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

March 4, 2019

William R. Gideon
Site Vice President
Brunswick Steam Electric Plant
8470 River Rd. SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000325/2019301 AND 05000324/2019301

Dear Mr. Gideon:

During the period January 7 – 15, 2019 the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Brunswick Steam Electric Plant. At the conclusion of the tests, the examiners discussed preliminary findings related to the operating tests with those members of your staff identified in the enclosed report. The written examination was administered by your staff on January 22, 2019.

Two 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 and passed the operating test. There were 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) 997-4551.

Sincerely,

/RA/

Gerald J. McCoy, Chief
Operations Branch 1
Division of Reactor Safety

Docket Nos: 50-325, 50-324
License Nos: DPR-71, DPR-62

Enclosures:
1. NRC ER 05000325/2019301, and 05000324/2019301
2. Facility Comments and NRC Resolution
3. Simulator Fidelity Report

cc: Distribution via Listserv

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000325/2019301 AND 05000324/2019301 dated March 4, 2019

*See previous page for concurrence

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ADAMS: ☐ Yes ACCESSION NUMBER:_ **ML19064A560** _☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

OFFICE	RII/ DRS/ OB1	RII/ DRS/ OB1	RII/ DRS/ OB1	RII/ DRS/ OB1	RII/ DRS/ OB1	
SIGNATURE	BLC2	MAB7	JDB10	NTL2	GJM1	
NAME	B.Caballero	M. Bates	J. Bundy	N. Lacy	G. McCoy	
DATE	2/ 28 / 2019	3/ 1 /2019	3/ 1 /2019	2/ 21 / 2019	3/ 4 / 2019	
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	

OFFICIAL RECORD COPY DOCUMENT NAME: G:\OLExams\Brunswick Examinations\Initial Exam 2019-301 (Bruno)\Correspondence\Brunswick 2019301ER

U.S. NUCLEAR REGULATORY COMMISSION

Examination Report

Docket No.:	50-325, 50-324
License No.:	DPR-71, DPR-62
Report No.:	05000325/2019301, 05000324/2019301
EPID No.:	L-2019-OLL-0034
Licensee:	Duke Energy Progress, Inc.
Facility:	Brunswick Steam Electric Plant, Units 1 and 2
Location:	Southport, NC
Dates:	Operating Test – January 7 – 15, 2019 Written Examination – January 22, 2019
Examiners:	Bruno Caballero, Chief Examiner, Senior Operations Engineer Mark Bates, Senior Operations Engineer Jason Bundy, Operations Engineer Newton Lacy, Operations Engineer
Approved by:	Gerald J. McCoy, Chief Operations Branch 1 Division of Reactor Safety

Enclosure 1

SUMMARY

ER 05000325/2019301, 05000324/2019301; operating test January 7 – 15, 2019 & written exam January 22, 2019; Brunswick Steam Electric Plant, Units 1 and 2; 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 Brunswick Steam Electric 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 January 7 – 15, 2019. Members of the Brunswick Steam Electric Plant training staff administered the written examination on January 22, 2019. Two 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. Twelve applicants were issued licenses commensurate with the level of examination administered.

There were two post-examination comments pertaining to the operating test.

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."

The NRC administered the operating tests during the period January 7 – 15, 2019. The NRC examiners evaluated two Reactor Operator (RO) and eleven Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. Members of the Brunswick Steam Electric Plant training staff administered the written examination on January 22, 2019. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Brunswick Steam Electric 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 Brunswick Steam Electric 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.

The original dates for the on-site preparatory week (October 22, 2018), administration of the operating test (weeks of November 26 and December 3, 2018) and administration of the written exam (week of December 10, 2018) were identified in the April 26, 2018 exam notification letter (ML18128A380). Due to the impact of Hurricane Florence, members of the Brunswick Steam Electric Plant training staff requested that the on-site preparatory week, and administration of the operating test and written exam be postponed. The new dates were identified in the October 4, 2018 exam postponement notification letter (ML18282A212).

Two 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. Two 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. 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 February 2, 2021, in the ADAMS system (ADAMS Accession Numbers ML19032A142, ML19032A147, and ML19032A163.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 15, 2019 the NRC examination team discussed generic issues associated with the operating test with Mr. Randy Gideon, Site Vice President, and members of the Brunswick Steam Electric 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

Randy Gideon, Site Vice President
Karl Moser, Plant General Manager
Frank Giannone, Training Manager
Ed Rau, Operations Training Supervisor
Craig Oliver, Operations Training Supervisor
Bruce Leitch, Operations Training Supervisor
Mark DeWire, Assistant Operations Manager – Shift
Dan Geraghty, Site Monitoring Lead
Jerry Pierce, Nuclear Support Services Manager
Stephen Yodersmith, Regulatory Affairs Engineer

NRC Personnel

Jeff Steward, Resident Inspector

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

A complete text of the licensee's post-examination comments can be found in ADAMS under Accession Number ML19032A163. The licensee's post-examination comments were both associated with the operating test; one comment was related to an administrative job performance measure (JPM) and the other related to a simulator scenario event.

Post-Examination Comment #1: Emergency Plan, Administrative JPM SOT-ADM-JP-301-A22, Complete an Emergency Notification Form, Rev. 1

The licensee contended that the answer key was incorrect for the Emergency Plan Administrative SRO JPM; the licensee contended that emergency plan classification was not a Site Area Emergency (FS1.1, loss or potential loss of any two barriers). Instead, the licensee contended that the correct classification was an Alert (FA1.1, any loss or any potential loss of either Fuel Clad or RCS). The licensee contended that the failure of one main steam line to auto-isolate did not constitute a loss of the primary containment barrier, because the Emergency Action Level Technical Basis document clarified that, if the condenser remained available, a "direct pathway to the environment" did not exist.

Background

The following plant conditions and initiating cues were provided to the applicants:

TASK CONDITIONS:

1. You are the Site Emergency Coordinator.
2. The Emergency Operations Facility is not yet fully staffed.
3. Unit 1 is at 100% power.
4. The following conditions exist on Unit 2:
   - RVCP and EDP are being performed
   - Low pressure systems are injecting
   - Compensated reactor water level is -29 inches and slowly rising
   - Inboard and Outboard C MSIVs failed to isolate
   - Primary coolant activity is 270 μCi/gm I-131 dose equivalent
   - Hi-Range Drywell Area Rad monitor indicates 500 R/hr
   - Reactor Building Negative Pressure (VA-PI-1297) indicates -0.3 inches water
   - Stack Rad Monitor indicates 2.01e+06 μCi/sec.
   - Wind speed is 6.8 mph, wind direction is 218.3°.

INITIATING CUE:

1. Evaluate the current conditions to determine EAL applicability (EC Judgment is NOT to be used). Write the time, classification and EAL identifier in the table below then immediately raise your hand so the evaluator can log your completion. (**Time Critical**)

TIME	CLASSIFICATION	EAL IDENTIFIER

2. Using the EAL Classification above, complete the Initial Nuclear Power Plant Emergency Notification Form and immediately raise your hand so the evaluator can collect your completed form. (**Time Critical**)

For the first initiating cue (determination of the EAL classification), the answer key included the following JPM steps and associated performance standards, which were critical steps:

Step 1 – Determine Classification threshold and associated EAL Number(s)
FS1.1 – Loss or Potential Loss of any two barriers (Table F-1)

****CRITICAL STEP** SAT/UNSAT**

Step 2 – Classification made within required the required time
(Declaration Time minus Start Time $\leq$ 15 minutes).
Classification declared within 15 minutes of Start Time.

**** CRITICAL STEP ** SAT/UNSAT**

The premise for why the JPM answer key identified a Site Area Emergency FS1.1 classification was that the task conditions indicated that the reactor coolant system (RCS) and primary containment barriers were lost, while the fuel clad barrier remained intact, based on the following:

- RCS Barrier Loss: A loss of the RCS existed because Emergency Depressurization was in progress, which meant that RCS safety relief valves were opened to the suppression pool.

- Primary Containment Barrier Loss: A loss of the primary containment barrier existed due to the failure of the inboard and outboard main steam isolation valves to close on the “C” steam line.

- Fuel Clad Barrier remained Intact: Drywell radiation was < 2000 R/hr and primary coolant activity was < 300 µCi/gm I-131 dose equivalent, which did not exceed the Table F-1 Fission Product Barrier Threshold Matrix values for the fuel clad barrier.

The facility licensee did not submit any post exam comments associated with the second initiating cue task (completing the Nuclear Power Plant Emergency Notification Form), which required the applicants to identify that a release was in progress based on the elevated Stack Radiation Monitor release rate at 2.01 E+6 µCi/sec.

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

Emergency Action Levels (EALs) are the plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions, as defined in the Brunswick Emergency Plan. Brunswick EALs are based on NEI 99-01, Methodology for the Development of EALs for Non-Passive Reactors, Rev. 6, November 2012 (ML12326A805).

PEP-02.1, Initial Emergency Actions, EAL-1, Modes 1, 2, & 3, Table F-1 Fission Product Barrier Threshold Matrix, identified the following threshold for Loss of the Primary Containment Barrier:

1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal

NEI 99-01, Section 9, Fission Product Barrier ICS/EALs, BWR Containment Barrier Thresholds, Primary Containment Isolation Failure, Loss 3.A, stated:

> *The use of the modifier "direct" in defining the release path discriminates against release paths through interfacing liquid systems or minor release pathways, such as instrument lines, not protected by the Primary Containment Isolation System (PCIS).*

PEP-02.2.1, EAL Technical Bases, Attachment 2, Fission Product Barrier Matrix and Basis, Category E, PC Integrity or Bypass, further clarified:

> *…the adjective "Direct" modifies "release pathway" to discriminate against release paths through interfacing liquid systems. Leakage into a closed system is to be considered only if the closed system is breached and thereby creates a significant pathway to the environment. Examples include unisolable Main steam line, HPCI steam line or RCIC steam line breaks, unisolable RWCU system breaks, and unisolable containment atmosphere vent paths.* ***If the main condenser is available with an unisolable main steam line, there may be releases through the steam jet air ejectors and gland seal exhausters. These pathways are monitored, however, and do not meet the intent of a nonisolable release path to the environment.***

The task conditions presented to the applicants did not include any information related to the main condenser availability; therefore, in accordance with NUREG-1021, Rev. 11, Appendix E, Section D, Walkthrough Test Guidelines, the applicant was expected to make decisions and take actions based on the indications available, i.e., the condenser remained available because no information was presented to the contrary. Therefore, based on PEP-02.2.1, EAL Technical Bases, the failure of one main steam line to auto-isolate did not constitute a loss of the primary containment barrier.

The task conditions indicated a release was in progress which was operationally valid for three reasons.

First, an ongoing release was operationally valid because the primary coolant activity was 270 µCi/gm I-131, and one main steam line failed to auto-isolate. With the main condenser available, the radioactive discharge from the steam jet air ejector and steam packing exhauster was ultimately routed to the main stack, which was monitored.

Secondly, the 2.01 E+06 µCi/sec release rate value was significantly elevated above normal release rates for dual unit operation. The 2.01 E+06 µCi/sec release rate value was greater than the PEP-02.1, Initial Emergency Actions, EAL-1, Modes 1, 2, & 3, Table R-1, Effluent Monitor Classification Thresholds, value for an Unusual Event related to gaseous releases, but less than the Alert threshold at 2.13 E+07 µCi/sec.

Additionally, a release in progress during an Alert classification was operationally valid because 0PEP-02.2.1, Section 5.1, Definitions, stated the following information related to off-site releases:

- Unusual Event: No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of Safety Systems occur.

- Alert: Any releases are expected to be limited to small fractions of the EPA PAG exposure levels.

- Site Area Emergency: Any releases are not expected to result in exposure levels which exceed EPA PAG exposure levels beyond the site boundary.

Therefore, the licensee's post-examination comment was accepted, and in accordance with NUREG-1021, Rev. 11, the final as-administered JPM was annotated with these approved changes before being entered to ADAMS.

Post-Examination Comment #2: Scenario 2, Event 2, Rod Drift

For Scenario 2, Event 2, the licensee contended that the scenario guide's control rod operability determination was incorrect; the licensee contended that control rod 06-23, which was fully inserted, satisfied the LCO for Tech Spec LCO 3.1.3, Control Rod Operability, and entry to Condition C [One or more control rods inoperable for reasons other than Condition A or B] was not appropriate.

Background

Scenario 2, Event 2 was Control Rod Drift. This event was designed with simulator malfunction RD001M, Control Rod Slow Insertion Drift. The expected actions identified in the Scenario Form ES-D-2, Required Operator Actions, were:

- Recognize control rod 06-23 was drifting inward from its initial full-out position 48.
- Implement annunciator procedure A-05, 3-2, Rod Drift, by attempting to arrest the inward rod movement by applying a withdraw signal to the rod.
- Once it was determined that the rod continued to drift inward, the expected action was to fully insert the rod.
- Enter Tech Spec 3.1.3, Condition "C", which included Action C.1 (Insert the rod within 3 hours) and Action C.2 (Disarm the rod within 4 hours).

Once the operator fully inserted the control rod to position 00, the simulator malfunction RD001M was deleted.

The premise for why the scenario guide (Form ES-D-2) identified Tech Spec 3.1.3, Condition "C" was the assumption that the Work Control Center would issue a hydraulic tagout for the rod, which would make its accumulator inoperable. In accordance with Tech Spec 3.1.5, Control Rod Scram Accumulators, Condition "A" [One control rod scram accumulator inoperable with reactor steam dome pressure ≥ 950 psig], either Action A.1 (Declare the rod slow) or Action A.2 (Declare the rod inoperable) was required when one accumulator was inoperable.

None of the crews entered Tech Spec 3.1.3, Condition "C" because they determined that the rod was performing its safety function while it remained at position 00 with its accumulator fully charged.

NRC Resolution: Licensee comment accepted.

The basis for Tech Spec LCO 3.1.3 stated:

> *The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position.*

The surveillance requirements for Tech Spec 3.1.3 were:

- *SR 3.1.3.1: Determine the position of each control rod*
- *SR 3.1.3.2: Insert each withdrawn control rod at least one notch*
- *SR 3.1.3.3: Verify each control rod scram time from fully withdrawn to notch position 06 is ≤ 7 seconds.*
- *SR 3.1.3.4: Verify each control rod does not go to the withdrawn overtravel position.*

In Scenario 2, Event 2, the control rod was latched at position 00 and there were no indications of abnormally elevated temperatures on the control rod drive (i.e., CRD HYD TEMP HIGH, A-05, 1-2, annunciator was not alarming); therefore, the rod scram time was not adversely affected. None of the Tech Spec 3.1.3 surveillance requirements were adversely affected. Subsequent Work Control Center actions required by plant procedures to isolate and disarm the control rod would have rendered the control rod inoperable, however, these actions were not completed during the scenarios; consequently, control rod 06-23 remained operable.

Therefore, the licensee's post-examination comment was accepted, and in accordance with NUREG-1021, Rev. 11, the final as-administered scenario was annotated with these approved changes before being entered to ADAMS.

SIMULATOR FIDELITY REPORT

Facility Licensee: Brunswick Steam Electric Plant

Facility Docket No.: 05000325, 05000324

Operating Test Administered: January 7 – 15, 2019

This form is 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.