August 5, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer and
Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville IL 60555

SUBJECT: BYRON STATION, UNITS 1 AND 2
NRC INITIAL LICENSE EXAMINATION REPORT 05000454/2008301(DRS); 05000455/2008301(DRS)

Dear Mr. Pardee:

On July 2, 2008, Nuclear Regulatory Commission (NRC) examiners completed the initial operator licensing examination process for an examination administered at your Byron Station, Units 1 and 2. The enclosed report documents the results of the examination which were discussed during a debrief on May 29, 2008, with Mr. B. Adams, Mr. M. Prospero, Ms. E. Bogue, and other members of your staff. An exit meeting was conducted by telephone on July 1, 2008, between Ms. E. Bogue, Training Manager, of your staff and Mr. M. Bielby, NRC Chief Examiner, to review the proposed final grading of the written examination for the license applicants. During the telephone conversation NRC resolutions to post examination comments initially received by the NRC on June 11, 2008, and supplemental information eventually received by letter on July 2, 2008, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of May 19 and 26, 2008. The written examination was administered by NRC examiners on May 30, 2008. Five (5) Senior Reactor Operator (SRO) and seven (7) Reactor Operator (RO) applicants were administered license examinations. The results of the examination were finalized on July 9, 2008. All twelve (12) applicants passed all sections of their respective examinations and were issued applicable operator licenses.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), 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/

Hironori Peterson, Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50-454; 50-455
License No. NPF-37; NPF-66

Enclosures:
1. Operator Licensing Examination Report 05000454/2008301(DRS); 05000455/2008301(DRS)
2. Simulation Facility Report
3. Post Examination Comments and Resolutions
4. Written Examinations and Answer Keys (RO and SRO)

cc w/encls 1 & 2:
Site Vice President - Byron Station
Plant Manager - Byron Station
Regulatory Assurance Manager - Byron Station
Chief Operating Officer and Senior Vice President
Senior Vice President - Midwest Operations
Senior Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Director - Licensing and Regulatory Affairs
Manager Licensing - Braidwood, Byron, and LaSalle
Associate General Counsel
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Assistant Attorney General
Illinois Emergency Management Agency
J. Klinger, State Liaison Officer,
  Illinois Emergency Management Agency
P. Schmidt, State Liaison Officer, State of Wisconsin
Chairman, Illinois Commerce Commission
B. Quigley, Byron Station

cc w/encls 1, 2, 3, & 4: E. Bogue Training Manager

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B. Quigley, Byron Station

cc w/encls 1, 2, 3, & 4: E. Bogue Training Manager

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SUBJECT: BYRON STATION, UNITS 1 AND 2
NRC INITIAL LICENSE EXAMINATION REPORT 05000454/2008301(DRS);
05000455/2008301(DRS)

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REGION III

Docket No.	50-454; 50-455
License No.	NPF-37; NPF-66
Report No:	05000454/2008301(DRS); 05000455/2008301(DRS);
Licensee:	Exelon Generation Company, LLC
Facility:	Byron Station, Units 1 and 2
Location:	Byron, IL
Dates:	May 19 – July 2, 2008
Examiners:	M. Bielby, Chief Examiner C. Moore, Examiner, Chief Examiner in Training C. Zoia, Examiner R. Daley, Examiner in Training
Approved by:	Hironori Peterson, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

ER 05000454/2008301(DRS); 05000455/2008301(DRS); 05/19/08 - 07/02/08; Exelon Generation Company, LLC, Byron Station, Units 1 and 2; Initial License Examination Report.

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

Examination Summary:

- Twelve initial license examinations were administered (five senior reactor operator (SRO) and seven reactor operator (RO).
- Twelve applicants passed all sections of their examinations resulting in the issuance of five SRO and seven RO licenses.

POST EXAMINATION COMMENTS AND RESOLUTIONS

REPORT DETAILS

4. OTHER ACTIVITIES (OA)

4OA5 Other

.1 Initial Licensing Examinations

a. Examination Scope

The NRC examiners conducted an announced operator licensing initial examination during the weeks of May 19 and 26, 2008. The NRC examiners used the guidance prescribed in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 9, to prepare the outline and develop the written examination and operating test. The examiners administered the operating test, consisting of job performance measures and dynamic simulator scenarios, during the period of May 20 through 29, 2008. The examiners administered the written examination on May 30, 2008. Five senior reactor operator and seven reactor operator applicants were examined. During the on-site validation week of April 28, 2008, the examiners audited two license applications for accuracy.

b. Findings

Written Examination

The NRC examiners developed the written examination. Written examination changes agreed upon between the NRC and the licensee were made according to NUREG-1021, Revision 9. Subsequent to administration, the NRC graded the written examination and conducted a review of each question to determine the accuracy and validity of the examination questions. The licensee submitted thirteen post-examination question comments by letter dated June 6, 2008, and received by NRC on June 11, 2008. The recommendations included deletion of four questions, changing the correct answer to one question, and accepting two correct answers for another question, as well as enhancements to seven additional questions. Based on NRC review of the recommendations, the licensee submitted revisions to comments by a second letter on July 2, 2008. The contents of the letter were discussed with the licensee in advance during a pre-scheduled exit meeting by telephone on July 1, 2008. The NRC completed final grading of the written examination on July 2, 2008, after receipt of the licensee's revised post-examination comments. The results of the NRC's review of the station's comments are documented in Enclosure 3, Post Examination Comments and Resolutions.

Operating Test

The NRC examiners developed the Operating Test. Operating Test changes agreed upon between the NRC and licensee were made according to NUREG-1021, Operator Licensing Examination Standards for Power Reactors. The licensee submitted no post-examination comments on the Operating Test. The NRC examiners completed operating test grading on July 2, 2008.

Examination Results

Twelve applicants passed all sections of their examinations resulting in the issuance of five senior reactor operator and seven reactor operator licenses.

.2 Examination Security

a. Inspection Scope

The NRC examiners briefed the facility contact on the NRC's requirements and guidelines related to examination physical security (e.g., access restrictions and simulator considerations) and integrity in accordance with 10 CFR 55.49, "Integrity of Examinations and Tests," and NUREG-1021, "Operator Licensing Examination Standard for Power Reactors." The examiners reviewed and observed the licensee's implementation and controls of examination security and integrity measures (e.g., security agreements) throughout the examination process.

b. Findings

No findings of significance were identified.

4OA6 Meetings

Exit Meeting

The chief examiner presented the examination team's preliminary observations and findings with Mr. B. Adams and other members of the licensee management and staff on May 29, 2008. An exit via teleconference was held on July 1, 2008, with Mr. M. Prospero, Ms. E. Bogue and other members of the licensee staff following receipt of the site post-examination comments. The inspectors stated that they had reviewed proprietary information during the preparation and administration of the examination, but that the proprietary information would not be included in the examination report. The licensee acknowledged the observations provided.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
B. Adams, Plant Manager
E. Bogue, Training Director
S. Deprest, Operations Training Program Specialist
M. Prospero, Operations Director
G. Smith, Initial License Examination Lead
R. Williams, Operations Training Manager

NRC
M. Bielby, Chief Examiner
C. Moore, Examiner / Chief Examiner in Training
C. Zoia, Examiner
R. Daley, Examiner in Training
B. Bartlett, Byron Senior Resident Inspector
R. Ng, Byron Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None.

LIST OF ACRONYMS

ADAMS	Agency-Wide Document Access and Management System
CFR	Code of Federal Regulations
CR	Condition Report
DRS	Division of Reactor Safety
ILT	Initial License Training
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records System
RO	Reactor Operator
SCR	Silicon Controlled Rectifier
SDP	Significance Determination Process
SRO	Senior Reactor Operator
SWO	Simulator Work Order

SIMULATION FACILITY REPORT

Facility Licensee: Byron Station Units 1 and 2

Facility Licensee Docket No. 50-454; 50-455

Operating Tests Administered: May 19 - 30, 2008

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
1	During a scenario an I/O bucket failed on the simulator. The failure resulted in numerous spurious indications. Scenario was halted; card replaced, bucket reset, and scenario re-started from the failure point. Training Action Request written (AR 00777711)

Enclosure 2

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 001 (1.00)

The following Unit 1 plant conditions exist:

- Unit 1 has experienced a reactor trip and SI.
- Containment pressure is 27 psig.
- RCS pressure is 300 psig.
- Seven of Eight SX Cooling Tower Fans are running in High Speed.
- 0A fan will NOT start in High Speed.

Which ONE of the following actions is required per 1BEP-0, Reactor Trip or Safety Injection, when aligning the SX Cooling Towers?

a. OPEN all EIGHT riser valves.

b. Restart 0A fan in Low Speed.

c. CLOSE all FOUR Hot Water Basin Bypass valves.

d. Ensure that ONLY the bypass valve associated with the non-running fan is CLOSED.

Answer: c.

Reference: 1BEP-0, Reactor Trip or Safety Injection; I1-EP-XL-01, 1BEP-0, Reactor Trip or Safety Injection

Applicant Comment:
The stem of this question was confusing, and required clarifying. Choice “a” is performed at 1BEP-0 step 14.g.1), but later the riser valve for 0A fan is closed when it won’t start. Recommend change to wording of question stem as follows for future use: Which of the following actions is required to be completed in 1BEP-0, Reactor Trip or Safety Injection, when aligning the SX Cooling Towers? No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with applicants’ recommended change.

NRC Resolution:
Recommendation not accepted. The recommendation appears to be based on personal preference and not supported by plant procedure or management expectation document. The style of language being used does not significantly alter the question being asked. Based on the pre-examination review by licensee training staff, licensed operators and NRC examiners, the question asked was clear. The recommended change would not change the question asked. Therefore, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 002 (1.00)

Unit 1 was at 100% power with all systems normally aligned when annunciator 1-12-B2, PZR PORV OR SAF VLV OPEN, alarms. The following indications are current:

- Actual PZR pressure is 2100 psig and lowering
- Channel 1PT-455 indicates 2500 psig
- PZR level is 62% and rising
- PRT temperature, pressure and level are rising
- All PZR Safety Valve indicator lights are GREEN

Action(s) to mitigate this transient is/are to . . .

a. close the PZR PORV block valve(s) for affected PORV(s).

b. manually trip the reactor and actuate SI.

c. verify insertion of control rods at 48 steps per minute.

d. manually trip the reactor, but DO NOT manually actuate SI.

Answer: a.

Reference: Horse Notes RY-2, PZR Pressure Control, Rev. 2; RY-1, Pressurizer, Revision 3; Lesson Plan, Pressurizer (RY), Revision 6, Attachment B; BAR 1-12-B2, PZR PORV OR SAF VLV OPEN, Revision 4; 1BOA INST-2, Revision 103

Applicant Comment:
Stem of question provided enough information to determine the correct answer for the situation. However, additional actions would be required for a complete answer since the spray valves are open. Recommend adding “Pressurizer spray valves have been manually closed” to the stem for future use. No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with applicants’ recommended change.

NRC Resolution:
Recommendation not accepted. The question being asked is clear and does not require enhancement. The recommendation would not change what is being asked and answered. As stated in the comment, the question incorporates sufficient information to determine the correct answer. The recommendation would not alter the question being asked. Therefore, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 013 (1.00)

Previously, 125 VDC Bus 211 was crosstied to Bus 111 due to equipment problems with Bus 111 Battery and Charger. Bus 111 Battery and Charger are Out-of-Service.

Presently,

- U-1 is in MODE 3.
- U-2 is in MODE 1.

Bus 111 conditions are:

- Crosstie loading due to the loading on Bus 111 is 183 Amps.
- Voltage on Bus 111 is 121 VDC.
- Then, a ground of 50 volts is detected on Bus 111.

Based upon the above conditions, which one of the following actions would be CORRECT?

a. Parameters on Bus 111 are normal and within limits. No action is necessary.

b. Enter into BOP, DC-15, DC Ground Isolation, due to an unexpected ground detected on Bus 111.

c. Shed non-essential loads from Bus 111 to lower Amperage to below 180 Amps to meet cross-tie loading restrictions.

d. Disconnect Bus 111 from Bus 211 in accordance with BOP DC-7, 125 VDC ESF Bus Crosstie/Restoration to ensure that the ground does not adversely affect loads on the operating unit.

Answer: a.

Reference: BOP DC-7, 125 VDC ESF Crosstie/Restoration

Applicants' Comment:
Battery 211 terminal voltage is required to be at least 127.6 VDC per 1BOSR 8.6.1-2, Unit Two 125VDC ESF Battery Bank And Charger 211 Operability Weekly Surveillance, and 2BOL 8.6 (TS 3.8.6 LCOAR), Note 5.

Operator rounds for DC bus 111 list a minimum value of 127.6 volts, and maximum value of 140 volts.

Main Control Board alarm responses 1/2-21-E-10 for both Unit 1 and Unit 2 125V DC busses have alarm setpoints of $\leq$ 123V DC.

The stem of the question states that "Voltage on Bus 111 is 121 VDC." This led the applicants to reject choice "a," "Parameters on Bus 111 are normal and within limits. No action is necessary."

Bus 111 voltage is NOT normal and within limits; bus voltage is at least 6.6 VDC too low per the operator rounds and BOSR. This question has no correct answer and should be deleted from the exam.

References: 1BOSR 8.6.1-1/2, Unit One/Two 125VDC ESF Battery Bank and Charger 111/211 Operability Weekly Surveillance; 1/2BOL 8.6 (TS 3.8.6 LCOAR), Note 5; BAR 1/(2)-21-E10, 125V DC PNL 111/113 (211/213) VOLT LOW; Operator rounds printout

Facility Proposed Resolution:
The licensee agrees with the applicants' comments. This question has no correct answer, should be deleted from the exam and the exam grading adjusted accordingly.

NRC Comments:
Upon review of the question, the applicant comment, and additional NRC requested engineering input from the licensee, the recommendation was accepted and the question was deleted from the examination based on no correct answer.

During the pre-examination review, the NRC examiners specifically questioned the licensee about whether or not the 121 VDC stated in the stem was too low of a voltage for the condition described. The licensee stated that the voltage of 121VDC was low but not excessively low. After receipt of the post-examination comment the NRC requested the licensee to have their engineering staff verify the voltage based on the conditions stated in the question stem. Prior to the engineering staff producing either a voltage calculation or profile for this condition, the NRC agreed that additional actions would be (and could be) performed to increase the voltage.

Choice a. was initially determined to be correct; however, based on information provided in the question stem additional actions would be required to correct the deficient condition. Specifically, Technical Specification 3.8.4, DC Sources, requires additional actions to restore the battery and charger to service. If these required actions can not be accomplished within the LCO completion time, the LCO requires a shutdown to mode 5. Therefore, Choice a. is not correct when it states, "no action is necessary". Therefore, no correct answer was provided for this question so it will be deleted from the exam.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 024 (1.00)

Unit 1 is starting up with the following conditions:

- Reactor power is at 7%.
- Due to IR Channel N35 reading a full decade lower than IR Channel N36, Channel N35 has been placed in BYPASS.

While withdrawing rods in Control Bank D, IR Channel N36 fails low and the LOSS OF DETECTOR VOLT light on the N36 drawer is lit.

Which one of the following is a required response for this condition?

a. Immediately trip the reactor and follow required actions in 1BEP-0, Reactor Trip or Safety Injection.

b. Immediately reduce power to less than P-6.

c. Immediately stop control rod withdrawal and suspend any other positive reactivity additions.

d. Continue power ascension to greater than P-10.

Answer: c.

Reference: Horse Notes, NI-3, Intermediate Range, Revision. 3; System Description, Gamma-Metric Source and Intermediate Range Nuclear Instrumentation, Revision 2; 1B0A INST-1, Nuclear Instrumentation Malfunction Unit 1, Attachment B, IR Channel Failure

Applicant Comment:
IR channels N35 and N36 do not have lights labeled “LOSS OF DETECTOR VOLT.” This was removed when the source and intermediate range instruments were modified to Gamma-Metrics instruments.

The stem of this question is technically inaccurate and confusing. Given the conditions, there was no way to answer it, so the question should be deleted from the exam.

If a “Loss of Detector Volt” were to be inferred to mean a Loss of Instrument Power, then the reactor would trip because of the Loss of Channel N36 when not BYPASSED.

References: TS 3.3.1, Reactor Trip System Instrumentation, condition G; I1-NI-XL-01, Gamma-Metric Source and Intermediate Range Nuclear Instrumentation System Lesson Plan, page 31; 1B0A INST-1, Nuclear Instrumentation Malfunction Unit 1, Attachment B, IR Channel Failure

Facility Proposed Resolution:
The licensee agrees with the applicants’ comments. This question is technically inaccurate, should be deleted, and the exam grading should be adjusted accordingly.

POST EXAMINATION COMMENTS AND RESOLUTIONS

NRC Resolution:
Upon review of the question and applicant comment, the recommendation was accepted and the question was deleted from the examination based on technically inaccurate information presented in the question stem. Specifically, the "Loss of Detector Volt" light was removed from the Intermediate Range Nuclear Instrumentation drawer by a modification in 2001. The question was reviewed prior to the examination by the licensee training staff, licensed senior reactor operators and reactor operators and corporate examination specialist all of whom did not identify the deficiency. In addition, there were no questions related to the deficiency asked by applicants during the examination.

The question stem is technically inaccurate because the "Loss of Detector Volt" light no longer exists on the Intermediate Range Nuclear Instrumentation drawer. As a result, no correct answer was provided for the question, so the question will be deleted from the examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 028 (1.00)

Unit 2 start up is in progress with Reactor Power at 16% and all systems normally aligned.

- An electrical transient causes the 2A and 2C RCP Breakers to trip open.
- 2B and 2D RCPs remain running
- The RCP Breaker Position Reactor Trip Circuit malfunctioned and NO Reactor trip occurred.

If NO operator action is taken, what will happen within 2 minutes?

The reactor will . . .

a. NOT automatically trip. RCS overpressure condition will NOT result.

b. NOT automatically trip. Excessive KW/ft condition will NOT result.

c. automatically trip. DNB condition will NOT result.

d. automatically trip. Loss of heat sink condition will result.

Answer: c.

Reference: I1-RC-XL-02, Reactor Coolant Pump

<u>Applicants' Comment</u>:
The portion of the stem that states "NO Reactor trip occurred" was somewhat confusing in context of what happened next. Recommend changing the wording of the stem to clarify it, such as "NO reactor trip occurred due to the failure of the RCP Breaker Position Reactor Trip circuit"; or providing a timeline, with at time 0, no reactor trip occurred.
No change to answers or grading is requested.

<u>Facility Proposed Resolution</u>:
The licensee agrees with the applicants' comments.

<u>NRC Resolution</u>:
Recommendation not accepted. The recommendation is requesting additional information that is not necessary to answer the question. Based on the pre-examination review by licensee training staff, licensed operators and NRC examiners, the question asked was clear. The recommended change would not change the question asked or the possible answers. Therefore, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 034 (1.00)

Unit 1 was initially operating at 100% power when a safety injection occurred. The plant has entered 1BEP-0, Reactor Trip or Safety Injection, to respond to the event. Present Unit 1 conditions are as follows:

- 1A Safety Injection Pump is Out-of-Service
- Containment pressure is 7.3 psig
- 1B SI Pump, 1A and 1B CV Pumps, and 1A and 1B RH Pumps are all running
- All RCPs are running
- RCS pressure is 1620 psig and slowly lowering
- Both PZR PORVs are closed
- RCS Temperature is 541°F and slowly lowering

The crew has learned that the thrust bearing temperature for the 1B SI pump is presently 208°F and rising; therefore, the 1B SI pump was stopped.

While at Step 25 of 1BEP-0, Reactor Trip or Safety Injection, which one of the following actions would be CORRECT in response to the event?

a. Stop RCPs. Stop dumping steam.

b. DO NOT stop RCPs. Establish a maximum cool down rate of 50°F/Hr.

c. DO NOT stop RCPs. Stop dumping steam.

d. DO NOT stop RCPs. Continue to depressurize the RCS by dumping steam to the condenser from intact SGs.

Answer: c.

Reference: 1BEP-0, Reactor Trip or Safety Injection

Applicants' Comment:
It is unclear what has happened to pressure since entering 1BEP-0. Recommend placing information earlier in stem that we are at step 25. Also, add title of step 25, Maintain RCS Temperature Control. No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with the applicants' comments.

NRC Resolution:
Recommendation not accepted. The question stem clearly states that an abnormal cooldown was occurring while performing the steps of 1BEP-0, Response to a Reactor Trip or Safety Injection. The applicants are expected to know the intent of this step is to stop an abnormal cooldown by isolating the most likely sources. The proposed clarification information would make the question a low level of difficulty that would be unacceptable for use on an NRC

examination. Therefore, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 035 (1.00)

Unit 2 is in MODE 4 with a plant cooldown in progress. The following plant conditions exist:

- RCS temperature is 300°F and slowly lowering due to the plant cooldown.
- 2A RH providing shutdown cooling.
- RCS pressure is 310 psig.
- LCO 3.4.12, Low Temperature Overpressure Protection (LTOP) System, is being met, and pressure relief capabilities for LTOP are met by the 2 PZR PORVs.

In these conditions, an inadvertent SI actuation occurred. With NO operator action, what would be the expected plant response? (NOTE: Unit 2 LTOP PORV Setpoint Curve is provided.)

a. One CV pump realigns to its ECCS lineup with the 2A RH suction relief valve being the first relief valve to lift.

b. BOTH CV pumps realign to their ECCS lineup causing pressure in the RCS to rise with the 2A RH suction relief valve being the first relief valve to lift.

c. One CV pump and BOTH SI pumps realign to their ECCS lineup causing pressure in the RCS to rise with the PORVs being the first relief valves to lift.

d. One CV pump realigns to its ECCS lineup with the PORVs being the first relief valves to lift.

Answer: a.

Reference: LCO 3.4.12 and Bases, LTOP System; I1-RH-XL-01, Residual Heat Removal System

Applicant Comment:
Better wording than “pressure relief capabilities for LTOP are met by the 2 PZR PORVs.” would be “Both Przr PORVS are selected to ARMED LOW TEMP.”
No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with the applicants’ comments.

NRC Resolution:
Recommendation not accepted. The proposed recommendation would decrease the question level of difficulty by supplying the applicant knowledge that he is expected to determine from the given information. The recommendation does not indicate a problem with technical accuracy or question clarity, but appears to be more editorial in nature. As a result the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 043 (1.00)

Reactor power is at 100% when the following events occur:

- The main turbine trips.
- The reactor does NOT automatically trip due to a failure of the Turbine Trip circuitry for the Reactor Trip System.

Assuming NO operator action, the reactor will still eventually automatically trip.

What Reactor Trip System Functions will initiate this reactor trip?

1. Overpower delta T.
2. Lo-Lo S/G Level.
3. Overtemperature delta T.
4. Pressurizer Pressure.

a. 1, 2, AND 3 ONLY.

b. 1 AND 4 ONLY.

c. 2 AND 3 ONLY.

d. 3 AND 4 ONLY.

Answer: d.

Reference: Main Steam System, I1-MS-XL-01

Applicant Comment:
Given the conditions stated, the assumption is made that all control systems are in their normal alignments. The plant response to this event is as follows:

After the turbine trips, the steam dumps open fully on the load reject.
RCS temperature rises, and control rods step in to lower Tave.
SG PORVs cycle open.
Feedwater pumps maintain normal feedwater flow to the SGs.
The reactor will trip on OTDT.
The RCS pressure rise is controlled by the Pressurizer PORVs.

This scenario was run on the Byron simulator with all automatic reactor trips defeated to verify that pressure never rose high enough nor dropped low enough to actuate a High or Low Pressure trip. The maximum pressure reached was 2340 psig, and the minimum pressure after 4.5 minutes of run time, was 1990 psig. The only trip setpoint reached was OTDT. OTDT is a component of 3 of the answers, but is not listed by itself.

This question has no correct answer and should be deleted from the exam.

References: Trends from simulator scenario are attached.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Facility Proposed Resolution:
The licensee agrees with the applicants' comments. This question has no correct answer, should be deleted from the exam and the exam grading adjusted accordingly. This appears to be a design basis question, and would be correct if stated this way:

"What Reactor Trip Function(s) is/are DESIGNED to initiate this back-up trip?"

The applicants answered the question based on plant response, as directed by the Appendix E Written Exam Guidelines.

NRC Resolution:
Recommendation accepted. The question asked what reactor trip system functions would initiate the reactor trip for the given conditions. The intent of the question was to ask a design bases question regarding the back-up reactor trips available when the turbine trip failed to actuate a reactor trip. As written the question only asked for the reactor trip that would initiate/generate a reactor trip from an operational point of view. Only one reactor trip will open the reactor trip breakers to initiate a reactor trip. While there may be other reactor trip signals generated only the first one of them will actually open the reactor trip breakers to initiate the reactor trip. As a result, the only correct answer would be Over Temperature Delta Temperature (OTDT) as noted by the applicant comment and the simulator reference. Since all four question choices identify more than one reactor trip, all are incorrect and the question has no correct answer and the question will be deleted from the examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 054 (1.00)

The following conditions exist in Unit 1:

- The Reactor is shut down in Mode 3.
- Containment pressure is 0.7 psig.
- You have made an emergency containment entry to investigate a steam leak, and are presently attempting to exit the containment through the personnel airlock doors.

While attempting to exit, you discover that the interior personnel airlock door will NOT open. Five minutes after mechanically opening the interior equalizing valve, it is discovered that pressure has still NOT equalized across the interior door.

Which of the following could be the reason(s) for this condition? (Consider each condition separately.)

1. The exterior equalizing valve is closed.
2. The exterior equalizing valve is open.
3. Containment pressure is too high to allow the inner airlock door to open.

a. 1 AND 3 ONLY.

b. 2 AND 3 ONLY.

c. 1 ONLY.

d. 2 ONLY.

Answer: d.

Reference: BAP 1450-8, Primary Containment Equipment/Emergency Hatch; Personnel Airlock Doors Operation

<u>Applicant Comment</u>:
The question states there is a steam leak in containment, and that containment pressure is (currently) 0.7 psig. A steam leak inside containment will cause containment pressure to rise. Given the information, it is impossible to determine if pressure is rising faster than the interior equalizing valve can allow airlock pressure to equalize with containment pressure.
The interior airlock door opens inward to containment, and the airlock door is approximately 5' wide by 7' tall. This results in a surface area of 5040 square inches. For the door to be held closed with a force of 100 ft-lbf, a DP of only 0.02 psid is required. Since it is impossible to determine from the information provided whether the interior equalizing valve can equalize faster than an unstated size steam leak can pressurize containment, and that a very small DP is all that is required to hold the door closed, containment pressure COULD (as asked) be too high to allow the inner airlock door to be opened. This results in choice "b" also being correct.

This question has two correct answers, "b" and "d."

POST EXAMINATION COMMENTS AND RESOLUTIONS

References: BAP 1450-8, Primary Containment Equipment/Emergency Hatch; Personnel Airlock Doors Operation

Facility Proposed Resolution:
The licensee agrees with the applicants' comments. This question has two correct answers, and the exam grading should be adjusted accordingly.

NRC Resolution:
Recommendation not accepted. The question stated that personnel were exiting containment after investigating a steam leak and they could not open the interior personnel airlock door after waiting 5 minutes for pressure to equalize. The question asked for a diagnosis of the problem based on the given choices. The applicant comment included an assumption not stated in the question stem. The assumption was that containment pressure was rising due to a steam leak at a rate that exceeded the capacity of the airlock equalizing system. As read to the applicants prior to starting the written examination, Appendix E of NUREG-1021, Operator Licensing Examination Standards for Power Reactors, states that the applicant should not make assumptions regarding conditions that are not specified in the stem of the question unless they occur as a consequence of other conditions that are stated in the question. Stating that a steam leak exists does not necessarily mean that containment pressure is increasing. This would depend on the size and location from which the steam is leaking.

Based on the original post-examination comment submittal, the NRC examiners requested the licensee to verify with their operations personnel whether or not personnel would be sent into containment with a steam leak causing pressure to rise at a rate that exceeded the capacity of the equalizing line. The licensee was also asked to verify with their engineering department what the rate of pressure rise would have to be to exceed the capacity of the containment airlock door equalizing line. The licensee consequently verified that a steam leak rising at a rate that exceeded the capacity of the containment airlock door equalizing line would require a large steam leak (loss of coolant accident) and it would be highly unlikely that operations would be sending personnel into containment to repair such a leak. As a result the licensee subsequently revised and withdrew the comment.

This question was discussed during the pre-examination review with training staff and licensed operators, all of whom agreed with the technical bases for the question, and the correct answer. There were no clarifying questions asked by applicants during the examination. Subsequent to the examination administration, the licensee initially agreed with the applicant comment until the NRC challenged the licensee to provide a more technically rigorous evaluation of the applicants concern. This did not met NRC expectations two ways. The first was that during the initial pre-examination review the licensee determined the question to be technically accurate, then reversed their evaluation based on the applicant comment regarding the question having two correct answers. This indicated the initial review of the question may have been less than thorough. The second expectation not met was the lack of rigor put into the post-examination comments by the facility. The NRC had to ask the licensee to obtain additional analysis information from their operations and engineering departments concerning conditions stated in the question stem that should have been obtained prior to the examination submittal.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Based on review of the applicant comments and subsequent information obtained by the licensee, the only correct answer is d. and the question answer will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 066 (1.00)

Which of the following prints would show the Flow Control Loop for the 0VC03CA Make-Up Fan?

a. 3040 series of prints

b. 3041 series of prints

c. 4030 series of prints

d. 4031 series of prints

Answer: d.

Reference: 0-4031VC04

Applicants' Comment:
Change the question to supply examples of the various drawings and ask to determine what it does. There were no names supplied for the numbers, this is a memory test without any context. No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with the applicants' comments.

NRC Resolution:
Recommendation not accepted. This question was discussed with both training staff and licensed operators during the pre-examination review and found to be acceptable and based on the operations department expectations. It was stated that applicants are expected to know what type or series of prints are available in the control room by series numbers. The recommendation would make the question a low level of difficulty that would be unacceptable for the written examination. The question was identified as Fundamental which is a memory type question. There were no applicant comments based on clarity or technical accuracy during or after the examination administration. As a result, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 076 (1.00)

The following Unit 1 plant conditions exist:

- A LOCA has occurred
- Command and Control has been transferred to the EOF
- The crew has transitioned to 1BFR-C.1, Response to Inadequate Core Cooling
- Containment pressure is 4 psig and stable
- CETC indicate 1250°F and rising
- SG levels are as follows:

1A	1B	1C	1D
0% NR	20% NR	0% NR	15% NR

- RCP #1 seal ΔPs are as follows:

	1A	1B	1C	1D
#1 seal ΔP (psid)	250	125	275	225

The crew is at step 17 in 1BFR-C.1 to check if RCPs should be started. The Unit RO recommends starting ONLY the 1D RCP to provide cooling to the core. Which of the following is the correct response to the RO recommendation?:

a. Direct the RO to start ONLY the 1D RCP.

b. Obtain authorization from the STA to start ONLY the 1D RCP.

c. Direct the RO to start the 1B and 1D RCP.

d. Obtain authorization from the EOF to start all RCPs.

Answer: c.

Reference: 1BFR-C.1, Response to Inadequate Core Cooling; FR-C.1 Background Information for WOG Emergency Response Guideline; BAP 1310-10, Revision 10, HU-AA-104-101, Procedure Use and Adherence; Byron Addendum EP-AA-112-100-F-01, Shift Emergency Director Checklist

Applicant Comment:
WOG background Step Description Table for FR C.1, 29: "To temporarily restore core cooling, the operator is instructed to start RCPs one at a time until CETCs are <1200°F."

Step 17 of 1BFR C.1 directs starting "RCP in any available idle RCS cooling loop," then rechecking CETCs and starting more RCPs as needed until CETCs <1200°F, rechecking CETCs between starts.

RCPs are to be started 1 at a time, so choice "a" is correct, lacking further information in the stem or choice "c" about checking CETCs between RCP starts.

References: I1-XL-FR-02, BFR C series lesson plan; WOG FR-C.1, Background information (HFRC1BG)

Facility Proposed Resolution:
The licensee agrees with the applicants' comments. The correct answer is choice "a," and the exam grading should be adjusted accordingly.

NRC Resolution:
Recommendation accepted. Emergency Procedure 1BFR – C.1, Response to Inadequate Core Cooling, Step 17, directs the operator to start any available reactor coolant pump (RCP) and then check core exit thermocouples (CETCs) to see if they are less than 1200°F before starting additional RCP's. As a result, per 1BFR – C.1 the only correct answer to the question asked is Choice a. because it only starts one pump.

This question was reviewed by the licensee training staff and licensed operators for technical accuracy, but this deficiency was not identified prior to administration of the examination. However, the accepted correct answer has been changed from Choice c. to Choice a.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 089 (1.00)

Unit 2 is operating at 100% power. The 2A DG has been INOPERABLE for 24 hours due to planned maintenance. The Unit Supervisor has just declared the 2B containment spray pump as INOPERABLE due to a motor failure.

AT THIS TIME, and based upon the selections below, what is/are REQUIRED Technical Specification action(s) for this condition? (NOTE: TS LCOs 3.6.6 and 3.8.1 are attached.)

1. Restore containment spray train B to OPERABLE status within 7 days.
2. Enter LCO 3.0.3 Immediately.
3. Be in MODE 3 within 6 hours.

a. 1 ONLY.

b. 1 AND 2 ONLY.

c. 2 ONLY.

d. 3 ONLY

Answer: a.

Reference: TS LCO and Base 3.6.6, Containment Spray and Cooling System; TS LCO 3.8.1, AC Sources - Operating

Applicant Comment:
Suggest including a timeline so that the candidate evaluates what action is to be taken at a specific time. Some confusion as to what was being asked. No change to answers or grading is requested.

Facility Proposed Resolution:
The licensee agrees with the applicants' comments.

NRC Resolution:
Recommendation not accepted. The question was reviewed by licensee training staff, licensed operators and corporate examination specialist all of whom agreed the clarity and technical accuracy were sufficient. In addition, there were no questions by applicants regarding either of these aspects during the examination. The recommendation would not change the question that is being asked and would only provide the same information in a different format. This recommendation appears to be based on personnel preferences and does not significantly alter the question being asked. Therefore, the question will remain unchanged from the as administered examination.

POST EXAMINATION COMMENTS AND RESOLUTIONS

Question: 095 (1.00)

The Station has experienced a large break Loss of Coolant Accident on Unit 2. The Shift Manager has assumed the duties of the Shift Emergency Director and is in Command and Control.

Which of the following is a list of the Shift Emergency Director's Non-delegable responsibilities?

a. Classification of the Emergency
Notification of the Site Vice President
Notification of the State and Federal Agencies
Site Assembly/Accountability

b. Classification of the Emergency
Authorization for Emergency Dose Exposure
Notification of the State and Federal Agencies
Determination of Protective Action Recommendations to the State

c. Classification of the Emergency
Authorization for Emergency Dose Exposure
Site Assembly / Accountability
Determination of Protective Action Recommendations to the State

d. Classification of the Emergency
Notification of the Site Vice President
Notification of State and Federal Agencies
Determination of Protective Actions for Plant Personnel

Answer: b.

Reference: LS-AA-104-1000, 50.59 Resource Manual; LS-AA-128, Regulatory Review of Proposed Changes to the Approved Fire Protection Program

Applicant Comment:
Choice "b" has an incomplete answer. It states one of the Shift Emergency Director's non-delegable responsibilities is "Notification of State and Federal Agencies." This statement implies the act of using the NARS phone to make notifications. In fact, the Shift Emergency Director approves the NARS and ENS forms used for the notifications. The actual notification is done by a designated communicator. This distinction led the applicants to reject choice "b" as a possible correct answer.

The list of non-delegable duties, according to EP-AA-1000, Standardized Radiological Emergency Plan, includes this statement: "Notification of offsite authorities (approval of state/local and NRC notifications)." This question has no correct answer and should be deleted from the exam.

References: EP-AA-1000, Standardized Radiological Emergency Plan

POST EXAMINATION COMMENTS AND RESOLUTIONS

Facility comment:
The licensee agrees with the applicants' comments. This question has no correct answer, should be deleted from the exam and the exam grading adjusted accordingly.

NRC Resolution:
Recommendation not accepted. Emergency Plan, EP-AA-1000, Standardized Radiological Emergency Plan, states that the Emergency Director has the non-delegable responsibility of "Notification of offsite authorities" (state/local and NRC notifications). The identified correct answer says that the Emergency Director's non-delegable responsibilities are, "Notification of the State and Federal Agencies." The shift Emergency Director may not make the actual notification to state and federal agencies; however, he is the station representative responsible for the information contained within the communication, and is also responsible for ensuring that it is transmitted within the required time frame. This question was reviewed during the pre-examination review and found to be acceptable by the licensee training staff, licensed operators and corporate examination specialist.

The licensee response to this post examination comment does not meet the NRC expectation for post examination comments. It is the NRC's position that the licensee performs a complete review and analysis of the applicants post examination comments prior to submitting the facilities recommendations regarding comments to the NRC. The licensee's agreement with this post examination comment does not reflect a complete review and analysis of the comment was performed because EP-AA-1000, Standardized Radiological Emergency Plan, clearly states that one of the Emergency Directors non-delegable Responsibilities is to ensure that State and Federal Agencies are notified of the Emergency situation occurring at the Plant.

The question and answer will remain unchanged from the as administered examination.

WRITTEN EXAMINATIONS AND ANSWER KEYS (RO/SRO)

RO/SRO Initial Examination ADAMS Accession #ML082130329.