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

August 12, 2019

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

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2—NRC INITIAL LICENSE EXAMINATION REPORT 05000456/2019301 AND 05000457/2019301

Dear Mr. Hanson:

On June 28, 2019, the U.S. Nuclear Regulatory Commission (NRC) completed the initial operator licensing examination process for license applicants employed at your Braidwood Station. The enclosed report documents the results of those examinations. Preliminary observations noted during the examination process were discussed on June 19, 2019, with Ms. M. Marchionda-Palmer, Site Vice President, and other members of your staff. An exit meeting was conducted by telephone on July 3, 2019, with Ms. M. Marchionda-Palmer, other members of your staff, and Mr. J. Seymour, Operations Engineer, to review the final grading of the written examination for the license applicants. During the telephone conversation, NRC resolutions of the station's post-examination comments, received by the NRC on June 28, 2019, were discussed.

The NRC examiners administered an initial license examination operating test during the weeks of June 10 and June 17, 2019. The written examination was administered by Braidwood Station training department personnel on June 19, 2019. Eight Senior Reactor Operator applicants were administered license examinations. The results of the examinations were finalized on July 16, 2019. One applicant failed one or more sections of the administered examination and was issued a preliminary results letter. Seven applicants passed all sections of their respective examinations and were issued senior operator licenses.

The administered written examination and operating test, as well as documents related to the development and review (outlines, review comments and resolution, etc.) of the examination will be withheld from public disclosure until June 19, 2021.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations* 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Rhex A. Edwards, III, Acting Chief
Operations Branch
Division of Reactor Safety

Docket Nos. 50–456; 50–457
License Nos. NPF–72; NPF–77

Enclosures:
1. OL Examination Report 05000456/2019301;
   05000457/2019301
2. Post-Examination Comment, Evaluation,
   and Resolution
3. Simulation Facility Fidelity Report

cc: Distribution via LISTSERV®
F. Jordan, Training Director,
Braidwood Station

Letter to Bryan C. Hanson from Rhex A. Edwards, III, dated August 12, 2019.

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2—NRC INITIAL LICENSE EXAMINATION REPORT 05000456/2019301 AND 05000457/2019301

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REGION III

Docket Nos:	05000456; 05000457
License Nos:	NPF–72; NPF–77
Report No:	05000456/2019301; 05000457/2019301
Enterprise Identifier:	L-2018-OLL-0004
Licensee:	Exelon Generation Company, LLC
Facility:	Braidwood Station, Units 1 and 2
Location:	Braceville, IL
Dates:	June 10, 2019, through June 28, 2019
Examiners:	J. Seymour, Operations Engineer, Chief Examiner C. Zoia, Senior Operations Engineer, Examiner E. Cushing, Reactor Engineer, Examiner
Approved By:	R. Edwards, III, Acting Chief Operations Branch Division of Reactor Safety

SUMMARY

Examination Report 05000456/2019301; 05000457/2019301; 06/10/2019-06/28/2019; Exelon Generation Company, LLC; Braidwood Station, Units 1 and 2; Initial License Examination Report.

The announced initial operator licensing examination was conducted by regional U.S. Nuclear Regulatory Commission examiners in accordance with the guidance of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11.

Examination Summary

Seven of eight applicants passed all sections of their respective examinations. Seven applicants were issued senior operator licenses. One applicant failed one or more sections of the administered examination and was issued a preliminary results letter. (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 11, to develop, validate, administer, and grade the written examination and operating test. The written examination outlines were prepared by the NRC staff and were transmitted to the facility licensee's staff. Members of the facility licensee's staff prepared the operating test outlines and developed the written examination and operating test. The NRC examiners validated the proposed examination during the week of May 6, 2019, 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 and dynamic simulator scenarios, during the period of June 10 through June 17, 2019. The facility licensee administered the written examination on June 19, 2019.

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.

During the validation of the written examination, several questions were modified or replaced. All changes made to the written examination were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and were documented on Form ES-401-9, "Written Examination Review Worksheet." 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) on June 19, 2021, (ADAMS Accession Numbers ML17214A825, ML17214A828, ML17214A830, and ML17214A832, respectively).

On June 28, 2019, the licensee submitted documentation noting that there were six 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 documented in Enclosure 2 to this report.

The NRC examiners graded the written examination on July 2, 2019, 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.

Following the review and validation of the operating test, minor modifications were made to several Job Performance Measures, and some minor modifications were made to the dynamic simulator scenarios. All changes made to the operating test were made in accordance with NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," and were documented on Form ES-301-7, "Operating Test Review Worksheet." The Form ES-301-7, the operating test outlines (ES-301-1, ES-301-2, and ES-D-1s), and both the proposed and final operating tests, will be available electronically in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS on June 19, 2021 (ADAMS Accession Numbers ML17214A825, ML17214A828, ML17214A830, and ML17214A832, respectively).

The NRC examiners completed operating test grading on July 16, 2019.

(3) Examination Results

Eight applicants at the senior reactor operator level were administered written examinations and operating tests. The results of the examinations were finalized on July 16, 2019. Seven applicants passed all portions of their examinations and were issued their respective operating licenses on July 16, 2019. One applicant failed one or more sections of the administered examination and was issued a preliminary results letter.

.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

None.

4OA6 Management Meetings

.1 Debrief

The chief examiner presented the examination team's preliminary observations and findings on June 18, 2019, to Ms. M. Marchionda-Palmer, Site Vice President, and other members of the Braidwood Station staff.

.2 Exit Meeting

The chief examiner conducted an exit meeting on July 3, 2019, with Ms. M. Marchionda-Palmer, Site Vice President, and other members of the Braidwood Station staff, by telephone. The examiners asked the licensee whether any of the material used to develop or administer the examination should be considered proprietary. Proprietary or sensitive information identified during the examination or debrief/exit meetings will be handled in accordance with the applicable requirements.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Marchionda-Palmer, Site Vice President
J. Keenan, Plant Manager
F. Jordan, Training Director
P. Moodie, Operations Director
M. Spillie, Acting Regulatory Assurance Manager
K. Lueshen, Operations Service Manager
J. Petty, Shift Operations Superintendent
J. Beard, Operations Training
D. Brunswick, Operations Training
J. Taff, Operations Training Manager
R. Schliessmann, Regulatory Assurance
R. Witcofski, Operations

U.S Nuclear Regulatory Commission

R. Baker, Branch Chief (Acting)
R. Edwards, Branch Chief (Acting)
D. Kimble, Senior Resident Inspector
J. Seymour, Operations Engineer, Chief Examiner
C. Zoia, Senior Operations Engineer, Examiner
E. Cushing, Reactor Engineer, Examiner

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened, Closed, and Discussed

None

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access and Management System
NRC	U.S. Nuclear Regulatory Commission

<u>Question #6</u>

Unit 2 is in MODE 5.

2BwGP 100-1, PLANT HEATUP, is in progress.

- RCS temperature is 100°F.
- Pressurizer Level is 50 percent.
- The 2A RH train is in shutdown cooling.
- The 2B RH train is in STANDBY.
- The 2A RH pump amps are fluctuating between 50 to 60 amps.
- The 2A RH pump flow is fluctuating between 4500 to 5000 gpm.

The crew will (1)__, and (2)__ to correct the issue.

A. (1) continue in 2BwGP 100-1, PLANT HEATUP, ONLY
(2) IMMEDIATELY trip the 2A RH pump to prevent damage

B. (1) continue in 2BwGP 100-1, PLANT HEATUP, ONLY
(2) take manual control of 2RH618, HX 1A BYP FLOW CONT VLV, to reduce flow

C. (1) enter 2BwOA PRI-10, LOSS OF RH COOLING UNIT 2
(2) IMMEDIATELY trip the 2A RH pump to prevent damage

D. (1) enter 2BwOA PRI-10, LOSS OF RH COOLING UNIT 2
(2) take manual control of 2RH618, HX 1A BYP FLOW CONT VLV, to reduce flow

<u>Answer</u> D

<u>Answer Explanation</u>

A – Plausible: continue in 2BwGP 100-1 and immediately trip the 2A RH pump are incorrect. The 2BwGP 100-1 directs securing all RH trains during the startup. A novice applicant may interpret this procedural flow path as adequate to address the RH pump current issue. The 2A RH pump amps are fluctuating near the red band. This would be correct if the 2A RH pump flow was reduced and did not stabilize parameters.

B – Plausible: continue in 2BwGP 100-1 is incorrect and Take manual control of 2RH618 to reduce flow is correct. The 2BwGP 100-1 directs securing all RH trains during the startup. A novice applicant may interpret this procedural flow path as adequate to address the RH pump current issue.

C – Plausible: 2BwOA PRI-10 is correct, immediately trip the 2A RH pump is incorrect. This would be correct if the 2A RH pump flow was reduced and did not stabilize parameters.

D – Correct: 2BwOA PRI-10 and Take manual control of 2RH618 to reduce flow are correct. Per 2BwOA PRI-10 the first mitigative strategy performed is to reduce RH pump flow. The crew should attempt to stabilize RH system operation prior to tripping the running RH pump and continuing to further mitigating actions.

Technical Reference and Revision #

2BwOA PRI-10, Revision 107, Page 2.
_BwOA PRI-10 Lesson Plan (I1-OA-XL-20) Revision 13, Page 2.

Applicant Comment

Q#6 Entry conditions for 2BwOA PRI-10 state "may" for entry criteria regarding fluctuating RH pump amps. This infers that entry is not required for the conditions given in the stem which result in excess RH cooling vice a "loss of RH cooling". Per BwOP RH-6, "immediately reduce RH flow if RH pump amps are >50." Per BwAP 340-1 "prudent operator actions to place the plant in a stable condition may be performed from memory." This creates a situation where B would be a second correct answer.

Facility Position on Applicant Comment

Q#6 – Accept B as a second correct answer. BwOP RH-6, PLACING THE RH SYSTEM IN SHUTDOWN COOLING, precaution D.4 states "during normal operations, do not exceed RH pump running current of 50.0 amps. If 50.0 amps is exceeded immediately reduce RH flow to lower current and contact engineering." The 2A RH train was running as given in the stem in shutdown cooling mode in the stem. Therefore, the open-ended precaution from BwOP RH-6, could be applied and RH system flow reduced without entry into 2BwOA PRI-10. Additionally, per BwAP 340-1, Use of Procedures for operating department, C.3.a.2 (page 6), "during transients, actions required to place the plant in a stable condition may be performed from memory unless a prompt response procedure exists." No prompt response procedure exists for the conditions in the stem and the action taken in answer B would stabilize the plant.

Additional Information on Position Provided by Facility on July 2, 2019

[T]he expectation would be to attempt to lower RH system flow without exiting 2BwGP 100-1 first, since the pump is experiencing potential run out conditions. If that was successful, no further procedure transition would take place. If not successful, the crew should enter 2BwOA PRI-10. Entry conditions for 2BwOA PRI-10 are phrased as a “may” for fluctuating RH pump amps, if the condition were corrected by adjusting RH system flow then BwOA entry would not be needed.

The stem did not state “actions taken are not successful”. This creates an unstated assumption that the action being taken could correct the issue and therefore no further procedure transitions are required. BwOP RH-6 would have been completed prior to where the stem begins the initial conditions for the question. Therefore, the applicant could utilize the precaution (D.4) from memory (per BwAP 340-1) to lower RH system flow and stabilize the plant preventing damage to the RH pump, without transitioning to another procedure. The conditions of the stem are consistent with an excess of RH cooling (RH pump flow fluctuating between 4500-5000 gpm), prudent operator action to correct this issue should be utilized to prevent damage or a trip of the 2A RH pump resulting in a loss of RH cooling.

Additional References

Excerpt from 2BwGP 100-1, Plant Heatup, Revision 39

F. 23. RH alignment for HEATUP:

a. VERIFY approximately 3300 GPM flow per operating RH Loop.

b. THROTTLE the following as necessary, to maintain the desired RCS Heatup Rate:

- ○ 2RH606
- ○ 2RH606

Excerpt from BwOP RH-6, Placing the RH System in Shutdown Cooling, Revision 59

D. PRECAUTIONS

1. If the RCS is solid, and the RH System is being used for Letdown and/or temperature control, place CV131 in MANUAL, prior to starting an RH Pump.

2. With the RCS solid, Monitor RCS pressure when opening the RCS Hot Leg(s) to RH Suction Valves.

3. Monitor CC System Pressure and the CC Pumps, when CC Flow is initiated to the RH Heat Exchanger. It may be necessary to start an additional CC Pump, prior to initiating flow to an RH Heat Exchanger.

4. During normal operations, do not exceed RH Pump running current of 50.0 amps. If 50.0 amps is exceeded immediately reduce RH flow to lower current and contact Engineering.

Excerpts from 2BwOA PRI-10, Loss of RH Cooling Unit 2, Revision 107

B. SYMPTOMS OR ENTRY CONDITIONS

a) The following annunciators may cause entry into this procedure:

- RH PUMP TRIP, (2-6-A1)
- RH SUCT PRESS HIGH, (2-6-A3)
- RH PUMP 2A(B) DSCH PRESS HIGH, (2-6-B1,B2)
- RH PUMP 2A(B) DSCH FLOW LOW, (2-6-C1,C2)
- RH PUMP 2A(B) CC FLOW LOW, (2-6-D1,D2)
- RH HX CC WTR FLOW HIGH LOW, (2-2-A6)
- RC SYSTEM COLD PRESS HIGH, (2-12-D4)
- RC PRESS HIGH AT LOW TEMP PORV OPEN, (2-12-C4)
- CC PUMP TRIP, (2-2-A4)
- CC SURGE TANK LEVEL HIGH LOW, (2-2-A5)
- CC PUMP DSCH HDR PRESS LOW, (2-2-B5)
- REACTOR VESSEL LEVEL LOW, (2-6-B3)

b) The following symptoms may cause entry into this procedure:

- Oscillating RH pump amps
- Fluctuating RH Pump discharge pressure
- Excessive RH Pump noise or vibration
- Fluctuating letdown flow when on RH letdown
- Dropping PZR level
- Dropping Reactor vessel level
- Rising RCS temperature
- Rising RH system temperature

1 CHECK RH SYSTEM OPERATION:

a. RH pumps - ANY RUNNING

a. IF power has been lost to both 4KV ESF busses, THEN **GO TO** 2BwOA ELEC-8, LOSS OF ALL AC POWER WHILE ON SHUTDOWN COOLING.

IF any 4KV ESF bus is energized, THEN **GO TO** Step 2 (Next Page).

b. Operating RH pump parameters - STABLE:

- RH letdown flow
- RH pump amps
- RH disch pressure
- RH disch flow
- Reactor Vessel level
- RH pump noise/vibration

b. Attempt to stabilize RH system operation:

- o Reduce RH pump flow.
- o Raise RCS level by raising charging flow and/or reducing letdown flow.

IF RH pump parameters do not stabilize, THEN trip the affected RH pump.
GO TO Step 2 (Next Page).

Excerpt from BwAP 340-1, Use of Procedures for Operating Department, Revision 30

C. 3. Prudent Operator Actions

a. Guidance for taking Prudent Operator Actions (actions from memory or actions taken not described by procedures) is contained in OP-AA-102-1001. The following is how this guidance will be applied at Braidwood:

1) Performing necessary actions where no procedure exists. If unforeseen circumstances arise that present imminent personal injury, equipment damage, injury to the public or similar consequence, then actions outside of procedures may be taken, provided those actions are approved by the Shift Manager. The exception to the requirement for Shift Manager approval is action to prevent personal injury or to save a life.

2) During transients, actions required to place the plant in a stable condition may be performed from memory unless a prompt response procedure exists.

NRC Resolution

The U.S Nuclear Regulatory Commission (NRC) notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #6.

The distractor/answer choices addressed by the applicant's comment consist of the following:

Distractor B:

(1) continue in 2BwGP 100-1, PLANT HEATUP, ONLY
(2) take manual control of 2RH618, HX 1A BYP FLOW CONT VLV, to reduce flow

Answer D:

(1) enter 2BwOA PRI-10, LOSS OF RH COOLING UNIT 2
(2) take manual control of 2RH618, HX 1A BYP FLOW CONT VLV, to reduce flow

The stem of the question explicitly provided, in part, the following key information to the applicant:

- 2BwGP 100-1 in effect
- 2A RH train aligned for shutdown cooling
- 2A RH pump amps fluctuating between 50 to 60 amps

Under the above conditions, BwOP RH-6, "Placing the RH System in Shutdown Cooling," would have been previously performed to establish shutdown cooling operations. As stated in the information provided by the facility, "BwOP RH-6 would have been completed prior to where the stem begins the initial conditions for the question." Based upon this, the precautions of BwOP RH-6 (including step D.4) are no longer procedurally in effect during the timeframe associated with the stem conditions.

The first half of distractor "B" states "continue in 2BwGP 100-1, PLANT HEATUP, ONLY." The inclusion of the word "ONLY" in this distractor would limit any procedurally driven corrective actions for addressing the oscillating RH Pump amps to solely guidance contained within 2BwGP 100-1. However, 2BwGP 100-1 does not contain any procedural guidance for addressing conditions of oscillating RH Pump amps. Thus, the corrective action contained in the second part of distractor "B" ("take manual control of 2RH618, HX 1A BYP FLOW CONT VLV, to reduce flow") would not have a procedural basis within the context of this distractor.

In contrast, 2BwOA PRI-10, Loss of RH Cooling Unit 2, lists the symptom of oscillating RH Pump amps occurring as an entry condition. Based upon this, it would be appropriate to enter this Procedure 2BwOA PRI-10, RNO step 1.b, subsequently directs the required reduction in RH pump flow. Thus, answer "D" provides a procedurally directed means of correcting the issue presented in the stem.

Therefore, the NRC concludes that no change should be made to the key regarding this exam question.

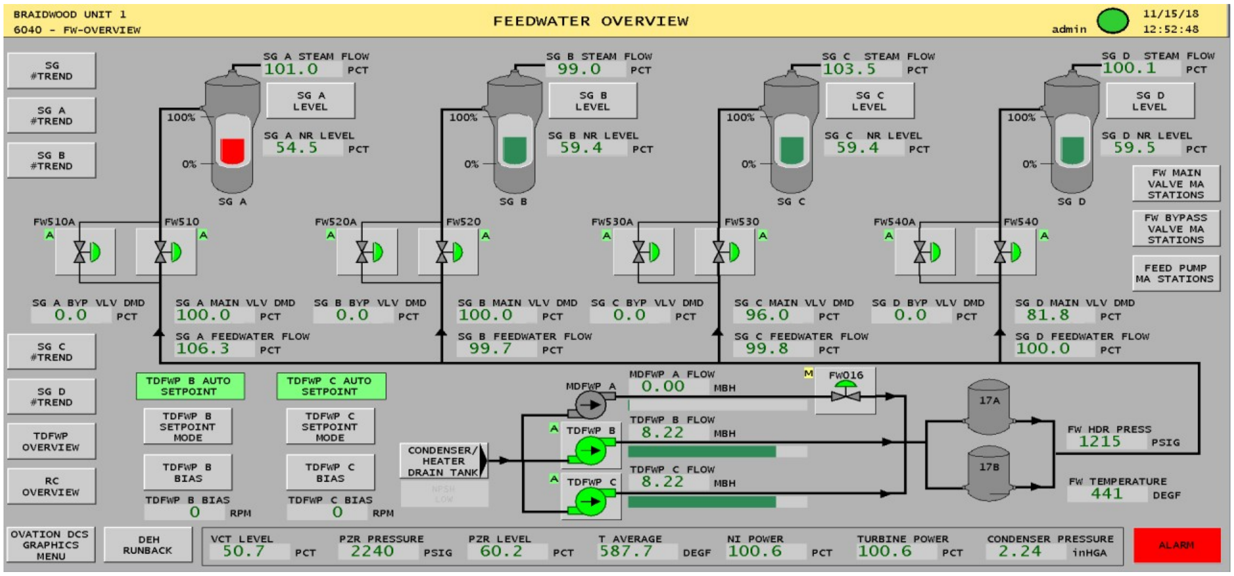
Question #10

Unit 1 is at 100 percent power.

The following annunciators have just alarmed:

- 1-1-A2, CNMT DRAIN LEAK DETECT FLOW HIGH
- 1-10-E4, OVATION SYSTEM TROUBLE
- 1-10-E5, OVATION ALTERNATE ACTION

The RO reviews OWS graphic 6040, FW OVERVIEW, and notes the following:



The crew will...

1. Reduce Unit 1 Turbine Loading
2. Trip Unit 1 Reactor
3. Initiate Safety Injection
4. Actuate Main Steamline Isolation

A. 1 ONLY.
B. 2 ONLY.
C. 2 and 3 ONLY.
D. 2, 3, and 4.

Answer D

Answer Explanation

A – Plausible: Reduce turbine loading only is incorrect. BwOA INST-2, OPERATION WITH A FAILED INSTRUMENT CHANNEL UNIT 1, Attachment E, NARROW RANGE SG LEVEL CHANNEL FAILURE, Step 2 RNO has actions to reduce turbine load. The examinee may plausibly conclude the narrow range SG level shown has been caused by a failed instrument and actions are needed to reduce power to less than 100 percent.

B – Plausible: Trip unit 1 reactor only is incorrect. The indications provided shows the containment leak detection flow high alarm in, coupled with the 1A SG level at 54.5 percent and 106.3 percent feed flow in the 1A SG. These conditions are indicative of a feedline break in containment, requiring a reactor trip. Incorrect because tripping the reactor, initiating SI and MSI are all high-level actions to mitigate the event in progress.

C – Plausible: Trip the reactor and initiate SI only is incorrect. The indications provided shows the containment leak detection flow high alarm in, coupled with the 1A SG level at 54.5 percent and 106.3 percent feed flow in the 1A SG. These conditions are indicative of a feedline break in containment, requiring a reactor trip. The examinee may plausibly conclude that only a reactor trip and SI is required to address this casualty since MSI does not close FWIVs. Incorrect because tripping the reactor, initiating SI and MSI are all high-level actions to mitigate the event in progress.

D – Correct: Trip the reactor, SI and MSI is correct. The indications provided shows the containment leak detection flow high alarm in, coupled with the 1A SG level at 54.5 percent and 106.3 percent feed flow in the 1A SG. These conditions are indicative of a feedline break in containment. The crew will trip the reactor, initiate SI and main steam isolation (MSI) as high-level actions to mitigate the event in progress.

Technical Reference and Revision #

1BwEP-2, Revision 300, Page 13.
1BwEP-0, Revision 303, Page 5.
_BwEP-2 Faulted Steam Generator Isolation Lesson Plan (I1-EP-XL-03), Revision 14, Page 6.

Applicant Comment

Q#10 The question is not bound by time and therefore could be understood to mean first vice all actions the crew will take.

Facility Position on Applicant Comment

Q#10 – No change required. The question did not require bounding by time in the stem since answer B was designated by "ONLY".

NRC Resolution

The NRC notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #10. The NRC also notes that, in accordance with NUREG-1021 ES-403, Section D.3.a and NUREG-1021 ES-501, Section D.2.d, the facility's performance analysis indicates that only 25 percent of applicants answered Question #10 incorrectly.

Although the applicant contends that the question stem does not specify a timeframe, this is irrelevant since the use of the word “only” in the answer/distractor choices logically results in “D” being the only correct answer to the question. Therefore, the NRC concludes that no change should be made to the key regarding this exam question. This is also consistent with the recommendation of the facility regarding this specific applicant comment.

Question #24

Both Units are at 100 percent power.

A fire occurs and the main control room (MCR) requires IMMEDIATE evacuation per 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY UNIT 1.

(1) Prior to leaving the MCR the reactor trip ______ be verified.
(2) Pressurizer LEVEL indication at the remote shutdown panel (shown below) …

A. (1) will
(2) will NOT require temperature correction.

B. (1) will
(2) WILL require temperature correction utilizing BwCB-1 FIGURE 31, PRESSURIZER LEVEL.

C. (1) will NOT
(2) will NOT require temperature correction.

D. (1) will NOT
(2) WILL require temperature correction utilizing BwCB-1 FIGURE 31, PRESSURIZER LEVEL.

Answer A

Answer Explanation

A – Correct: will and will NOT require temperature correction, are correct. Per the mitigating strategy of 1BwOA PRI-5 ONLY the reactor trip will be verified when an immediate evacuation of the MCR is required. Per 1BwGP 100-5 only the cold cal pressurizer level (1LT-462) is required to be corrected for pressurizer vessel liquid temperature.

B – Plausible: will is correct, WILL require temperature correction is incorrect. Temperature correction would be correct if the stem asked for monitoring the cold calibrated pressurizer level channel (1LT-462) during a normal shutdown per 1BwGP 100-5.

C – Plausible: will NOT is incorrect, will NOT require temperature correction is correct. This would be correct if the stem asked if the turbine trip will be verified during an immediate evacuation.

D – Plausible: will NOT and WILL require temperature correction are incorrect. This would be correct if the stem asked if the turbine trip will be verified during an immediate evacuation. Temperature correction would be correct if the stem asked for monitoring the cold calibrated pressurizer level channel (1LT-462) during a normal shutdown per 1BwGP 100-5.

Technical Reference and Revision #

1BwOA PRI-5, Revision 109, Page 3
Control Room Inaccessibility (_BwOA PRI-5) Lesson Plan, Revision 8, Page 10
1BwGP 100-5, Revision 58, Page 49

Applicant Comment

Q#24 Distractors B & D did not contain the entire name of the BwCB reference (BwCB-1 FIG. 31, Pressurizer Level 462 Cold Calibration). This made the question confusing to the applicants since the level instrument in the stem will have accuracy affected during the cooldown.

Facility Position on Applicant Comment

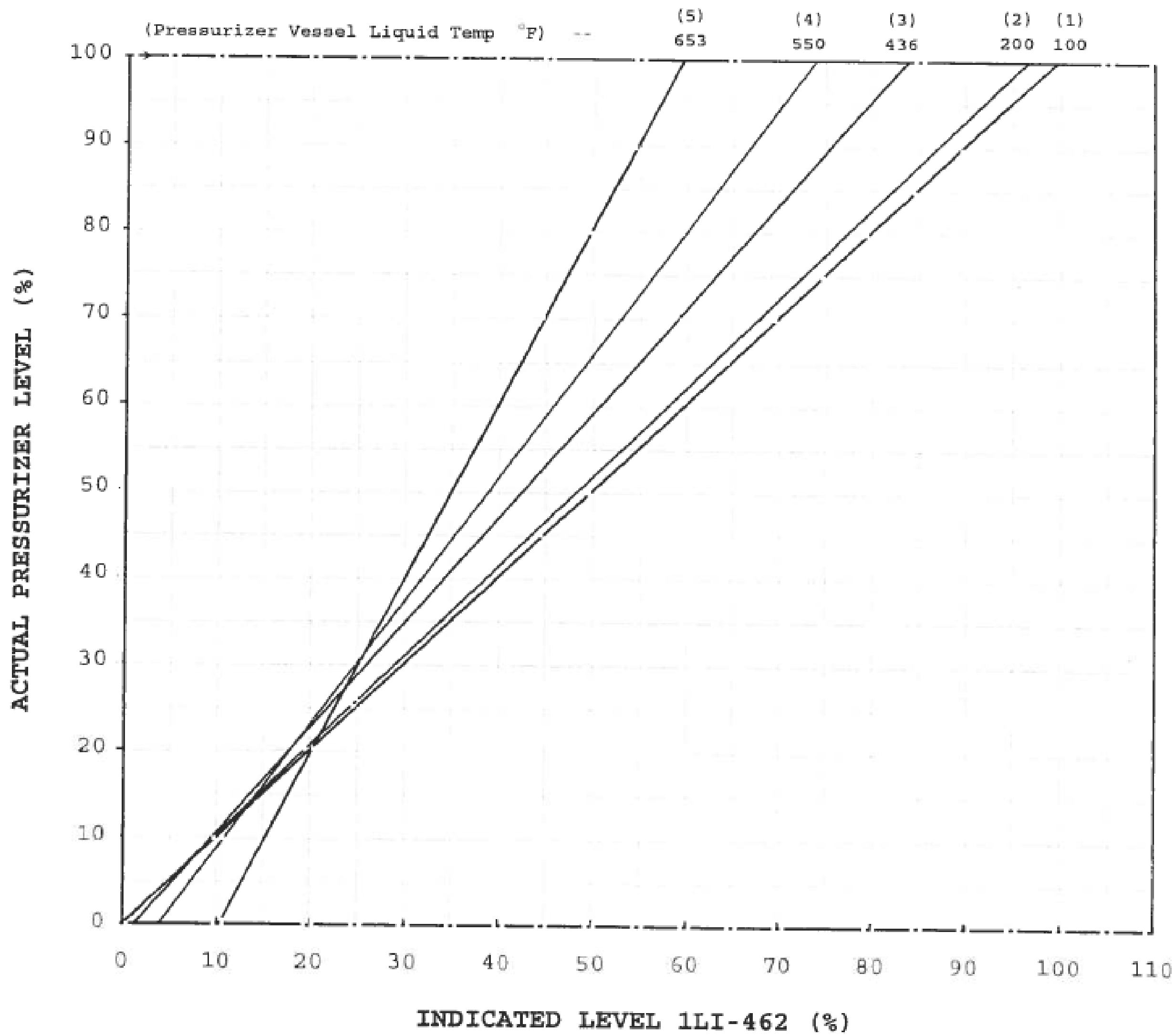
Q#24 – Accept B as a second correct answer. 1BwOA PRI-5 will direct a plant cooldown and depressurization if desired by the shift manager at step 23. This creates a situation where, during the cooldown, the accuracy of the hot calibrated pressurizer level instrumentation (1LI-459B & 1LI-460B as shown in the stem) could be inaccurate. Since the full title of the BwCB was not given and the figure was not provided as a reference, the question confused the applicant due to an unintended typographical error. Therefore, the applicant did not have the required information to determine the BwCB-1 Fig. 31 would not have been applicable.

Additional References

BwCB-1 Figure 31, Pressurizer Level 462 Cold Calibration, Revision 1

BwCB-1
Figure 31
Rev. 1
Page 1 of 1

PRESSURIZER LEVEL
462 COLD CALIBRATION



Cases
(1) : T= 100°F
(2) : T= 200°F
(3) : T= 436°F, P= 350 PSIG
(4) : T= 550°F, P= 1030 PSIG
(5) : T= 653°F, P= 2235 PSIG, NOT/NOP CONDITIONS

Excerpt from 1BwGP 100-5, Plant Shutdown and Cooldown, Revision 58

F. 52. After RCS temperature reaches less than 350°F, but prior 330°F, PERFORM the following:

f. RAISE PZR level while continuing with the cooldown, if the RCS will be placed in a solid condition:

1) DETERMINE 1LI-462 indication for 80% ACTUAL PZR level from BwCB-1 Figure 31 for the current PZR temperature.

2) ADJUST Charging/Letdown flow to control ACTUAL PZR level at 80%:

- BwCB-1 Figure 31
- 1LI-462.

NRC Resolution

The NRC notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #24. The NRC also notes that, in accordance with NUREG-1021 ES-403, Section D.3.a and NUREG-1021 ES-501, Section D.2.d, the facility's performance analysis indicates that only 37.5 percent of applicants answered Question #24 incorrectly.

Regardless of any reference made to "BwCB-1 FIGURE 31" in the answer/distractor combinations, the key wording in the available choices consists of "WILL require temperature correction" or "will NOT require temperature correction." 1BwGP 100-5 discusses correcting only the 1LI-462 (the cold calibrated instrument) level indication based upon pressurizer temperature conditions. Thus, including any reference to BwCB-1 FIGURE 31 was ultimately unnecessary, since only the understanding that 1LI-462 requires temperature correction during a cooldown (as specified in 1BwGP 100-5) was necessary to select between the options presented by the second half of the two-part question. Therefore, the NRC concludes that no change should be made to the key regarding this exam question.

Question #37

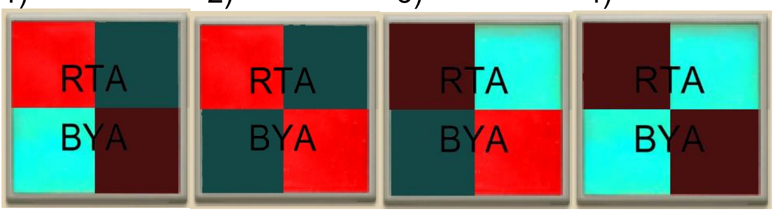
Unit 2 is at 100 percent power.

2BwOSR 3.3.1.4-1, UNIT TWO SSPS, REACTOR TRIP BREAKER, AND REACTOR TRIP BYPASS BREAKER SURVEILLANCE (TRAIN A) is in progress.

- The EO has racked the Train A Reactor Trip Bypass Breaker to the TEST position and has just completed step F.2.2.c, AT 2RD05E CLOSE THE TRAIN A REACTOR TRIP BYPASS BREAKER (BYA).

Which of the following indications at 2PM05J reflect the current status?

1) 2) 3) 4)



A. 1
B. 2
C. 3
D. 4

Answer C

Answer Explanation

A – Plausible: 1 is incorrect. With the BYA racked to the test position the indicating lights will have power. However, this configuration shows the RTA as red and BYA as green. This configuration is opposite of the correct configuration of RTA being green and BYA being red. The examinee may confuse the indications given and select this answer, since they are opposite.

B – Plausible: 2 is incorrect. With the BYA racked to the test position the closed indicating lights will have power. However, this configuration shows RTA as red and BYA as red. The indication for BYA is correct for the condition in the stem. The indication for RTA is incorrect. This answer would be correct if performance of 2BwOSR 3.3.1.4-1, Section 3.8.e had occurred.

C – Correct: 3 is correct. With the BYA racked to the test position the indicating lights will have power and the red indicating light will be lit (BYA closed). This answer shows this configuration.

D – Plausible: 4 is incorrect. With the BYA racked to the test position the indicating lights will have power. However, the green indicating light will not be lit (indicates BYA is open). The dark board / green board concept is frequently misunderstood by novice applicants and could cause them to select this answer. This answer would be correct if the EO had not closed the breaker.

Technical Reference and Revision #

2BwOSR 3.3.1.4-1, Revision 043

Applicant Comment

Q#37 The stem picture quality on the printed version of the exam caused confusion as to which lights were lit and which lights were out.

Facility Position on Applicant Comment

Q#37 – No change required. The exam was adequate to determine which indicating lights were lit. If this was unclear at the time, the examinee could have asked for clarification.

NRC Resolution

The NRC notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #37. The NRC also notes that, in accordance with NUREG-1021 ES-403, Section D.3.a and NUREG-1021 ES-501, Section D.2.d, the facility's performance analysis indicates that only 12.5 percent of applicants answered Question #37 incorrectly.

The applicant contends that their ability to answer this question was negatively affected by the print quality of the graphics on the administered examination. To validate this, the NRC was provided the actual page from the applicant's examination containing the graphics in question. A review of this page indicated the print quality was satisfactory and that the colors used in the graphics provided sufficient fidelity to those that would appear in the actual plant to allow for accurate interpretation by applicants. As previously noted, no clarification was requested by any applicant during the exam for this question and, furthermore, the facility's performance analysis indicates that only a single applicant answered this question incorrectly. Therefore, the NRC concludes that no change should be made to the key regarding this exam question. This is also consistent with the recommendation of the facility regarding this specific applicant comment.

Question #45

Per 1BwGP 100-3, POWER ASCENSION 5 PERCENT TO 100 PERCENT, what is the approximate steam flow from a SG, when the FW Bypass Reg Valves (1FW510A/520A/530A and 540A) automatically close?

A. 5%
B. 10%
C. 20%
D. 30%

Answer D

Answer Explanation

A – Plausible: 5 percent is plausible since the unit 2 MFW system allows tempering flow only to the SG less than 5 percent.

B – Plausible: 10 percent is plausible since this is the max power level the startup feedwater pump can go to.

C – Plausible: 20 percent is plausible since this is the approximate steam dump demand when the main generator is synchronized.

D – Correct: This is the power limit 1BwGP 100-3 states as the approximate power level where feed flow is transferred from the FW Bypass valves to the Main FW Reg Valves.

Technical Reference and Revision #

1BwGP 100-3, Revision 077, Page 16

Applicant Comment

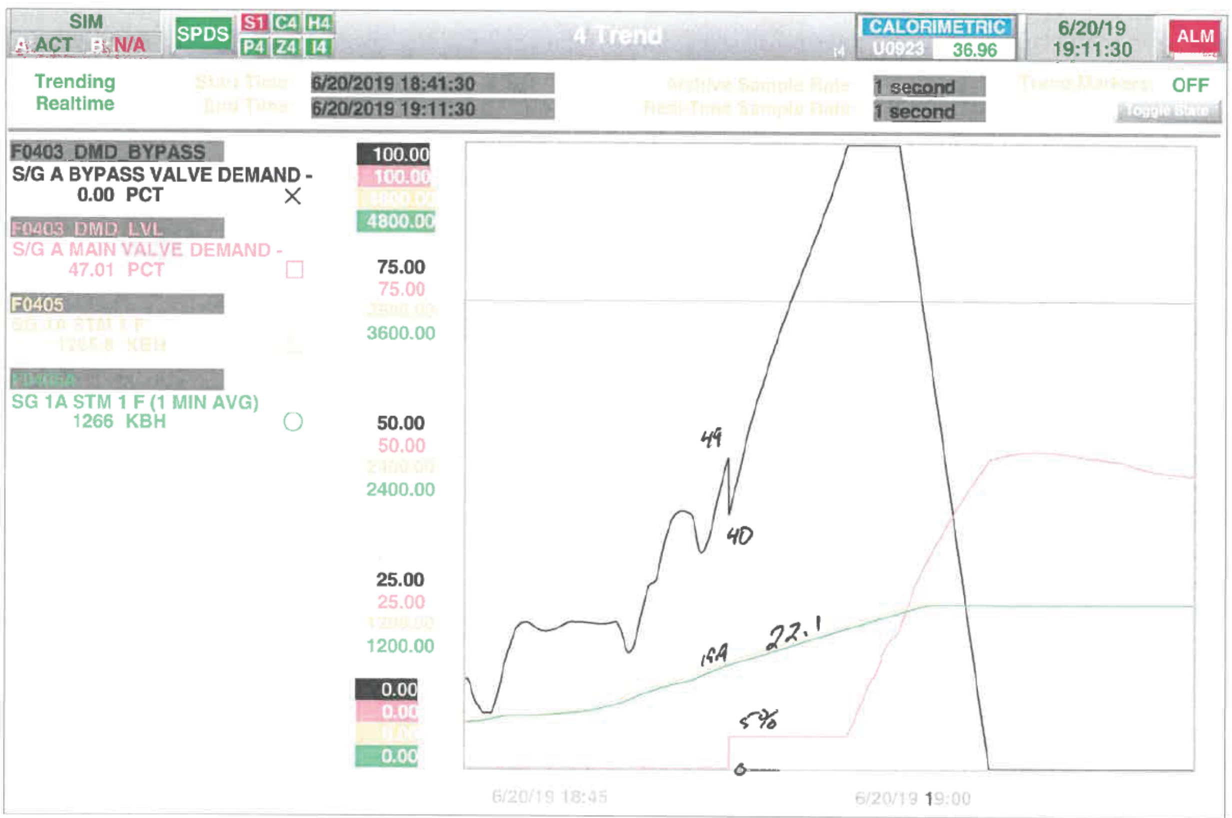
Q#45 The stem wording caused confusion. The mentioning of "automatically close" vice "get a closed signal" created an unstated assumption that the feed reg bypass valve would start to close. This would occur as control is transferred from the feed reg bypass valve to the feed reg valve. These conditions created a second possible correct answer, C.

Facility Position on Applicant Comment

Q#45 – Accept answer C as a second correct answer. Results from simulator testing show that at ~20% steam flow, the FRV Bypass valve does automatically throttle more closed, from 49% to 40%, as the FRV opens to 5% (See attached PPC printout from simulator trial). This makes it reasonable for the applicant to assume that the question was asking for FRV bypass demand response vice getting a closed signal as was intended.

Additional References

Simulator Response Data Provided by Facility



Excerpt from 1BwGP 100-3, Power Ascension 5 percent to 100 percent, Revision 77

E. 4. h. Automatic Feed Reg valve and Feedwater Reg Bypass valve control.

During power ascension the Feedwater Reg Bypass valves will initially throttle to maintain program level. When the Feedwater Reg Bypass valves are at approx. 50% open the associated Main Feedwater valves will get a signal to open to 5%. The Feed Reg Bypass valve will continue to throttle open and when the Feed Reg Bypass Valve is at 100% open the Main Feedwater Reg valve will throttle open which may cause the Bypass valve to throttle closed slightly. At this point the Main Feedwater Reg valve and the Feedwater Reg Bypass valve will work together to maintain program level. When the individual steam flow to a steam generator reaches approx. 30% (approx. 1.2 MLBM) the Feedwater Reg Bypass valve will get a closed signal transferring control solely to the Main Feedwater valve. Should the Feedwater Reg Bypass valve throttle closed to approx. 15% open the Main Feedwater valve will get a closed signal. During power descension The Feedwater Reg Bypass valve will start to open when steam flow for that individual steam generator is <25% (approx. 1 MLBM) and will slowly slew to 100% open and the Main Feedwater valve will throttle closed. The main valve will continue to control until the main valve demand is 5% then the bypass valve will begin to close from 100% and control the feedwater flow. When the Feedwater Reg Bypass valve throttles to approx. 15% open the Main Feedwater Reg valve will go closed transferring control solely to the Feedwater Reg Bypass valve.

NRC Resolution

The NRC notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #45.

The stem of the question states "[p]er 1BwGP 100-3…"; this focuses the context of the solicited response to information contained in that procedure 1BwGP 100-3, Section E.4.h, in turn states that "[w]hen the individual steam flow to a steam generator reaches approx. 30 percent (approx. 1.2 MLBM) the Feedwater Reg Bypass valve will get a closed signal transferring control solely to the Main Feedwater valve." It should be noted that the stem of the question does not ask at what power level that the FW Bypass Reg Valves begin to throttle; the stem instead asks "what is the approximate steam flow from a SG, when the FW Bypass Reg Valves… automatically close?" Thus, the question is clear in asking for the value provided by 1BwGP 100-3 for automatic Feedwater Reg Bypass valve closure. Therefore, the NRC concludes that no change should be made to the key regarding this exam question.

Question #53

Unit 1 is at 100 percent power.

Unit 2 is DEFUELED during a refueling outage.

- The 1A and 2B SX pumps are RUNNING.
- The 1B SX pump is in STANDBY.
- The 2A SX pump is OOS for maintenance.
- The 2B SX pump TRIPS on overcurrent.
- The 1B SX pump is STARTED.

Conditions of LCO 3.7.8, ESSENTIAL SERVICE WATER SYSTEM, are...

A. NOT met on Unit 1 ONLY.

B. NOT met on Unit 2 ONLY.

C. NOT met on BOTH Unit 1 and Unit 2.

D. MET on BOTH Unit 1 and Unit 2.

Answer A

Answer Explanation

A – Correct: NOT met on Unit 1 ONLY is correct. LCO 3.7.8 requires “One opposite-unit SX train for unit-specific support”, the conditions in the stem have both Unit 2 SX pumps inoperable.

B – Plausible: NOT met on Unit 2 ONLY is incorrect. LCO 3.7.8 is only applicable in modes 1-4. This would be the correct answer if the outage and online unit were reversed. The applicability of LCO 3.7.8, in various plant conditions, is frequently misunderstood by novice applicants.

C – Plausible: NOT met on BOTH Unit 1 and Unit 2 is incorrect. This would be the correct answer if Unit 2 was in mode 1-4. The applicability of LCO 3.7.8, in various plant conditions, is frequently misunderstood by novice applicants.

D – Plausible: MET on BOTH Unit 1 and Unit 2 is incorrect. This would be the correct answer if only one Unit 2 SX pump were inoperable or if both units were in mode 5. The applicability of LCO 3.7.8, in various plant conditions, is frequently misunderstood by novice applicants.

Technical Reference and Revision #

TS 3.7.8, Amendment 193
TS 3.7.9, Amendment 189

Applicant Comment

Q#53 The conditions of 3.7.8 are NOT met for both units. Therefore, answer c could also be correct. The stem lacked adequate focus and therefore created a confusing situation where the LCO conditions not being met was not an issue since Unit 2 was not in the mode of applicability.

Facility Position on Applicant Comment

Q#53 – No change required. The narrow interpretation of the term "conditions" in the stem, to determine an LCO entry is required for unit 2, is inappropriate.

Additional References

Excerpt from Braidwood Technical Specifications:

3.7 PLANT SYSTEMS

3.7.8 Essential Service Water (SX) System

LCO 3.7.8 The following SX trains shall be OPERABLE:

a. Two unit-specific SX trains; and

b. One opposite-unit SX train for unit-specific support.

APPLICABILITY: MODES 1, 2, 3, and 4.

NRC Resolution

The NRC notes that, in accordance with NUREG-1021 ES-403, Section D.1.a, and NUREG-1021 Appendix E, Section B.7, no questions were posed by any applicants during the examination regarding Question #53. The NRC also notes that, in accordance with NUREG-1021 ES-403, Section D.3.a and NUREG-1021 ES-501, Section D.2.d, the facility's performance analysis indicates that only 37.5 percent of applicants answered Question #53 incorrectly.

The applicant contends that distractor “C” should be a second correct answer. However, the applicant’s comment appears to not recognize that Unit 2 (listed as being defueled in the stem and thus being in “no mode”) is no longer within the Modes of Applicability of Technical Specification 3.7.8 (e.g., Modes 1, 2, 3, and 4). For distractor “C” to be correct, LCO 3.7.8 would also need to be applicable to Unit 2; based upon the conditions provided in the stem, it is not. Therefore, the NRC concludes that no change should be made to the key regarding this exam question. This is also consistent with the recommendation of the facility regarding this specific applicant comment.

SIMULATION FACILITY FIDELITY REPORT

Facility Licensee: Braidwood Station

Facility Docket Nos: 50–456 and 50–457

Operating Tests Administered: June 10, 2019, through June 17, 2019

The following documents observations made by the U.S Nuclear Regulatory Commission 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 U.S. Nuclear Regulatory Commission 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
Simulator Work Request #135325 Action Request #04259276	On June 11, 2019, an issue with the software code for the simulator resulted in an unplanned reactor trip in the middle of an applicant operating test scenario, necessitating cancellation of the remaining sessions of that scenario for the remainder of the day, as well as the administration of a “spare” scenario on a subsequent day. The issue involved a simulator error created by the way in which the simulator modeled Chemical Volume Control System flow and boron concentrations when a Centrifugal Charging Pump experienced a shaft shear event. The overall effect of this issue was that an erroneous boron calculation caused the modeling of an extremely large boron concentration in the discharge flow path of the standby CV pump. Subsequently, when this standby CV pump was started by an applicant crew, a rapid Reactor Coolant System temperature and pressure transient resulted in a reactor trip.