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

December 30, 2014

Mr. Benjamin C. Waldrep
Vice President
Duke Energy Progress, Inc.
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000400/2014302

Dear Mr. Waldrep:

During the period November 17 – 21, 2014 the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Shearon Harris Nuclear 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 November 25, 2014.

All applicants passed both the operating test and written examination. There were four post-administration comments concerning the written examination. 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 9, Supplement 1.

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 No: 50-400
License No: NPF-63

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

cc: Distribution via Listserv

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ADAMS:x ☐ Yes ACCESSION NUMBER:_ML14364A375______________ ☐ SUNSI REVIEW COMPLETE ☐ FORM 665 ATTACHED

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-400
License No.:	NPF-63
Report No.:	05000400/2014302
Licensee:	Duke Energy Progress, Inc.
Facility:	Shearon Harris Nuclear Plant, Unit 1
Location:	5413 Shearon Harris Road New Hill, NC 27562
Dates:	Operating Test – November 17-21, 2014 Written Examination – November 25, 2014
Examiners:	David Lanyi, Chief Examiner, Senior Operations Engineer Newton Lacy, Operations Engineer Joseph Viera, Operations Engineer
Approved by:	Gerald McCoy Operations Branch Division of Reactor Safety

Enclosure 1

SUMMARY

ER 05000400/2014302; operating test November 17 – 21, 2014 & written exam November 21, 2014; Shearon Harris Nuclear Plant; Operator License Examinations.

Nuclear Regulatory Commission (NRC) examiners conducted an initial examination in accordance with the guidelines in Revision 9, Supplement 1, 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 Shearon Harris 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 November 17 – 21, 2014. Members of the Shearon Harris Nuclear Plant training staff administered the written examination on November 25, 2014. All Reactor Operator (RO) and Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. All 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."

The NRC administered the operating tests during the period November 17 – 21, 2014. The NRC examiners evaluated four Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. Members of the Shearon Harris Nuclear Plant training staff administered the written examination on November 25, 2014. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Shearon Harris 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. Shearon Harris 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 9, Supplement 1, 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.

All applicants passed both the operating test and written examination and 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 four post-examination 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 November 25, 2016--two years after administration of the written exam, in the ADAMS system (ADAMS Accession Numbers ML14338A038 and ML14338A036).

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 21, 2014, the NRC examination team discussed generic issues associated with the operating test with Mr. Benjamin Waldrep, Site Vice-President, and members of the Shearon Harris 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

B. Waldrep, Site Vice-President
J. Dufner, Plant General Manager
D. Hayes, Operations Manager
D. Griffith, Training Manager
S. Schwindt, Operations Training Manager
D. Corbett, Manager – Regulatory Affairs
E. Betram, Operations Instructor
R. Horton, Senior Nuclear Operations Instructor
A. Lucky, Senior Nuclear Operations Instructor
G. Pickar, Initial Training Supervisor
S. Rua, NLO / Exam Supervisor
S. Scott, Assistant Operations Manager – Training
R. Vondenberg, Assistant Operations Manager – Shift
M Wallace, Senior Technical Specialist – Regulatory Affairs

NRC personnel

J. Austin, Senior 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 ML14338A046.

Item

Question 5, K/A 015AA1.16

Comment

The licensee recommends that there is no correct answer for RO question #5.

The question asks whether or not a reactor trip would occur based on plant conditions described by Bistable Status on the Bypass Permissive Light Box (BPLB) and the Trip Status Light Boxes (TSLB). The first half of the question asked whether a reactor trip WOULD or WOULD NOT occur. The second half of the question asked which Reactor Protection System (RPS) Permissive caused the reactor to trip/not trip. The keyed answer is that a reactor trip would occur because of the status of P-7 (Low Power Trips Blocked).

The stem of the question states that the BPLB and TSLB both indicate that both the P-7 and P-10 (Power Range > 10%) bistables were lit. This is not possible unless there is a fault, because one illuminated light indicates greater than 10% power and the other indicates less than 10% power.

Clarification for this question was given during the exam in that the Bypass Permissive Light Box window names for each RPS Permissive were written down. The clarification provided implied that the P-7 BPLB light was being referenced in the question.

The stem of the question and the additional guidance provided in the clarification led candidates to answer the question based on BPLB status. Since the conditions in the stem of the question could not exist in any plant condition using BPLB indications, this is an invalid question without a correct answer.

NRC Resolution

The licensee's recommendation was accepted.

The question was written upon the mistaken supposition that if the P-7 permissive light were lit, the interlock would be met. The stem also stated that the P-10 permissive light was lit. Further review of the wiring diagrams reveal that P-7 would be extinguished when power is greater than 10% and that P-10 would be lit when power was greater than 10%. The configuration of having both lights lit at the same time is not possible without a circuitry failure.

The question was written to test the applicants' knowledge of low-power reactor trip block status lights during a loss of Reactor Coolant flow. Specifically, the intent was to examine their knowledge that the reactor trip was due to loss of two out of three loops when power was between the P-7 power and P-8 (Single Loop Low Flow Trip Blocked) power (approximately 49% power). The concept was that with the P-7 light on and the P-8 light off, the applicant was to infer that power was between these two permissives. They would then be able to choose the correct answer that a trip would occur because two of the three Reactor Coolant Pumps would have lost power (P-7 permissive). However with the P-7 light illuminated, the stem of the question led the applicants to conclude that power was actually less than 10% and the P-7 induced loss of flow trips were not valid. No trip would have occurred in this case. No answer was provided that would have allowed “no trip occurring” due to the status of P-7.

Since the P-7 permissive appeared to not be met, there were no correct answers provided and the question was deleted.

Item

Question 27, K/A WE15EA1.3

Comment

The licensee recommends that there is no correct answer for RO question #27.

The first part of the question requires the candidate to evaluate containment parameters and choose the appropriate Function Restoration (FR) Procedure to implement. The current construction of the Containment Critical Safety Function Status Tree (CSFST) as adopted by the licensee evaluates containment pressure before containment sump level or radiation levels. CSFST rules of usage as described in EOP USERS GUIDE section 5.2 states “At any given time, a Critical Safety Function status is represented by a single path through its tree. Since each path is unique, it is uniquely labeled at its end point, or terminus. This labeling consists of color coding and/or line-pattern-coding of the terminus and last branch line, plus a transition to an appropriate FR if required by that safety status.” Since containment pressure is evaluated before evaluation of sump levels, the Containment Status Tree would result in a YELLOW terminus requiring transition to FR-Z.1. This YELLOW path effectively blocks evaluation of the ORANGE path terminus for Containment flooding. This results in an entry into EOP-FR-Z.1 Response to High Containment Pressure, eliminating answers “C” and “D” from being correct.

The second part of this question requires the candidate to know what needs to be sampled as required by the implementing procedure. The stem of the question stated that bus 1A2-SA was de-energized due to a fault. This fault results in a loss of one train of Containment Spray Pumps and Emergency Service Water (ESW) Booster Pumps. Since one train of ESW booster pumps remain in service (B train), service water for the de-energized train is simply isolated per step 9.a RNO, not sampled. Since nothing is sampled in EOP-FR-Z.1 with the current plant conditions, “A” and “B” are also incorrect leaving no correct answer for this question.

NRC Resolution

The licensee’s recommendation was accepted.

The Westinghouse Emergency Response Guidelines (ERG) Background Document for F-0.5, Section 2 states “When the status tree ‘rules of usage’ are applied to F-0.5, CONTAINMENT, with a spray pump running and containment pressure between the spray actuation pressure (T.02) and the design pressure (T.03), then a YELLOW priority will result. The operator should be aware that this YELLOW priority can be reached without evaluating the ORANGE priority for entry into FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, based on high containment sump level. This priority scheme should not present conflicts for plants with a large, dry containment (like the reference plant) due to the containment pressure behavior following an event that releases sufficient mass and energy into the containment atmosphere to actuate containment spray, and the value of footnote (T.06) for entry into FR-Z.2, RESPONSE TO CONTAINMENT FLOODING." The background document goes on to provide options for how to change the Containment Status Tree if it were determined that containment flooding should be evaluated before pressure was reduced less than the Containment Spray actuation setpoint. The licensee has not adopted any of these changes and currently uses the ERG version of the Containment Status Tree as is. Therefore, the question as written would recommend entry into FR-Z.1. Based upon that procedure, no guidance would be given on sampling any water in containment. Therefore, none of the answers provided were correct.

Since no correct answers were provided, the question has been deleted.

Item

Question 51, K/A 064A2.04

Comment

The licensee recommends that there is no correct answer based on the information provided in the stem of the question.

The first part of the question requires the candidate to evaluate the time required to shut down the Emergency Diesel Generator (EDG) from 35% load. The second part of the question asks for the impact of taking the action per the first part of the question. There is no comment on the second part of the question. The basis for the timely shutdown of the EDG is to minimize carbon buildup which is both "B" and "D" answers.

The comment on the first part of the question is there was not enough information provided in the stem of the question to provide an operationally accurate response. Section 7.1.2, step 9 of OP-155, "Diesel Generator Emergency Power System" states "At the MCB (Main Control Board), WHEN stack exhaust temperatures are less than 500°F,THEN POSITION DIESEL GENERATOR A-SA (B-SB) control switch to STOP". Since these temperatures were not provided there was not enough information provided for the candidate to determine if the note that states "The EDG should be shutdown from 35% load in less than 5 minutes to minimize carbon buildup" was applicable.

Without the stack temperatures it is not possible to determine the time the EDG should be shutdown.

NRC Resolution

The licensee's recommendation was rejected.

The lack of stack exhaust temperature data was irrelevant for the question asked. Although there have been occasions in the past where the EDGs had to be operated unloaded for periods greater than five minutes. The note to shutdown the EDG within 5 minutes is always applicable unless other conditions prevent performing it in that time frame. In this case, no information was provided on the stack exhaust temperature, therefore the logical assumption would be that the temperature did not hinder a timely shutdown.

The lack of information in the stem would not cause confusion about what was being asked. If any confusion was experienced by the applicants by the stem of the question, they were repeatedly informed that they should ask for clarification. The applicants asked no clarifying questions about lack of stack exhaust temperature data. Therefore there was adequate information in the stem to answer the question.

Item

Question 73, K/A G2.4.17

Comment

The licensee recommends that the correct answer to this question is “A” and not “B” as keyed.

The question required the applicant to evaluate the condition of Reactor Coolant System (RCS) RCS pressure during an operator controlled cooldown following a small break Loss of Coolant Accident (LOCA) and the basis for that decision. The EOP USERS GUIDE section 6.5 states “The operator is frequently asked to check RCS and SG (Steam Generator) pressures and temperature as STABLE (or RISING). STABLE does not necessarily imply constant. RCS and/or SG pressure may be dropping slowly due to an operator-controlled cooldown and still be considered stable. If the operator can control the rate and magnitude of the pressure change, then pressure should be considered stable.” Therefore RCS pressure should be considered STABLE due to the pressure drop seen being a direct result of an operator controlled cooldown, which makes answer “A” correct and answers “C” and “D” incorrect.

Answer “B” provides a basis for calling RCS pressure stable as RCS subcooling is rising. While a rising subcooling is a diverse indication of a stable/rising pressure, it is not an indication that is referenced in the EOP users guide revision that was used to write this exam question or by the applicants as they prepared for the exam.

Based on the information provided in the EOP Users Guide answer A is the more correct answer to this question.

NRC Resolution

The licensee’s recommendation was partially accepted

Based upon the assumption made by the applicants during the test that a controlled cooldown was in progress, it is reasonable to understand that RCS pressure would be slowly lowering due to the operator’s actions. However, rising subcooling with a slowly lowering RCS pressure continues to be a legitimate indication of stable pressure.

Both answer “A” and “B” were accepted as correct answers.

SIMULATOR FIDELITY REPORT

Facility Licensee: Shearon Harris Nuclear Plant

Facility Docket No.: 50-400

Operating Test Administered: November 17 – 21, 2014

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.