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January 24, 2019

Serial: RA-19-0083

U.S. Nuclear Regulatory Commission, Region II
ATTN: Ms. Catherine Haney, Regional Administrator
245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

Subject: Brunswick Steam Electric Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-71 and DPR-62
Docket Nos. 50-325 and 50-324
Reactor Operator and Senior Reactor Operator License Post-Examination Documentation

Reference: Letter from Gerald J. McCoy (NRC) to William R. Gideon (Duke Energy), "Brunswick Steam Electric Plant – Postponement of Licensed Operator Initial Examination 05000325/2018301 and 05000324/2018301 due to Hurricane Florence," dated October 4, 2018, ADAMS Accession Number ML18282A212

Dear Ms. Haney:

In accordance with the guidance contained in Revision 11 of NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Section ES-501, "Initial Post-Examination Activities," Duke Energy Progress, LLC (Duke Energy), is providing the NRC the specified post examination documentation for the reactor operator and senior reactor operator examinations. The written examinations were administered at the Brunswick Steam Electric Plant on Tuesday, January 22, 2019.

The documentation listed in the enclosure of this letter is being provided electronically via a secure File Transfer Protocol (FTP) website to Mr. Bruno Caballero, the assigned NRC chief examiner, only. The documentation listed in the enclosure is not included herein. As confirmed acceptable with the NRC chief examiner, the ES-201-3 form, "Examination Security Agreement," with all the pre- and post-examination signatures will be provided via email upon completion.

This document contains no regulatory commitments. Please refer any questions regarding this submittal to Mr. Jerry Pierce, Manager – Nuclear Support Services, at (910) 832-7931.

Sincerely,

William R. Gideon

SBY/sby

Enclosure: List of Post-Examination Documentation

cc:

U.S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Bruno Caballero, Chief Examiner
245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

U.S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Gerald J. McCoy, Chief
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245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

U.S. Nuclear Regulatory Commission
ATTN: Mr. Dennis J. Galvin (Mail Stop OWFN 8B1A)
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Rockville, MD 20852-2738

U.S. Nuclear Regulatory Commission
ATTN: Mr. Gale Smith, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

Chair - North Carolina Utilities Commission **(Electronic Copy Only)**
4325 Mail Service Center
Raleigh, NC 27626-0510
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List of Post-Examination Documentation

The following documentation is being provided electronically via a secure File Transfer Protocol (FTP) website to Mr. Bruno Caballero, the assigned NRC chief examiner, only. The documentation is not included herein.

1. Graded Written Examinations and Applicants' Answer Sheets
2. Master Examination and Answer Key
3. Applicants' Questions Asked and Answers Given During the Written Examination
4. Review Comments
5. Written Examination Seating Chart
6. ES-403-1, "Written Examination Grading Quality Checklist"
7. Written Examination Performance Analysis Results (with recommended substantive changes)

Facility Post Exam Comment for the SRO EP JPM

JPM 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°.

JPM 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

The JPM answer key states:

Intermediate declarations are not required and are not Critical. The final declaration is a critical step. Final classification:

FS1.1 – Loss or Potential Loss of **any** two barriers (Table F-1)

Primary Containment Barrier - Loss - E1

Reactor Coolant System Barrier - Loss - B2 or D1

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT
F Fission Product Barriers	FG1.1 1 2 3 Loss of any two barriers AND Loss or potential loss of third barrier (Table F-1)	FS1.1 1 2 3 Loss or potential loss of any two barriers (Table F-1)	FA1.1 1 2 3 Any loss or any potential loss of either Fuel Clad or RCS (Table F-1)

EAL Table F-1:

	Reactor Coolant System Barrier		Primary Containment Barrier	
	Loss	**Potential Loss**	**Loss**	**Potential Loss**
A. RPV Water Level	1. RPV level cannot be restored and maintained > TAF or cannot be determined	None	None	1. Entry to SAMG-01 required
B. RCS Leak Rate	1. UNISOLABLE break in any of the following: - Main steam line - HPCI steam line - RCIC steam line - RWCU - Feedwater 2. Emergency Depressurization is required	1. UNISOLABLE primary system leakage that results in exceeding EITHER of the following: • One or more Secondary Containment area radiation Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-3) • One or more Secondary Containment area temperature Maximum Normal Operating Limits (0EOP-03-SCCP Table SC-1)	1. UNISOLABLE primary system leakage that results in exceeding one or more Secondary Containment area temperature Maximum Safe Operating Limits (0EOP-03-SCCP Table SC-1)	None
C. PC Conditions	1. Primary Containment pressure > 1.7 psig due to RCS leakage	None	1. UNPLANNED rapid drop in Primary Containment pressure following Primary Containment pressure rise 2. Primary Containment pressure response not consistent with LOCA conditions	1. Primary Containment pressure > 62 psig 2. Deflagration concentrations exist inside PC (H2 ≥ 6% AND O2 ≥ 5%) 3. Heat Capacity Temperature Limit (HCTL) exceeded
D. PC Rad / RCS Activity	1. Drywell radiation > 27 R/hr with reactor shutdown	None	None	1. Drywell radiation > 20,000 R/hr
E. PC Integrity or Bypass	None	None	1. UNISOLABLE direct downstream pathway to the environment exists after Primary Containment isolation signal 2. Intentional Primary Containment venting per EOPs	None

Based on OPEP-02.2.1, Emergency Action Level Technical Bases, the determination that the Primary Containment Barrier (E1) was lost is incorrect. See the highlighted portion of the bases document below.

EP also has confirmed that the bases is correct and follows the norm within the industry.

Barrier: Primary Containment

Category: E. PC Integrity or Bypass

Degradation Threat: Loss

Threshold:

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

Definition(s):

UNISOLABLE - An open or breached system line that cannot be isolated, remotely or locally.

Basis:

This threshold addresses failure of open isolation devices which should close upon receipt of a manual or automatic containment isolation signal resulting in a significant radiological release pathway directly to the environment. The concern is the unisolable open pathway to the environment. A failure of the ability to isolate any one line indicates a breach of primary containment integrity.

As stated above, 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. These minor releases are assessed using the Category R, Abnormal Rad Release / Rad Effluent, EALs.

The existence of an in–line charcoal filter (SBGT) does not make a release path indirect since the filter is not effective at removing fission noble gases. Typical filters have an efficiency of 95-99% removal of iodine. Given the magnitude of the core inventory of iodine, significant releases could still occur. In addition, since the fission product release would be driven by boiling in the reactor vessel, the high humidity in the release stream can be expected to render the filters ineffective in a short period.

The answer key should be changed to the following final classification:

FA1.1 – **Any** loss or **any** potential loss of either Fuel Clad or RCS (Table F-1)

Reactor Coolant System Barrier - Loss - B2 or D1

Prepared by: [signature]

EP Approval: [signature]

Facility Rep: [signature] Leitch

Facility Comment on the Tech Spec determination for Scenario 2 Event 2.

The answer key stated to declare the control rod inoperable IAW TS 3.1.3. This is not correct based on the following (follows the IDO format):

Entry Condition

Control rod drifting in to 00.

From Tech Spec bases for LCO 3.1.3:

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. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

There is no mission time applicable for the control rod system.

Surveillance requirements for control rod per TS 3.1.3 are:

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

Basis for Reasonable Expectation of Operability

With a control rod drifting in to position 00 no technical specification surveillance requirements are adversely impacted for the control rod. In this scenario the control rod was latched at position 00 and was performing its required function. Also, there were no indications of abnormally elevated temperatures on the control rod drive (i.e. CRD Hydraulic Temperature High alarm was not received) so scram times would not be adversely affected. Subsequent actions required by plant procedures to isolate and disarm the control rod would render the control rod inoperable, however, these actions were not completed in the observed scenarios. Therefore, given the identified condition of the control rod drifting in to 00 the control rod remains operable.