UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

October 17, 2019

Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000321/2019301 AND 05000366/2019301

Dear Ms. Gayheart:

During the period August 12 – 22, 2019 the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Edwin I. Hatch Nuclear Plant. At the conclusion of the tests, the examiners discussed preliminary findings related to the operating tests and the written examination submittal with those members of your staff identified in the enclosed report. The written examination was administered by your staff on August 29, 2019.

Four Reactor Operator (RO) and ten Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant failed the written examination. There were two post-administration comments concerning the written examination and two post-administration comments concerning the operating test. These comments, and the NRC resolution of these comments, are summarized in Enclosure 2. A Simulator Fidelity Report is included in this report as Enclosure 3.

The initial examination submittal was within the range of acceptability expected for a proposed examination. All examination changes agreed upon between the NRC and your staff were made according to NUREG-1021, “Operator Licensing Examination Standards for Power Reactors,” Revision 11.

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 (PARS) component of the NRC’s document

system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

If you have any questions concerning this letter, please contact me at (404) 997-4662.

Sincerely,

/RA/

Eugene F. Guthrie, Chief
Operations Branch 2
Division of Reactor Safety

Docket Nos.: 50-321, 50-366
License Nos.: DPR-57, NPF-5

Enclosures: 1. Report Details
2. Facility Comments and NRC Resolution
3. Simulator Fidelity Report

cc: Distribution via Listserv

SUBJECT: EDWIN I. HATCH –LICENSED OPERATOR INITIAL EXAMINATION 05000321/2019301 AND 05000366/2019301 dated October 17, 2019

***** See previous page for concurrence

☒ PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE ☒ NON-SENSITIVE

ADAMS: ☐ Yes ACCESSION NUMBER **ML 19290G553** ☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII/DRS/OB	RII/DRS/OB	RII/DRS/OB			
NAME	DEgelstad	BCaballero	GGuthre			
DATE	10/9/2019	10/ 15 /2019	10/ 17 /2019			
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Examination Report

Docket No.:	50-321, 50-366
License No.:	DPR-57, NPF-5
Report No.:	05000321/2019301 and 05000366/2019301
Enterprise Identifier:	L-2019-OLL-0025
Licensee:	Southern Nuclear Operating Company (SNC)
Facility:	Edwin I. Hatch Nuclear Plant, Units 1 and 2
Location:	Baxley, GA
Dates:	Operating Test – August 12 – 22, 2019 Written Examination – August 29, 2019
Examiners:	B. Caballero, Chief Examiner, Senior Operations Engineer D. Dumbacher, Senior Operations Engineer J. Bundy, Operations Engineer M. Emrich, Senior Reactor Technology Instructor T. Stephen, Senior Resident Inspector
Approved by:	Eugene F. Guthrie, Chief Operations Branch 2 Division of Reactor Safety

SUMMARY

ER 05000321/2019301, 05000366/2019301; August 12 – 22, 2019 & August 29, 2019; Edwin I. Hatch Nuclear Plant; Operator License Examinations.

Nuclear Regulatory Commission (NRC) examiners conducted an initial examination in accordance with the guidelines in Revision 11, of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors." This examination implemented the operator licensing requirements identified in 10 CFR §55.41, §55.43, and §55.45, as applicable.

Members of the Edwin I. Hatch Nuclear Plant staff developed both the operating tests and the written examination. The initial operating test, written RO examination, and written SRO examination submittals met the quality guidelines contained in NUREG-1021.

The NRC administered the operating tests during the period August 12 – 22, 2019. Members of the Edwin I. Hatch Nuclear Plant training staff administered the written examination on August 29, 2019. Four Reactor Operator (RO) and ten Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. Fourteen applicants were issued licenses commensurate with the level of examination administered.

There were four post-examination comments.

No findings were identified.

REPORT DETAILS

4. OTHER ACTIVITIES

4OA5 Operator Licensing Examinations

a. Inspection Scope

The NRC evaluated the submitted operating test by combining the scenario events and JPMs in order to determine the percentage of submitted test items that required replacement or significant modification. The NRC also evaluated the submitted written examination questions (RO and SRO questions considered separately) in order to determine the percentage of submitted questions that required replacement or significant modification, or that clearly did not conform with the intent of the approved knowledge and ability (K/A) statement. Any questions that were deleted during the grading process, or for which the answer key had to be changed, were also included in the count of unacceptable questions. The percentage of submitted test items that were unacceptable was compared to the acceptance criteria of NUREG-1021, "Operator Licensing Standards for Power Reactors."

The NRC reviewed the licensee's examination security measures while preparing and administering the examinations in order to ensure compliance with 10 CFR §55.49, "Integrity of examinations and tests." During the on-site preparatory week (July 8 – 12, 2019) the NRC audited a sample (approximately 10%) of the license applications (i.e., NRC Form 398) to confirm they accurately reflected the subject applicants' qualifications.

The NRC administered the operating tests during the period August 12 – 22, 2019. The NRC examiners evaluated four Reactor Operator (RO) and eleven Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. Members of the Edwin I. Hatch Nuclear Plant training staff administered the written examination on August 29, 2019. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Edwin I. Hatch Nuclear Plant, met the requirements specified in 10 CFR Part 55, "Operators' Licenses."

The NRC evaluated the performance or fidelity of the simulation facility during the preparation and conduct of the operating tests.

b. Findings

No findings were identified.

The NRC developed the written examination sample plan outline. Members of the Edwin I. Hatch Nuclear Plant training staff developed both the operating tests and the written examination. All examination material was developed in accordance with the guidelines contained in Revision 11, of NUREG-1021. The NRC examination team reviewed the proposed examination. Examination changes agreed upon between the NRC and the licensee were made per NUREG-1021 and incorporated into the final version of the examination materials.

The NRC determined, using NUREG-1021, that the licensee's initial examination submittal was within the range of acceptability expected for a proposed examination.One issue related to examination security was identified during preparation of the operating test. On April 30, 2019 the facility licensee generated Condition Report (CR) #10606406 and notified the NRC Chief Examiner that an unauthorized camera mount was discovered in the simulator, which had been installed for approximately two weeks by the class mentor to facilitate critique feedback during applicant training scenarios. On June 11, 2019, the NRC Office of Investigations (OI) conducted interviews with the personnel involved and toured the simulator. The NRC subsequently determined that no violation of NRC requirements existed, and further investigation was not warranted.

Four RO applicants and ten SRO applicants passed both the operating test and written examination. One SRO applicant passed the operating test but did not pass the written examination. Four RO applicants and ten SRO applicants were issued licenses.

Copies of all individual examination reports were sent to the facility Training Manager for evaluation of weaknesses and determination of appropriate remedial training.

The licensee submitted two post-examination comments concerning the operating test and two comments concerning the written examination. A copy of the final written examination and answer key, with all changes incorporated, and the licensee's post-examination comments, may be accessed not earlier than September 20, 2021, in the ADAMS system (ADAMS Accession Number(s) ML19275K795, ML19275K820, and ML19275K870).

4OA6 Meetings, Including Exit

Exit Meeting Summary

On August 22, 2019 the NRC examination team discussed generic issues associated with the operating test with Mr. Sonny Dean, Site Vice President, and members of the Edwin I. Hatch Nuclear Plant staff. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

KEY POINTS OF CONTACT

Licensee personnel

Sonny Dean, Site Vice President
Daniel A. Komm, Plant Manager
Tim Krienke, Operations Director
Michael Torrance, Engineering Director
Paul Mothena, Radiation Protection Manager
Keith Long, Regulatory Affairs Manager
Mike Kelly, Training Director
Mark Verbeck, Operations Training Manager
Gary Dudek, Operations Training Corporate Functional Area Manager
Ron Wheeler, Shift Supervisor
Terry Jones, Class Coordinator
Hank Strahley, Initial License Training Lead
John J. Payne, Lead Reactor Engineer
Zachary Howell, Reactor Engineer
Andrew R. Belcher, Services Manager
James Love, Licensing Engineer
Charlie Edmund, NRC Exam Reviewer
Anthony Ball, Exam Author
Arthur Genereux, Operations Training Instructor

NRC personnel

Clinton Jones, Senior Resident Inspector
Alan Blamey, Branch Chief, Division of Reactor Projects

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

A complete text of the facility licensee and applicant post-examination comments can be found in ADAMS under Accession Number ML19275K870. The facility licensee's post-examination comments were for two operating test administrative job performance measures (JPMs) and written exam SRO test items #78 and #93; the applicants' post-examination comments were only associated with the two written exam SRO test items.

Post-Examination Comment #1: **SRO Admin JPM, Conduct of Operations 34SUV-019-2, Surveillance & Tech Spec Required Actions**

The facility licensee contended that Initial Condition #2 ("*2T48-N30**3A**, Torus Temperature, is out-of-service and inoperable*") on the applicant's task cue sheet was a typographical error that was not detected prior to administration of the JPM; the facility licensee contended Initial Condition #2 should have said 2T48-N30**4B** (instead of -N303A). The reason the facility licensee contended that Initial Condition #2 was a typographical error was because it did not match the yellow colored inoperable elements on Attachment 1, Unit 2 SPDS Torus Temperature Diagnostic, which was also provided to the applicants as Initial Condition #6. The facility licensee contended that the typographical error caused the initial plant status presented to the applicants to not be clear with respect to the total number of out-of-service torus temperature elements. The facility licensee contended that the answer key should be revised to allow two correct methods of performing the torus temperature calculation – one method for applicants who performed the calculation with two out-of-service torus temperature elements, since there were only two yellow elements indicated Attachment 1, and another correct method of performing the calculation for the applicants who performed the calculation with three out-of-service torus temperature elements because Initial Condition #2 said that 2T48-N303A was out-of-service and inoperable.

Background

For this administrative JPM, the applicants were expected to complete Section 7.4 of 34SUV-019-2, Surveillance Checks, to perform channel checks (critical steps) of two groups of torus temperature elements and to calculate torus bulk average temperature (critical step). The applicants were expected to provide the examiner with a completed copy of Section 7.4; several line items in Section 7.4 were critical steps for satisfactory completion of the JPM. The seven initial conditions presented to the applicants were:

INITIAL CONDITIONS:

1. Unit 2 is operating at 100 % RTP.
2. 2T48-N303A, Torus Temperature, is out of service and inoperable.
3. 2T48-N308A, Torus Temperature, is out of service and inoperable.
4. 2T48-R647, Torus Bulk Average Temperature, on 2H11-P689 panel is indicating 95.0°F.
5. The Shift Supervisor has directed this surveillance to be completed as a paper version.
6. Attachment 1, Unit 2 SPDS Torus Water Temperature Diagnostic
7. Attachment 2, 2T47-R626 and 2T47-R627 Temperature Recorders

According to Initial Condition #2 and #3, the only two inoperable torus temperature elements were N303A and N308A; however, the Unit 2 SPDS Torus Water Temperature Diagnostic (Initial Condition #6) provided to the applicants did not reflect N303A as "yellow", i.e., inoperable. The Unit 2 SPDS Torus Water Temperature Diagnostic provided to the applicants reflected N304B and N308A as inoperable (yellow). The discrepancy between the Initial Conditions on the task cue sheet and the torus temperature element status on the Unit 2 SPDS Torus Water Temperature Diagnostic (Initial Condition #6) is summarized as follows:

Task Cue Sheet	Unit 2 SPDS Torus Water Temperature Diagnostic
N30**3A**	N30**4B** (yellow)
N308A	N308A (yellow)

The difference between the Task Cue Sheet and Attachment 1 was not detected during validation of the administrative JPM.

NRC Resolution: Licensee comment accepted

NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Rev. 11, Appendix C, Job Performance Measure Guidelines, Section B, Developing and Reviewing JPMs, stated:

1. *Specify Initial Conditions*
   Determine those system and plant conditions that would permit the task to be performed realistically. They should provide sufficient information about the status of the plant and system to facilitate task performance, without coaching the examinee.

Since there was an undetected discrepancy between the Task Cue Sheet and the Unit 2 SPDS Torus Water Temperature Diagnostic, the initial condition information provided to the applicants

could be interpreted to mean that there were THREE inoperable torus temperature elements (N303A, N304B, and N308A). It is operationally valid for a situation to exist where a torus temperature element remained out-of-service (inoperable) even though the element's parameter data field on the SPDS Torus Water Temperature Diagnostic was not yellow, i.e., appeared as "good" data. For example, when a torus temperature element was previously out-of-service and subsequently repaired, the element's parameter data field may NOT be yellow (because its instrument signal would be valid) but the instrument would remain inoperable (out-of-service) until post-maintenance testing was completed.

The intent of the JPM answer key was that only two elements were inoperable (N304B and N308A) since these temperature elements' parameter data fields on the Unit 2 SPDS Torus Water Temperature Diagnostic were yellow.

During administration of the JPM, some of the applicants interpreted the initial condition information to mean there were THREE inoperable torus temperature elements whereas other applicants performed the channel check and bulk average temperature calculation assuming that the TWO "yellow" temperature elements were the only inoperable elements. Therefore, the initial condition information provided to the applicants was not clear.

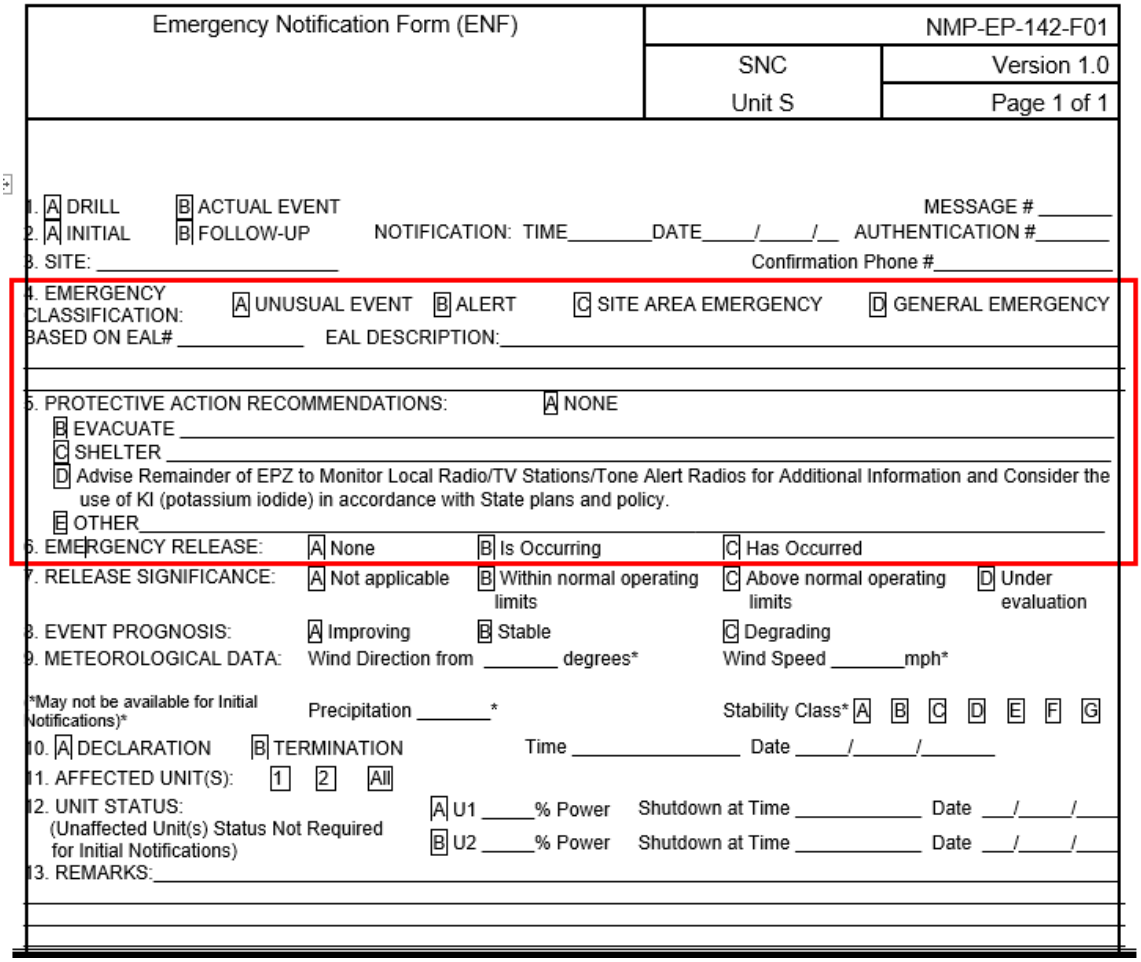
The facility licensee's recommendation to revise the answer key to allow two correct methods of performing the torus temperature calculation – one method for applicants who performed the calculation with two out-of-service torus temperature elements, since there were only two yellow elements indicated on the Unit 2 SPDS Torus Water Temperature Diagnostic, and another correct method of performing the calculation for the applicants who performed the calculation with three out-of-service torus temperature elements, was accepted.

Post-Examination Comment #2: SRO Admin JPM, Emergency Classification – Complete NMP-EP-142-F01, Emergency Notification Form

The facility licensee contended that the Unit 2 Shutdown Time (Line Item #12) on Emergency Notification Form (NMP-EP-142-F01) was undefinable based on the Initial Conditions provided to the applicants. For this reason, the facility licensee contended that the corresponding JPM Step #34 should NOT be a critical step.

Background

For this administrative JPM, the applicants were first expected to identify emergency declaration identifier "SS1" for Unit 2, including the basis for the classification, within 15 minutes (time critical step). After the applicants identified an emergency declaration identifier, the applicants were provided with an Emergency Notification Form (ENF) and were expected to complete Line Items 2 through 7, and Line Items 9 through 12, except unaffected unit status, within 15 minutes (time critical steps). The facility licensee's post exam comment was related to the "Shutdown at Time" portion of Line #12 (see below).



Emergency Notification Form (ENF) | NMP-EP-142-F01 | SNC Unit S | Version 1.0 | Page 1 of 1

1. [A] DRILL [B] ACTUAL EVENT MESSAGE # _____
2. [A] INITIAL [B] FOLLOW-UP NOTIFICATION: TIME_____DATE____/____/__ AUTHENTICATION #_____
3. SITE: _____ Confirmation Phone #_____
4. EMERGENCY CLASSIFICATION: [A] UNUSUAL EVENT [B] ALERT [C] SITE AREA EMERGENCY [D] GENERAL EMERGENCY
BASED ON EAL# _____ EAL DESCRIPTION:_____
5. PROTECTIVE ACTION RECOMMENDATIONS: [A] NONE
[B] EVACUATE _____
[C] SHELTER _____
[D] Advise Remainder of EPZ to Monitor Local Radio/TV Stations/Tone Alert Radios for Additional Information and Consider the use of KI (potassium iodide) in accordance with State plans and policy.
[E] OTHER_____
6. EMERGENCY RELEASE: [A] None [B] Is Occurring [C] Has Occurred
7. RELEASE SIGNIFICANCE: [A] Not applicable [B] Within normal operating limits [C] Above normal operating limits [D] Under evaluation
8. EVENT PROGNOSIS: [A] Improving [B] Stable [C] Degrading
9. METEOROLOGICAL DATA: Wind Direction from _____ degrees* Wind Speed _____mph*
May not be available for Initial Notifications) Precipitation _____* Stability Class* [A] [B] [C] [D] [E] [F] [G]
10. [A] DECLARATION [B] TERMINATION Time _____ Date ____/_____/_____
11. AFFECTED UNIT(S): [1] [2] [All]
12. UNIT STATUS: (Unaffected Unit(s) Status Not Required for Initial Notifications)
[A] U1 _____% Power Shutdown at Time _____ Date ___/_____/____
[B] U2 _____% Power Shutdown at Time _____ Date ___/_____/____
13. REMARKS:_____

The corresponding answer key for Line #12 was JPM Step 34, which was identified as a critical step.

**34.	NMP-EP-142-F01, step 12 Enter Unit Status: ********** ********** ********** ********** U2 Shutdown Time **********	NMP-EP-142-F01, step 12 The operator has entered the JPM start time in the space provided for U2 Shutdown Time.

<u>NRC Resolution: Licensee comment accepted</u>

NUREG-1021, Operator Licensing Examination Standards for Power Reactors, Rev. 11, Appendix C, Job Performance Measure Guidelines, Section B, Developing and Reviewing JPMs, states:

2. *Specify Initial Conditions*
   Determine those system and plant conditions that would permit the task to be performed realistically. They should provide <u>sufficient</u> information about the status of the plant and system to facilitate task performance, without coaching the examinee.

The Unit 2 shutdown time was not provided to the applicants in the Initial Conditions; the ten Initial Conditions provided to the applicants were:

INITIAL CONDITIONS:

1. You are the On Shift Shift Manager.
2. **Unit 2** is in a LOSP due to a failure of the "2C", "2D" and "2E" SUTs and a Loss of all onsite Emergency AC power due to a failure of all three Emergency Diesel Generators to start.
3. The Reactor scrammed and All rods fully inserted on the scram signal.
4. For the last 16 minutes, Emergency Diesel Generator start attempts have not been successful on any Emergency Diesel.
5. Maintenance estimates that the "2A" Emergency Diesel can be returned to service in 1 hour.
6. RWL is at -30 inches and being restored to the normal band using HPCI and RCIC.
7. All other Unit 2 parameters are in the desired bands.
8. Current meteorological conditions available on SPDS MIDAS Information provided
9. The following **Unit 1** conditions exists:
   100% power
   All parameters normal
10. NO Peer Check is available.

Initial Condition #3 stated that "the reactor scrammed and all rods inserted"; however, the time of the reactor shutdown was not included. The absence of a clear time of shutdown for Unit 2 was not detected during validation of the administrative JPM. During administration of the JPM, several applicants asked the proctor for the Unit 2 Shutdown Time. Therefore, the facility licensee's recommendation that JPM Step #34 (i.e., Unit 2 Shutdown Time portion of ENF Line #12) was not critical was accepted.

<u>Post-Examination Comment #3</u>: SRO Question #78

Six applicants contended that Question #78 should be deleted because the question stem did not include all the necessary information to answer the first part of the question. Specifically, the applicants contended that the question stem did not include information that 1) a second licensed operator was assigned to verify rod withdrawal sequence and 2) the Rod Worth Minimizer (RWM) was operable. The applicants contended that these two items were necessary for the minimum allowable number of operable Intermediate Range Monitors (IRMs) to be FOUR (one in each core quadrant) when the plant was in Mode 2. The applicants contended that since the two items were missing from the stem of the question, an assumption

was required to be made that these two conditions existed. The applicants contended that making the assumption that a second licensed operator was available and that the RWM was operable was contrary to the guidance in NUREG-1021, Appendix E, Policies and Guidelines For Taking NRC Examinations, Section B.7, which stated, in part:

> "*When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question.*"

The facility licensee concurred with the applicants' contention and recommended deleting the question because the stem did not include information that a second licensed operator was available and that the RWM was operable. The facility licensee based their recommendation on NUREG-1021, ES-403, Grading Initial Site-Specific Written Examinations, Section D.1.b, which stated, in part:

> "*Despite the extensive reviews performed by both the NRC and the facility licensee before examination administration…it is possible that a few isolated errors may be discovered only after an examination has been administered. The following types of errors are…are most likely to result in post-examination changes agreeable to the NRC:*
> - *A question with an unclear stem that confused the applicant or did not provide all the necessary information.*"

Background

Question #78 was:

78.

Unit 2 is in MODE 2 performing a Plant Startup IAW 34GO-OPS-001-2, Plant Startup.

Based on the above conditions, which ONE of the choices below completes the following statements?

IAW 34GO-OPS-001-2, the MINIMUM number of IRMs REQUIRED to be operable in Mode 2 is _____ .

IAW TS Bases 3.3.1.1, RPS Instrumentation, the bases for the IRM function is to ensure _____ .

A. 3 IRMs per RPS trip system;
peak fuel energy depositions are below the failure threshold

B. 3 IRMs per RPS trip system;
the MCPR SAFETY limit is not exceeded

C. 1 IRM channel in each core quadrant;
peak fuel energy depositions are below the failure threshold

D. 1 IRM channel in each core quadrant;
the MCPR SAFETY limit is not exceeded

The answer key indicated that Choice "C" was the correct answer.

NRC Resolution: Licensee comment **NOT** accepted

Question #78 tested the applicants' knowledge of the MINIMUM allowable number of IRMs; the question was NOT testing the requirements needed to support operation with the minimum number of IRMs. In accordance with 34GO-OPS-001-2, Plant Startup, the MINIMUM number of IRMs required to be operable in Mode 2 was one channel in each quadrant of the core, i.e., FOUR IRMs. No assumption was required to answer the first part of Question #78 because 1) control room staffing during a plant startup does not preclude a second licensed operator from being assigned to verify the control rod withdrawal sequence, and 2) the stem of the question did not contain any information that directly or indirectly implied the RWM was inoperable.

NUREG-1021, Appendix E, Policies and Guidelines For Taking NRC Examinations, Section B.7, which stated, in part:

> "*When answering a question, do not make assumptions regarding conditions that are not specified in the question unless they occur as a consequence of other conditions that are stated in the question.*"

The stem of the question did not contain any conditions/information that would, as a consequence, lead to a lack of control room staffing or the RWM being inoperable.

The first part of the question tested the applicants' knowledge of 34GO-OPS-001-2, Plant Startup, Precaution & Limitation 5.1.3, which stated:

> *A minimum of four IRM channels **must be operable** for the control rod withdrawal block instrumentation to be considered operable.*
> *One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second Licensed Operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.*

Additionally, 34GO-OPS-001-2, Step 7.1.14.2 and 7.1.14.3 stated:

> *#7.1.14.2 **Check** at least three IRM channels per RPS trip system are operable. (N/A IF < 3 IRM channels per trip system operable).*
>
> *#7.1.14.3 **Check** at least two IRM channels per RPS trip system are operable. (N/A IF 3 IRM channels per trip system are operable).*
>
> *#7.1.14.3.1 **Assign** a second Licensed Operator to verify compliance with the withdrawal sequence.*
>
> *#7.1.14.3.2 **Check** the RWM is operable.*
>
> *#7.1.14.3.3 **Check** one IRM channel in each quadrant of the core is OPERABLE.*

Tech Spec 3.3.1.1 Reactor Protection System (RPS) Instrumentation required TWO Channels per RPS trip system in Mode 2. This Tech Spec requirement was modified by NOTE "d", which stated:

> (d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

Tech Spec 3.3.2.1, Control Rod Block Instrumentation, did NOT include a requirement for IRM operability; Tech Spec 3.3.2.1 only included requirements for the Rod Block Monitor (RBM), RWM, and Reactor Mode Switch Shutdown Position functions.

The first part of Question #78 asked for the MINIMUM number of IRMs required to be operable in Mode 2. 34GO-OPS-001-2, Step 7.1.14.3 allowed for a minimum of TWO IRMs to be operable, provided that at least one IRM channel was operable in each quadrant of the core. Choice "C" (correct answer) stated: "1 IRM channel in each core quadrant". Since there are four core quadrants, a minimum of four IRMs were required to be operable. Choice "C" met the Tech Spec 3.3.1.1 RPS requirement that two channels per RPS trip system were required in Mode 2 because Choice "C" specified a minimum of four IRM channels. Therefore, the applicants' and facility licensee's recommendation that Question #78 be deleted was NOT accepted.

Post-Examination Comment #4: SRO Question #93

Five applicants contended that the answer to Question #93 should be changed from Choice "A" to Choice "B" because technical information, which was not known by the exam authors during the written exam development, was subsequently discovered. Specifically, the applicants contended that Reactor Building Component Cooling Water (RBCCW) heat exchanger outlet valves (2P41-F440A and -F440B) are normally full-open (instead of throttled) and the common downstream valve (2P41-F491) was used to adjust RBCCW temperature (instead of the heat exchanger outlet valve).

The applicants contended that the RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2) annunciator response procedure did NOT contain a specific step that provided direct guidance for adjusting the *correct* Plant Service Water Valve since it provided guidance for adjusting 2P41-F440A or -F440B, instead of adjusting the common downstream valve 2P41-F491. Therefore, the applicants contended that the answer key should be changed from Choice "A" to Choice "B."

The facility licensee concurred with the applicants' contention and cited NUREG-1021, ES-403, Grading Initial Site-Specific Written Examinations, Section D.1.b, which stated, in part:

> "*Despite the extensive reviews performed by both the NRC and the facility licensee before examination administration…it is possible that a few isolated errors may be discovered only after an examination has been administered. The following types of errors are…are most likely to result in post-examination changes agreeable to the NRC:*
>
> - *Newly discovered technical information that supports a change in the answer key.*"

Background

Question #93 was:

93.

Unit 2 is at 30% RTP during a startup after an extended outage when the following alarm is received:

RBCCW HX OUTLET TEMP HIGH (650-249)

The control room operator dispatches the Systems Operator to adjust Plant Service Water flow through the heat exchanger.

Based on the above conditions, which ONE of the choices below completes both statements?

The result of locally adjusting Plant Service Water flow is a _____ .

34AR-650-249-2 _____ contain a specific step that provides the direct guidance for adjusting the Plant Service Water valve.

A. reduced PSW to RBCCW differential pressure; does

B. reduced PSW to RBCCW differential pressure; does NOT

C. raised PSW to RBCCW differential pressure; does

D. raised PSW to RBCCW differential pressure; does NOT

The answer key indicated that Choice "A" was the correct answer. Only the second part of Question #93 was contested.

NRC Resolution: Licensee comment **NOT** accepted; question deleted

The guidance in the annunciator response procedure [RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2)] was inadequate for a high RBCCW heat exchanger outlet temperature condition because it directed the operator to throttle a valve that was already fully open; therefore, the question was technically inaccurate. The facility licensee generated condition report (CR) #10647129 to correct the annunciator procedure.

The second part of Question #93 tested the applicants' knowledge of whether the RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2) annunciator response procedure contained "a specific step that provides the direct guidance for adjusting the Plant Service Water valve."

The answer key indicated that Choice "A" was correct based on Step 5.2 of the RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2) annunciator response procedure which stated:

5.2 THROTTLE OPEN 2P41-F440A(2P41-F440B), RBCCW HX Serv Water Disch Vlv, to decrease RBCCW HX outlet temperature as indicated locally on 2P42-R033.

However, 34SO-P41-001-2, Plant Service Water System, Attachment 2, Plant Service Water System Valve Lineup, identified that the following normal valve positions:

VALVE NUMBER	DESCRIPTION	NORMAL POSITION
2P41-F440A	PSW Return from RBCCW Heat Exchanger, 2P42-B001A, Isolation Valve	OPEN
2P41-F491	PSW Return from RBCCW Heat Exchangers Isol. Valve	OPEN *
2P41-F440B	PSW Return from RBCCW Heat Exchanger, 2P42-B001B, Isolation Valve	OPEN

* **2P41-F491 may be throttled to maintain PSW/RBCCW differential pressure acceptable**

The PSW Return from RBCCW Heat Exchanger 2P42-B001A and -B001B Isolation Valves (2P41-F440A and -F440B) were required to be full open even though the RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2) annunciator response procedure provided guidance to throttle open the valves during a high temperature condition. Therefore, the annunciator response procedure guidance was inadequate for a high RBCCW heat exchanger outlet temperature condition because it directed the operator to throttle a valve that was already fully open.

34SO-P41-001-2, Section 7.1.2 (Initial Division 1 Pump Startup) and Section 7.1.3 (Initial Division 2 Pump Startup) included the following Step for correcting a high RBCCW heat exchanger outlet temperature condition:

IF necessary,
throttle 2P41-F491, PSW return valve from the RBCCW Hx,
to maintain Hx, PSW/RBCCW Diff pressure AND
RBCCW Hx outlet temperature WITHIN limits. ☐

The applicants' and facility licensee's recommendation to change the answer key from Choice "A" to Choice "B" was not incorporated because the test item was based on an inadequate procedure. The facility licensee initiated condition report (CR) 10647129 to correct the annunciator procedure. Although Step 5.2 of RBCCW HX OUTLET TEMP HIGH (34AR-650-249-2) annunciator response procedure did technically include "a specific step that provides the direct guidance for adjusting the Plant Service Water valve", the guidance was incorrect because it directed adjusting Isolation Valves 2P41-F440A and -F440B even though these valves are already fully open. Therefore, the question was deleted from the exam.

SIMULATOR FIDELITY REPORT

Facility Licensee: Hatch

Facility Docket No.: 05000321/2019301 & 05000366/2019301

Operating Test Administered: August 12 – 22, 2019

This form is to be used only to report observations. These observations do not constitute audit or inspection findings and, without further verification and review in accordance with Inspection Procedure 71111.11 are not indicative of noncompliance with 10 CFR 55.46. No licensee action is required in response to these observations.

No simulator fidelity or configuration issues were identified.