be no change to key for RO question #53.

RO Question 58 (Post-Exam Comment #5)

Original Question:

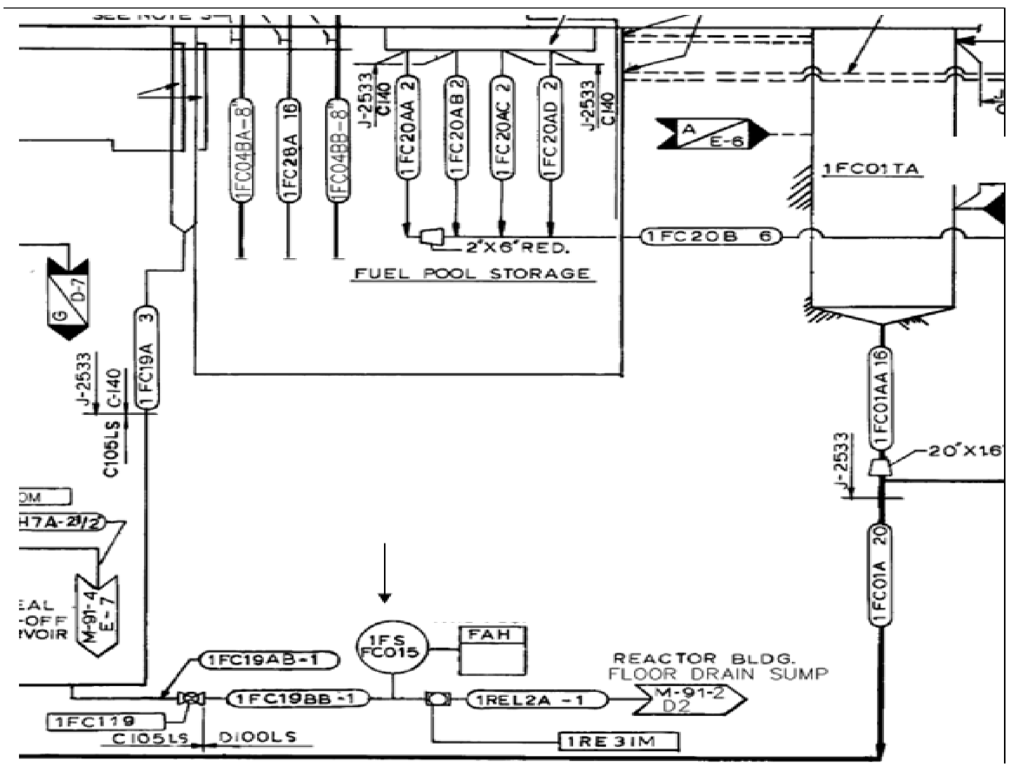
Unit 1 is at 100% power.
Leakage flow has become just high enough to activate the Flow Switch 1FS-FC015 shown below.
Identify the location(s) where this Flow switch initiates an alarm.

A. In the MCR ONLY on panel 1PM13J

B. In the MCR ONLY on panel 1H13-P601.

C. In the Turbine Building at the Fuel Pool Cooling Panel ONLY

D. In the MCR AND in the Turbine Building at the Fuel Pool Cooling Panel



Answer: B

Applicant Feedback:

The Main Control Room can be identified as the correct from the print, and filling the reactor building floor drain sump will cause an alarm on 1PM13J. The applicant believes that the question requires memorization of all Main Control Room Alarm tiles and that the question should be removed from the exam.

Facility Response:

Leakage through 1FS-FC015 flow switch passes through the 1REL2A-1 drain line as shown on the provided drawing from M-98-L. This continues to M-91-2 to M-104-1 to M-104-2 to the 1RF02 sump. If the water addition to this sump may cause an alarm 1PM13J-8402 on the 1PML3J for excessive pump-out time, excessive pump start frequency or tank hi-hi level. The stem of the question is discussing the high flow alarm from flow switch 1FS-FC015 which will alarm on 1H13-P601. There are two alarms that could result from leakage flow from the fuel pool. These alarms are on two different panels in the Main Control Room, the 1PM13J and the 1H13-P601. Answers A and B are correct and since they state ONLY then neither answer can be completely correct, therefore there is no correct answer.

STATION RECOMMENDATION: There is NO correct answer

References:

LOR-1H13-P601-C207, Fuel Pool Cooling System Trouble, Revision 3

This reference states in section A that:

LIST OF ALARMS

REC. INPUT	SETPOINT	ACTUATING DEVICE	ALARM NO.	PRINTOUT MESSAGE	STATUS
R0529	842'1"	1LIS-FC043	1FC01L	FUEL POOL LEVEL	LO
R0530	145 psig	1PS-FC007B	1FC01A	1A FC PMP DSCH PRESS	HI
R0531	145 psig	1PS-FC008B	1FC01B	1B FC PMP DSCH PRESS	HI
R0532	16'6"	1LS-FC016B	1FC01C	SKIMMER SURGE TANK LVL	HI
R0533	4'6"	1LS-FC016C	1FC01D	SKIMMER SURGE TANK LVL	LO
R0534	842'7.75"	1LIS-FC043	1FC01E	FUEL POOL LEVEL	HI
R0535	1.92 gpm	0FS-FC009	1FC01F	FUEL POOL XFR CANAL LKG	ALARM
R0536	1.92 gpm	1FS-FC039	1FC01G	DRYER-SEP STRG PIT LKG	ALARM
R0537	1.92 gpm	1FS-FC040	1FC01H	FUEL POOL RX WELL LKG	ALARM
R0538	1.92 gpm	1FS-FC041	1FC01I	FUEL POOL LEAKAGE	ALARM
R0539	1.31 gpm	1FS-FC014	1FC01J	RX VESSEL DRYWELL LEAK	ALARM
R0540	0.99 gpm	1FS-FC015	1FC01K	FUEL POOL GATE LEAKAGE	ALARM

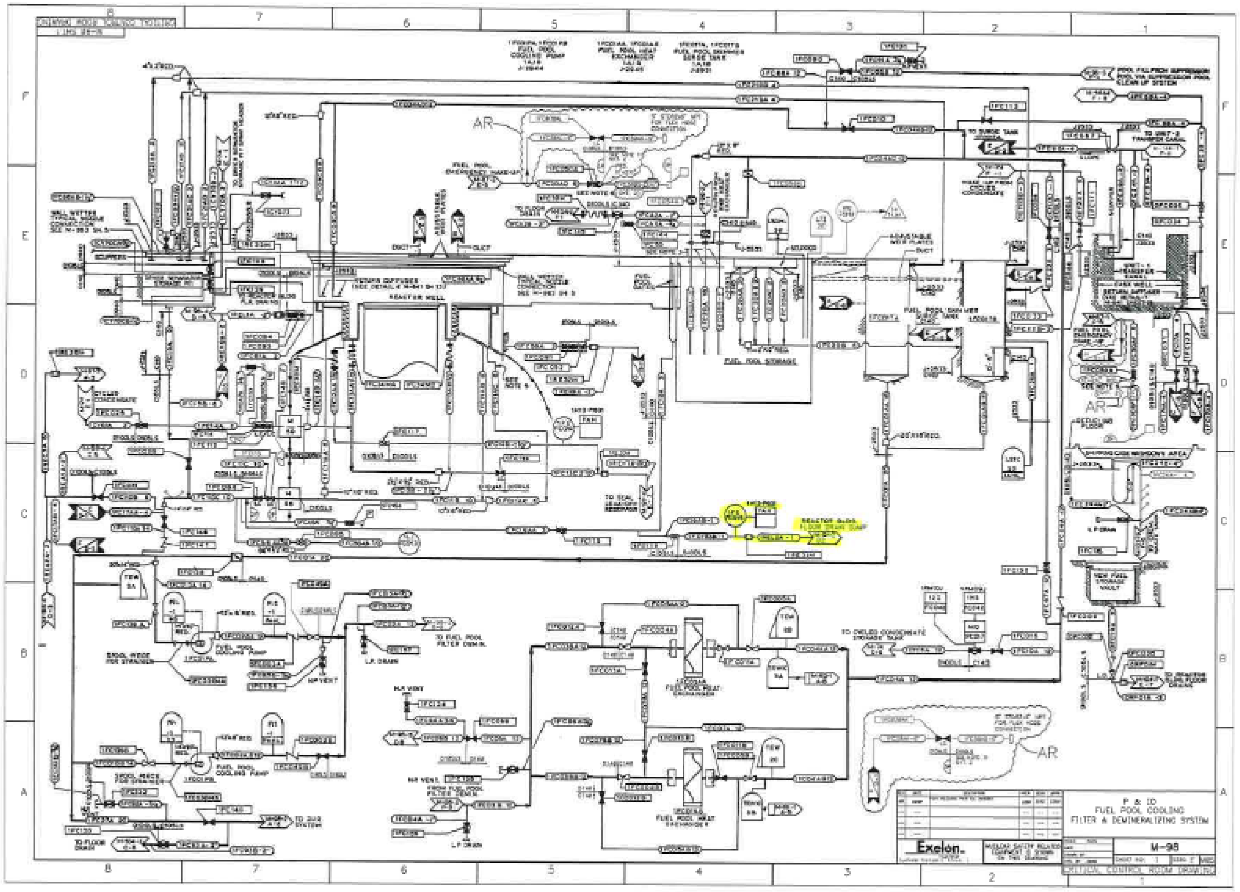
LOR-1PM13J-B402, South Reactor Building Floor Drain Sump Excessive Pump Out Time, Pump Excessive Start Freq., or Hi-Hi Level, Revision 2

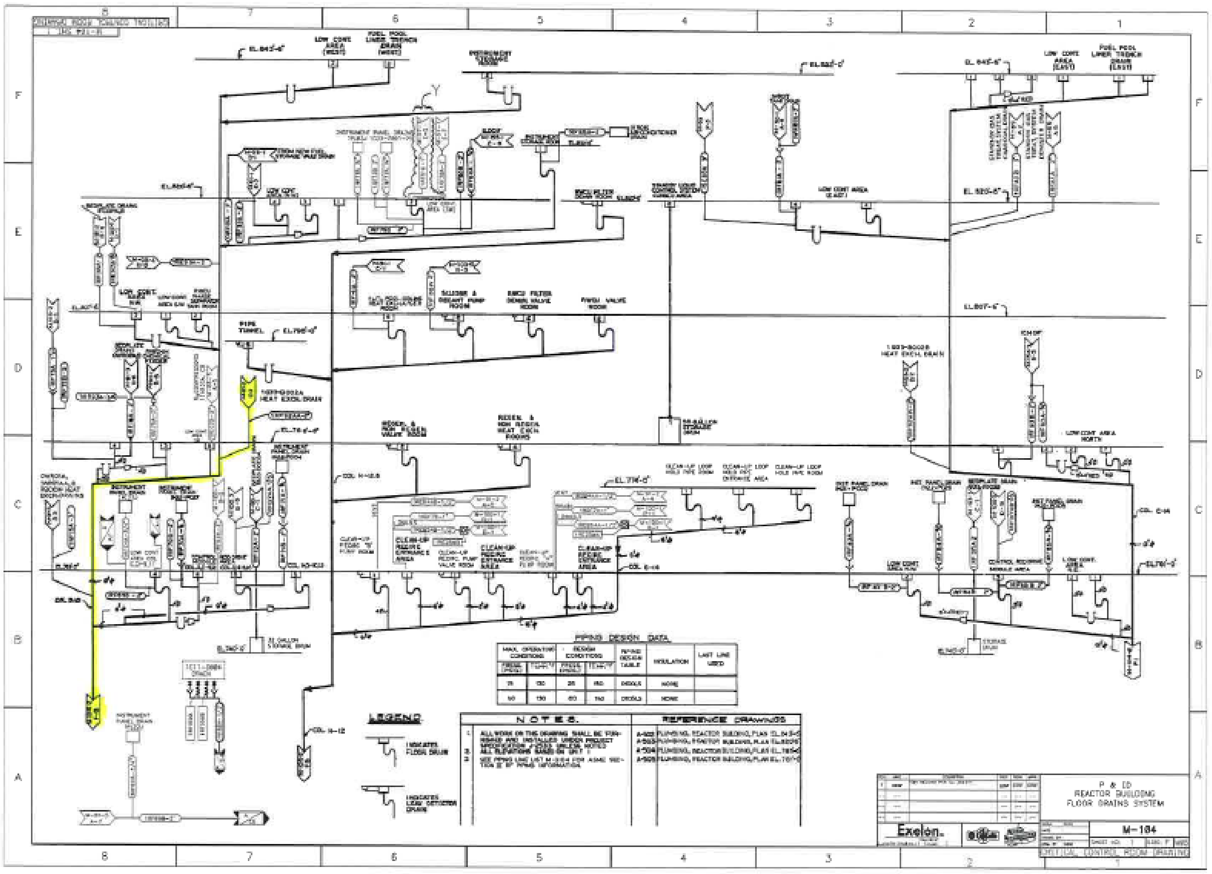
This reference states in section A that:

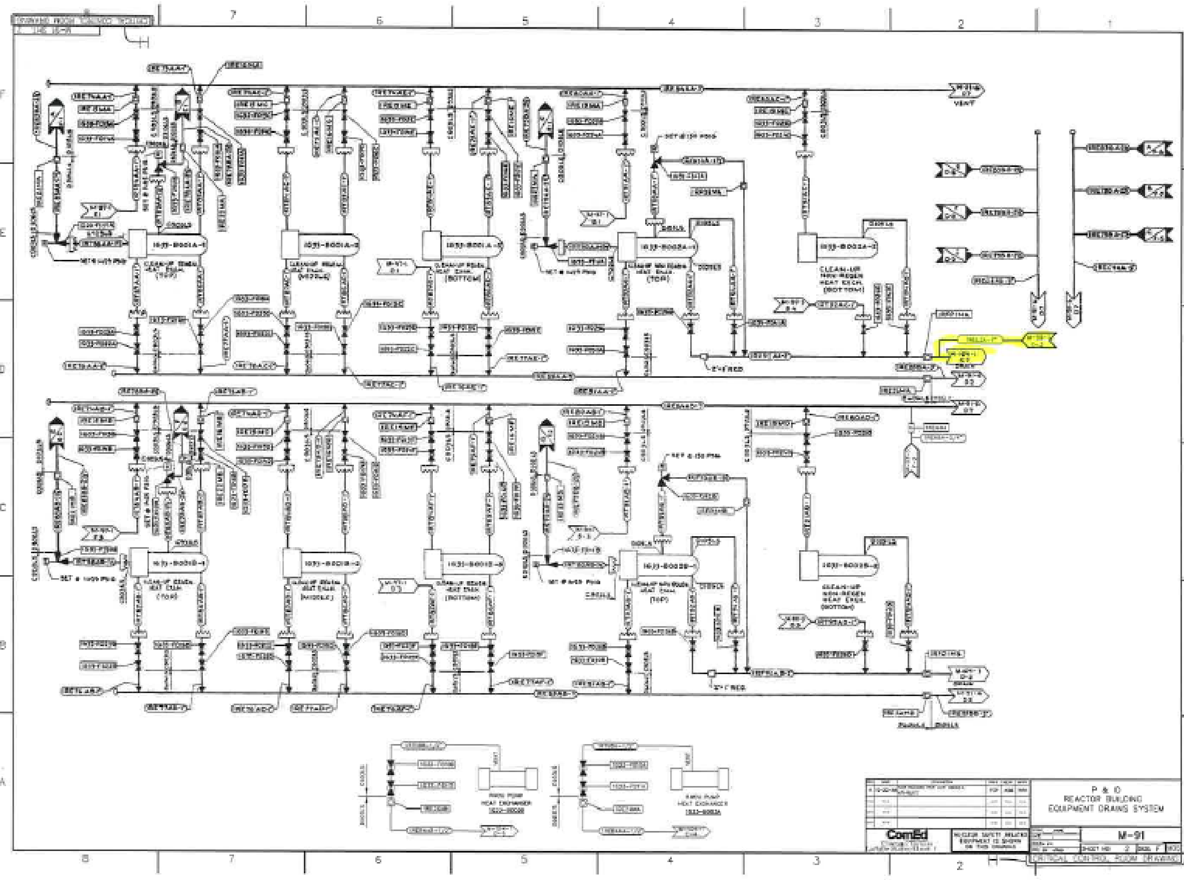
A. AUTOMATIC ACTIONS

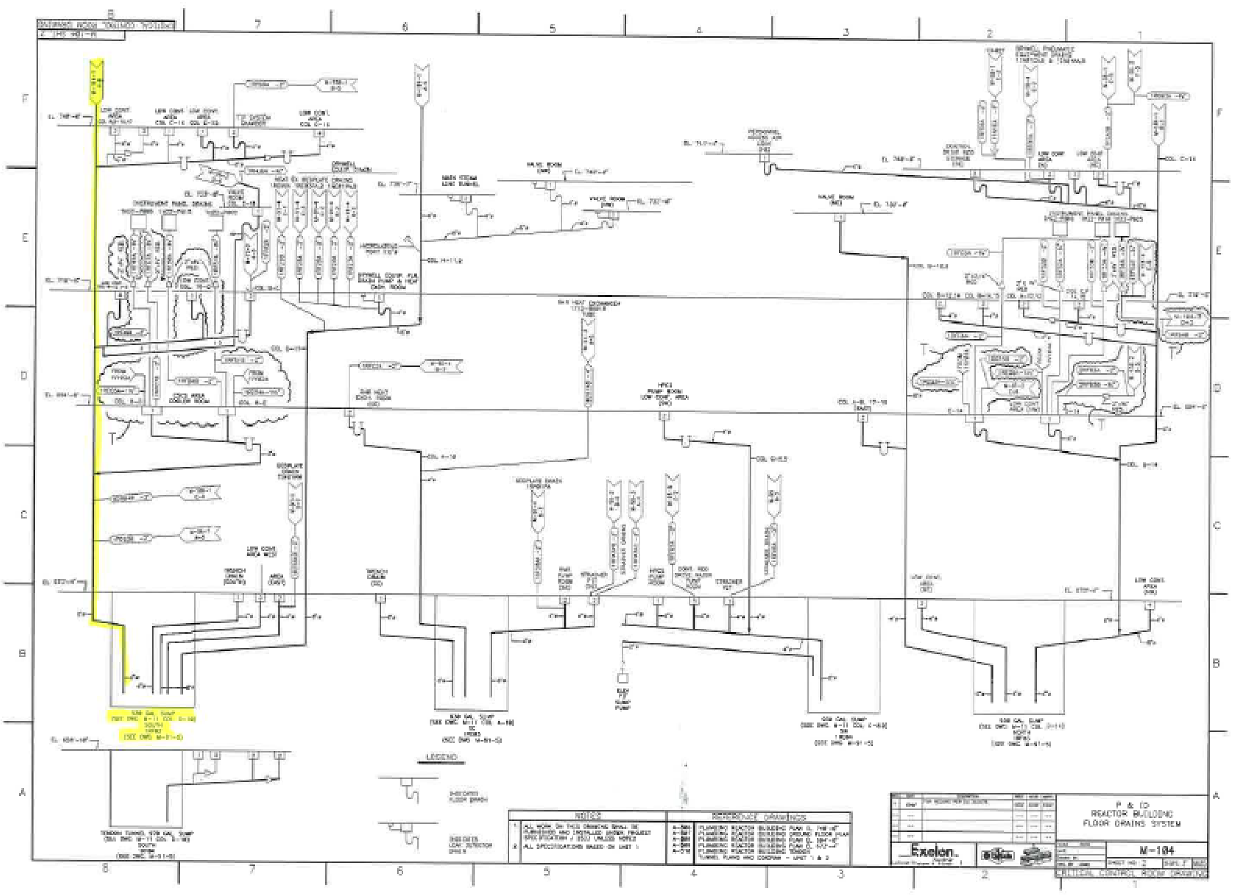
1. STARTS a second South Reactor Building Floor Drain Sump Pump 1RF02PA/B on Hi-Hi Level.

Applicable System Drawings




PIPING DESIGN DATA
LEGEND
NOTES
REFERENCE DRAWINGS
P & ID
REACTOR BUILDING
FLOOR DRAINS SYSTEM
Exelon
M-104
CRITICAL CONTROL ROOM DRAWING


P & ID
REACTOR BUILDING
EQUIPMENT DRAINS SYSTEM
M-91
ComEd
CRITICAL CONTROL ROOM DRAWING



NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. This question was modified from the 2013 NRC exam at the same facility. The NRC disagreed with the facility position regarding this question. The stem of the question asks specifically about where one particular flow switch causes an alarm; the applicable LOR procedure indicates that this alarm occurs on the 601 panel. The question is clear in asking where the flow switch causes the alarm and not what alarms could occur with high leakage. Based upon these considerations, the NRC concluded that there should be no change to key for RO question #58.

SRO Question 86 (Post-Exam Comment #6)

Original Question:

Unit 1 is operating at 100% power.
Unit 2 is in a refueling outage, and fuel shuffles are in progress.

There is an inadvertent Unit 1 Reactor Building Ventilation isolation and SBGT initiation.

The assist NSO reports that 1VG001, Inlet Isolation Damper, opened, reclosed and CANNOT be manually reopened.

The Unit 2 Supervisor will direct (1)_______, and Tech Spec 3.6.4.3 requires entry into an LCO to (2)_________.

A. (1) Unit 1 SBGT train to be secured
- restore SBGT to operable status, ONLY

B. (1) Unit 1 SBGT train to be secured
- restore SBGT to operable status, AND immediately suspend core alterations and movement of irradiated fuel

C. (1) Unit 2 SBGT train to be secured
- restore SBGT to operable status, ONLY

D. (1) Unit 2 SBGT train to be secured
- restore SBGT to operable status, AND immediately suspend core alterations and movement of irradiated fuel

Answer: A

Applicant Feedback:

With the VG train initiated inadvertently it is inoperable and therefore LCO 3.6.4.3 Required Actions E.1, E.2, and E.3 are required to be entered. The applicant believes that 'B' is the correct answer.

Facility Response:

The stem provides a scenario in which a Unit 1 Reactor Building Ventilation isolation signal occurs, which also causes a SBGT system initiation. Secondary Containment Isolation Instrumentation is required per Tech Spec 3.3.6.2. Per LCO 3.0.6, when a supported system LCO is not met solely due to a support system LCO not being met, the conditions and required actions associated with the supported system are not required to be entered. Only the support system required LCO actions are required to be entered. The question is focused on SBGT train status and LCO 3.6.4.3 based on the given plant conditions.

STATION RECOMMENDATION: ACCEPT ONLY (A) as the correct answer

References:

Technical Specification and Bases 3.6.4.3

These references state, in part, that:

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. It was determined that the applicable technical specifications and bases clearly support the keyed answer of the question. Furthermore, it was determined that the technical specification issue raised by the applicant does not form a technically valid basis for the selection of a different correct answer. Based upon these considerations, the NRC concluded that there should be no change to key for SRO question #86.

SRO Question 94 (Post-Exam Comment #7)

Original Question:

Core Alterations have been stopped by the Refuel SRO due to a Refuel Bridge equipment failure.

Who has the authority to grant permission to resume fuel movement?

A. Dedicated Refueling NSO

B. Operations Shift Manager

C. Qualified Nuclear Engineer

D. Outage Services Director

Answer: B

Applicant Feedback:

Fuel moves require an NSO to give permission. The applicant believes that both 'A' an 'B' are correct answers.

Facility Response:

The question provides a scenario where fuel moves were stopped due to a refueling bridge equipment failure. The Shift Manager must grant permission to resume fuel movements following refueling equipment repairs and the NSO must give the Fuel Handling personnel permission to proceed, as stated in LFP-100-1 "An NSO assigned to CORE ALTERATIONS shall CHECK the Step for correctness. If it is in agreement with the NCTL (Nuclear Component Transfer List) then the NSO gives the Fuel Handling personnel permission to proceed. The stem of the question does not specifically direct who is being granted permission to resume fuel moves, but merely states who has the authority to grant permission to resume fuel movements. In this case the Operations Shift Manager would provide the NSO permission to resume fuel movements and the NSO gives the Fuel Handling personnel permission to proceed with fuel movements.

STATION RECOMMENDATION: ACCEPT (A) AND (B) as the correct answer

References:

LFP-100-1, Master Refuel Procedure, Revision 61

This reference states in section D.3 that:

D.3	In the event that a situation arises such that normal Refuel Bridge operation is impaired due to equipment failure or plant conditions, the Fuel Handling Supervisor has the responsibility to PLACE the fuel in a 'Safe Condition'. The Fuel Handling Supervisor shall INFORM the Shift Manager of intended actions, RECEIVE acknowledgment, and use the best available methods and resources at the time.
D.3.1	IMPLEMENT Administrative Controls to: • Require SRO/LSRO to immediately INFORM the Control Room of unexpected/unplanned Fuel Handling issues, AND • Operations Shift Manager permission shall be GRANTED prior to resuming fuel movement. (Ref. G.19)

This reference states in section E.3.8 that:

E.3.8	**An NSO assigned to CORE ALTERATIONS shall CHECK the Step for correctness. IF it is in agreement with NCTL, THEN the NSO gives the Fuel Handling personnel permission to proceed, per Attachment A.**

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. During the administration of the written exam, in response to an applicant question, additional stem information was provided that repairs had been completed. It was determined that the Stem wording of the question and LFP-100-1 support a possible interpretation that makes distractor 'A' also a correct answer. Based upon these considerations, the NRC concluded that both 'A' and 'B' should be accepted as correct answers for SRO question #94.

SRO Question 81 (Post-Exam Comment #8)

Original Question:

Unit 1 is in Mode 5, and irradiated fuel assemblies are being moved within the Reactor Pressure Vessel (RPV).

In accordance with LCO 3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel,

RPV cavity water level must be greater than or equal to (1) above the RPV flange; otherwise, movement of irradiated fuel assemblies in the RPV must be IMMEDIATELY SUSPENDED to maintain sufficient water level to (2) .

A. (1) 22 feet
(2) ensure 99.5% of the total iodine released from a damaged fuel assembly is retained in the water

B. (1) 23 feet
(2) ensure 99.5% of the total iodine released from a damaged fuel assembly is retained in the water

C. (1) 23 feet
(2) retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within limits

D. (1) 22 feet
(2) retain iodine fission product activity in the event of a fuel handling accident, keeping offsite doses within limits

Answer: D

Applicant Feedback:

The answers for both (A) and (D) are included in the Tech Spec Basis for Tech Spec 3.9.6.

Facility Response:

The Tech Spec Bases for 3.9.6 discusses that the LCO is based on a minimum water level of 22 feet. The analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits. The bases also includes a discussion that 23 feet of water allows a decontamination factor of 200 to be used in accident analysis for iodine which relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the fuel. The damaged fuel assembly rods are retained in the refueling cavity water. The discussion in the bases is supporting information and not directly related to the LCO statement.

STATION RECOMMENDATION: ACCEPT ONLY (D) as the correct answer

References:

Technical Specification 3.9.6 and Bases

These references state in part that:

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level—Irradiated Fuel

LCO 3.9.6 RPV water level shall be ≥ 22 ft above the top of the RPV flange.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RPV water level not within limit.	A.1 Suspend movement of irradiated fuel assemblies within the RPV.	Immediately

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel storage pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to 10 CFR 50.67 limits, as modified by the guidance of Reference 1.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.183 (Ref. 1). A minimum water level of 23 ft allows a decontamination factor of 200 to be used in the accident analysis for iodine (Ref. 1). This relates to the assumption that 99.5% of the total iodine released from the pellet to cladding gap of all the damaged fuel assembly rods is retained by the refueling cavity water.

NRC Final Resolution:

The NRC reviewed the aforementioned material related to this question. The NRC agreed with the facility position concerning how this question should be dispositioned. It was determined that although the Technical Specification Bases do contain the 99.5% criteria, it is associated with the 23' level value, and not the 22' level value. Additionally, while the 22' level part of the correct answer is RO knowledge, all second part answers consist of SRO-only knowledge. Based upon these considerations, the NRC concluded that no key change should be made for SRO question #81.

Letter to Bryan C. Hanson to Robert J. Orlikowski dated December 22, 2016

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 – NRC INITIAL LICENSE EXAMINATION REPORT 05000373/2016301; 05000374/2016301

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