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

February 16, 2021

Mr. Don Moul
Executive Vice President,
Nuclear Division and Chief Nuclear Office Florida Power & Light Company
Mail Stop: EX/JB
700 Universe Blvd.
Juno Beach, FL 33408

SUBJECT: ST. LUCIE NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000335/2020301 AND 05000389/2020301

Dear Mr. Moul:

During the period December 7 – 12, 2020, the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the St. Lucie 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 December 17, 2020.

Two Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant received an excusal for the operating test and passed the written examination. One SRO applicant passed the operating test, but did not pass the SRO-only portion of the written examination. There were five post-exam comments submitted on the written exam and two post-administration comments concerning the operating test. These comments, and the NRC resolution of these comments, are summarized in Enclosure 2.

The initial operating test examination submitted by your staff did not meet the guidelines for quality contained in NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Revision 11, as described in the enclosed report. The RO and SRO written examinations did meet the NUREG-1021 quality guidelines.

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: 50-335, 50-389
License Nos: DPR-67, NPF-16

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

cc: Distribution via Listserv

SUBJECT: ST. LUCIE NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000335/2020301 AND 05000389/2020301 dated February 16, 2021

DISTRIBUTION:
M. Meeks, RII
K. Kirchbaum, RII
G. McCoy, RII

* See previous page for concurrence

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ADAMS: ☒ Yes ACCESSION NUMBER ML21047A496: ☒ SUNSI REVIEW COMPLETE ☒ FORM 665 ATTACHED

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	RII:DRS
NAME	MMEEKS	KKIRCHBAUM	JBUNDY	TMORRISSEY	G. McCoy
DATE	2/16 /2020	2/ 16 /2020	2/16/2020	2/ 15 /2020	2/16/2021
E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Examination Report

Docket No.:	05000335, 05000389
License No.:	DPR-67, NPF-16
Report No.:	05000335/2020301, 05000389/2020301
EPID No.:	L-2020-OLL-0022
Licensee:	Florida Power and Light Company (FP&L)
Facility:	St. Lucie Nuclear Plant, Units 1 & 2
Location:	6351 S. Ocean Drive Jensen Beach, FL 34957
Dates:	Operating Test – December 7-12, 2020 Written Examination – December 17, 2020
Examiners:	M. Meeks, Chief Examiner, Senior Operations Engineer K. Kirchbaum, Operations Engineer, Chief Examiner Under Instruction J. Bundy, Operations Engineer T. Morrissey, Senior Resident Inspector (Examiner Qualified)
Approved by:	Gerald J. McCoy, Chief Operations Branch 1 Division of Reactor Safety

SUMMARY

ER 05000335/2020301, 05000389/2020301; operating test December 7-12, 2020 & written exam December 17, 2020; St. Lucie 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.

The NRC developed the written examination outline. Members of the St. Lucie Nuclear Plant staff developed both the operating tests and the written examination. The initial operating test submittal did not meet the quality guidelines contained in NUREG-1021.

The NRC administered the operating tests during the period December 7-12, 2020. Members of the St. Lucie Nuclear Plant training staff administered the written examination on December 17, 2020. Two Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant received an excusal for the operating test and passed the written examination. One SRO applicant passed the operating test, but did not pass the SRO-only portion of the written examination. Nine applicants were issued licenses commensurate with the level of examination administered.

There were seven 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 performed an audit of the license applications during the preparatory site visit in order to confirm that they accurately reflected the subject applicants’ qualifications in accordance with NUREG-1021.

The NRC administered the operating tests during the period of December 7-12, 2020. The NRC examiners evaluated two Reactor Operator (RO) and seven Senior Reactor Operator (SRO) applicants using the guidelines contained in NUREG-1021. Members of the St. Lucie Nuclear Plant training staff administered the written examination on December 17, 2020. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the St. Lucie 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 St. Lucie 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 that the licensee's operating examination submittal was outside the range of acceptable quality specified by NUREG-1021. The initial operating test submittal required multiple JPM replacements after the onsite validation for JPMs that did not meeting the standards of NUREG-1021. Also, multiple simulator scenarios were required to be modified during onsite validation week and re-validated during that week. The initial events of one simulator scenario required re-validation during exam week prior to administration. Future examination submittals need to incorporate lessons learned.

The NRC determined that the licensee's initial written examination submittal (RO and SRO) was within the range of acceptability expected for a proposed examination.

Two Reactor Operator (RO) and six Senior Reactor Operator (SRO) applicants passed both the operating test and written examination. One SRO applicant received an excusal for the operating test and passed the written examination. One SRO applicant passed the operating test, but did not pass the written examination. Nine applicants were issued licenses commensurate with the level of examination administered.

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 five post-examination comments concerning the written examination. A full copy of the licensee post-examination comments may be accessed in the ADAMS system under ADAMS Accession Number ML21034A531. A copy of the final written examination and answer key, with all changes incorporated, may be accessed not earlier than December 19, 2022, in the ADAMS system as ML21034A519.

4OA6 <u>Meetings, Including Exit</u>

<u>Exit Meeting Summary</u>

On December 15, 2020, the NRC examination team discussed generic issues associated with the operating test with Mr. Dan DeBoer, Site Vice President, and members of the St. Lucie Nuclear Plant staff via video conference. The examiners asked the licensee if any of the examination material was proprietary. No proprietary information was identified.

On February 2, 2021, the NRC examiners discussed the examination results, licensing actions, and other items to be documented in the examination report with Mr. Seth Duston, Site Training Manager, and other members of the St. Lucie Training department staff via video conference.

KEY POINTS OF CONTACT

Licensee personnel

B. Beltz, Safety Assurance and Learning
D. DeBoer, Site Vice President
S. Duston, Training Manager
T. Fisher, Operations Instructor
W. Godes, Licensing Manager
C. Hill, Corporate Training Manager
B. Hinze, Operations Training Supervisor
K. Keith, Operations Class Mentor
C. Martin, Chemistry/Radiation Protection Manager
T. Ouret, Operations Training Supervisor
K. Paez, Licensing
T. Spillman, Assistant Operations Manager
R. Story, Outage Manager
R. Virgin, Operations Training Supervisor
S. Wylie, Examination Author

NRC personnel

T. Morrissey, Senior Resident Inspector

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

The facility submitted seven post-examination comments as indicated in a letter which can be found in ADAMS under Accession Number ML21034A531.

Post-Examination Comment #1: Question 12, K/A 056 AK3.01, Loss of Offsite Power / Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer

Comment:

Assumption was made that the timeline was too vague to determine when the Emergency Diesel Generator (EDG) output breaker was closed. 4 [sic] applicants assumed that the EDG output breaker was closed in the 10 seconds between 00:05:00 and 00:05:10 and not at time 00:05:10 as the author intended. Therefore, the HPSI pump would have started by design on its 6 second block prior to 00:05:10. Therefore, accept answers A and C as both correct.

Facility Licensee Recommendation:

The station does not have plant data that would support the EDG Output breaker closing any sooner than 00:05:08, therefore, at no time in the provided information would the HPSI Breakers be closed. The station recommends no changes to question 12.

NRC Resolution:

Although the NRC did not accept the comment or the facility licensee recommendation, the answer for question 12 was changed from distractor “C” only to “A” only.

During the review of the technical accuracy of this question post-exam-administration, the NRC determined that the examination key for the exam, as administered, did not identify the correct answer. The question presented a timeline that stated the following information:

(1) following a Loss of Coolant Accident (LOCA) from 100% power at time 00:00:00, a Safety Injection Actuation (SIAS) signal actuated at time 00:03:00;
(2) at time 00:05:00, a Loss of Offsite Power (LOOP) occurred; and
(3) at time 00:05:10, both the 1A and 1B EDGs had started and output breakers were CLOSED.

The first part of the question required the applicant to determine whether or not the 1A/1B HPSI pumps were running at time 00:05:10. The initial answer key, as submitted by the facility and approved by the NRC, was that the 1A/1B HPSI pumps would NOT be running at time 00:05:10.

All parties agree that the HPSI Pumps receive an automatic start signal on a 6 second load block after the associated EDG breaker closes. It typically takes 8 to 10 seconds for an EDG to come to rated speed and voltage before the output breaker closes and loads begin to sequence on to the EDG. During a normal LOOP with a simultaneous SIAS signal present, it would be expected that the HPSI pump would start 6 seconds following the EDG breaker closure. The original answer to the question assumed that at 00:05:10, the EDG breaker had just closed and it would be an additional 6 seconds (*i.e.,* time 00:05:16) before the HPSI pumps would start.

The question, as written, stated that the SIAS signal occurred 2 minutes prior to the LOOP. The SIAS actuation will cause the EDGs to receive an automatic start signal, independent of whether or not a LOOP is present. Given the timeline presented by the question, the EDGs started and would be running at rated speed and voltage with the respective output breaker open when the LOOP occurs. In other words, at time 00:05:00, when the LOOP occurs, the EDGs did not require 10 seconds to come to rated speed. Since they are already running and ready to supply power to the deenergized busses, the EDG output breaker would close in approximately 2 seconds post-LOOP.

The NRC identified that St. Lucie Training Material, PSL OPS 0711501, Emergency Diesel Generators (rev 30), provided the following information:

> **SIAS followed by delayed LOOP**: The SIAS will immediately auto start all SIAS designated loads and start the EDG. When the UV/DV (LOOP) signal is sensed, the bus loads are shed, and after a ~ 2 second time delay, the EDG Output breaker will close and SIAS loads will sequence on. The reason for the time delay was to ensure that pump motor residual fields would collapse without causing a voltage/current spike on motor reload for those pump motor breakers that do not load shed.

Therefore, based on the available technical information, the correct timeline would read as follows:

(1) at time 00:03:00, SIAS actuation and auto-start of the 1A and 1B EDGs;
(2) at time ~00:03:10, 1A and 1B EDGs at rated speed and voltage with output breakers open;
(3) at time 00:05:00, a LOOP occurred causing bus load shed;
(4) at time ~00:05:02, 1A and 1B EDG output breakers close and SIAS loads begin to sequence on;
(5) at time ~00:05:08, 1A and 1B HPSI pumps automatically start due to +6 second SIAS load sequencer
(6) at time 00:05:10, 1A and 1B HPSI pumps <u>are</u> running due to load sequence

Section D.1.b of ES-403 of NUREG-1021 stated the following, in part: "…newly discovered technical information that supports a change in the answer key…" would be one case where it is "…most likely to result in post-examination changes agreeable to the NRC." In accordance with NUREG-1021, the newly discovered technical information supports the NRC determination that the HPSI Pumps would be running by the 00:05:10 time mark, based on the timeline given in the question; therefore, the correct answer was changed from distractor "C" only to "A" only. All applicants were evaluated against this change to the written examination answer key.

Post-Examination Comment #2: Question 15, K/A 062 AA1.07, Loss of Nuclear Service Water: Ability to operate and / or monitor the following as they apply to the Loss of Nuclear Service Water (SWS): Flow rates to the components and systems that are serviced by the SWS; interactions among the components

Comment:

The applicants understand that the Circ Water seal water system is designed to handle a loss of Intake Cooling Water (ICW) with backup from the domestic water system, it is their opinion that the Component Cooling Water (CCW) seal system would still see a small temperature change. Therefore, accept answers C and D as both correct.

Facility Licensee Recommendation:

After receiving the feedback from the applicants and discussing system design and operation with operations personnel, it is possible that the system does see some perturbation and temperature change on the Circulating Water Pump Seal coolers. Therefore, the station recommends accepting C and D as correct answers.

[*N.B.:* Following the initial receipt of the post-exam comments, NRC management requested that the facility licensee be contacted to ascertain if additional technical information could be provided on the backup cooling system to the Circulating Water Pump seal coolers. The following response was obtained.]

Amplifying Information Provided Post NRC Feedback:

Given the different operating temperatures and pressures between Intake Cooling Water and Domestic Water, and the basic manually-controlled throttling mechanism of the Circulating Water Pump seal water flow; it is reasonable to assume that after SIAS, when Circulating Water Pump seal water swaps over to the Domestic Water backup supply, the Circulating Water Pump seal water flow and temperature will change, thereby causing a change to the packing gland temperature.

The lubricating water for the pump seals and bearings comes from the Intake Cooling Water (ICW) system through a series of self-cleaning and manually cleaned strainers. Flow is monitored via a flow indicating switch at each pump and is controlled via a manually throttled ball valve. An alternate supply of lube water comes from the domestic water system, which draws suction from the City Water Storage Tanks (CWST).

During a SIAS, ICW supply to the Circulating Water Pumps will be isolated by the automatic closure of MV-21-2/-3, A/B ICW TRAIN TO TCW HXS. A low-pressure condition created from isolation of ICW will cause PCV-21-26, B/U LUBE WATER TO CW PUMPS, to OPEN and supply backup cooling water to the Circulating Water Pump seals. This backup cooling water will be of a different pressure (75 psig vs 40 psig) and temperature (74°F vs 67-88°F) than the Intake Cooling Water system, and thus change the flow and temperature of water supplying the Circulating Water Pump seals.

Additionally, 2-NOP-21.02, Circulating Water System Operations, contains instructions for aligning alternate sources of lube water to the Circulating Water Pumps. Given a potential system pressure difference between the alternate vs normal seal water source, the procedure directs throttling of seal water flow to the 6-10 gpm band and validating adequate lube water leak off from the Circulating Water Pump packing gland. During alignment of the alternate water sources, changing seal water flow may require monitoring of packing gland temperature to validate no rising temperatures. Based on these instructions, when aligning alternate sources of lube water to the Circulating Water Pumps, some seal water temperatures changes are expected.

PSL Station Recommendation: Therefore, the station still recommends accepting C and D as correct answers.

NRC Resolution

The licensee's recommendation was partially accepted; however, the NRC determined that the question should be deleted from the written examination.

The question as written presented the applicants a condition where Intake Cooling Water (ICW) was taken away from non-essential loads. The question then asked, between the Circulating Water (CW) Pump Seals and the Instrument Air Compressors (IAC), which system would experience a temperature change due to the ICW system alignment. The CW Pump seals, which are normally directly cooled from ICW, will automatically shift to a Domestic Water cooling source on loss of ICW pressure. The IACs would lose the heat sink entirely with a loss of ICW cooling. The second part question statement required the applicants to determine if the CW Pump seal coolers or the IACs "[would] experience a temperature change due to the ICW alignment." The initial answer key proposed by the facility and approved by the NRC was that the IACs will experience a temperature change due to the ICW alignment, and that the CW Pump seal coolers will not experience a temperature change due to the ICW alignment.

The station provided the NRC with temperature trends for ICW and Domestic Water. However, these temperature trends for ICW and Domestic Water were obtained using portable instrumentation, because there is no installed instrumentation at the CW pump seals to determine actual temperature. Accordingly, the station was unable to provide direct plant data if CW pump seal temperature changes based on the cooling source changing from ICW to Domestic Water. Moreover, due to potential differences in cooling water temperatures and flow rates from the two systems, it cannot be determined if the CW Pump seals will or will not have a change in temperature when shifting to the backup cooling source. Note that the facility recommendation quoted above was not technically specific concerning the temperature trend: "…it is reasonable to assume that after SIAS, when Circulating Water Pump seal water swaps over to the Domestic Water backup supply, the Circulating Water Pump seal water flow and temperature will change, thereby causing a change to the packing gland temperature. [emphasis added]"

Given the benefit of hindsight, the question would have been better written to elicit which system does, or does not, have an automatic cooling water supplied given the conditions of the question stem. It could be argued that the lack of installed station instrumentation for CW pump seal temperatures caused the question asked on the exam (which system experience[s] a temperature change) to be non-operationally valid for inclusion on an initial licensed operator examination.

Section D.1.b of ES-403 of NUREG-1021 stated the following, in part: "…a question with an unclear stem that … did not provide all the necessary information" would be one example of a question "…most likely to result in post-examination changes agreeable to the NRC."

Furthermore, the same section also stated: “If … there is no correct answer, the question shall be deleted.”

Based on the information provided by the facility licensee and the NRC evaluation, the NRC determined that not enough given information was provided in the question stem for an applicant to determine a correct answer; in fact, it was indeterminate that such an answer could be determined at all based upon a lack of installed plant equipment. In accordance with NUREG-1021, the NRC decided that in the lack of a definite correct answer, the question should be removed from the written examination. All applicants were evaluated against this change to the written examination answer key.

Post-Examination Comment #3: Question 21, K/A 032 AA2.07 Loss of Source Range NI / Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Maximum allowable channel disagreement

Comment

The question stem states that the Reactor has been determined to be critical. However, the applicants identified that the both of the Wide Range Nuclear Instruments (WRNI) indications were approximately TWO decades too low for where the Reactor would normally be while criticality is declared (normally near 1×10^{-5} %). Additionally, the information provided in the question only shows WR power rising ~ ONE decade (from 3×10^{-8} to 3×10^{-7}). This is also contrary to normal reactor behavior during an approach to criticality, where WRNI power typically rises by approximately two decades. Actual Unit 2 plant data from March of 2020 during a reactor startup shows that the Reactor went critical at a WRNI power of 1.89×10^{-5} following a rise of over two decades. This expectation has also been validated during performance of 2-PTP-81, Reload Startup Physics Testing, which showed WRNI power (after raising power two decades above critical) was recorded as 1.0 e-3, so criticality likely occurred at ~1.0 e-5.

Therefore, the response of the WR nuclear instruments would indicate that the reactor should only be half way to criticality. This would align with the response of S/U channel B (which has only shown 3 doublings). That means the response of 3 out of 4 Nuclear Instruments show the reactor only being half way to criticality, leading the applicants to determine that Startup Channel A is NOT reading correctly, and Startup Channel B IS reading correctly. Therefore, accept answers A and C as both correct.

Facility Licensee Recommendation:

The question provided the candidates with 2 sets of nuclear instrument data for the candidate to determine which set of data was expected for the conditions. The correct answer is based on an original source range value that then doubles 6 times therefore by definition, the reactor is critical. However in March of 2020, the most recent Unit 2 Startup, the Reactor went critical at 1.89 x 10-5 with nuclear instrument response on par with the candidates description. Due to the new information related to the March 2020 reactor startup, the station is recommending accepting A and C as correct answers.

NRC Resolution

The licensee’s recommendation was not accepted, and no changes were made to the written examination answer key.

The licensee recommends that both A and C are correct answers. However, it is provided in the question that, “Below is the listed initial power levels power to commence approach to criticality and the power when the reactor was announced critical”

	SU Channel BF3 (CPS)		WRNI (% Power)	
	A	B	A	B
INITIAL	50	50	3×10^{-8}	3×10^{-8}
CRITICAL	3200	400	4×10^{-7}	3×10^{-7}

This is a given statement in the question, and cannot be assumed to be an incorrect statement within the bounds of the question. In other words, based on the given information, the applicant must analyze the question keeping in mind that the reactor actually is critical.

It is expected that reactor power will double 5 to 7 times (5-7 doublings) from the beginning of the reactor start up to the point that the reactor is critical, as indicated on Source Range instruments. The final reactor power at the point of criticality is dependent on the source term, or source neutrons, present at the beginning of the reactor start up. This value can vary significantly from one reactor startup to another. The longer a reactor is shutdown, the lower the initial source term will be, based in part upon the decay of the various neutron emitting sources within the reactor core. Yet there are still 5-7 doublings of the source term, which is proportional to source range power level during a reactor start up to Criticality. The point at which a reactor can go critical can be well below the Wide Range Power indication lower scale, given the time since shutdown has been significant.

The licensee position is that distractor choice “C,” which corresponds to SR Channel B ONLY indicating as expected, is also a correct answer. But SR Channel B indication went only from 50 cps to 400 cps which corresponds to 3 doublings. This is outside the expected range for criticality, and since it is given that the reactor is critical, this indication must be incorrect or suspected as being inoperable. And since it was provided that the reactor was critical, this cannot be a correct answer.

The Wide Range indication provided in the question is low as compared to the single, recent start up that is provided by the licensee for reference. However, as stated in 2-GOP-302, Precaution 2.1.1, *Criticality shall be anticipated whenever positive reactivity additions are being made (i.e., CEA withdrawal, boron dilution, etc.).* Although the provided indications are lower than those observed during the last core start up, there are multiple variables that can affect the WR power level at critical conditions. An example of a critical reactor with an extremely low or off-scale WR value can be found during startups following extremely long shutdowns where the source term is extremely small. The 5-7 doubling to Criticality applies, even when WR are barely on scale.

Based on the above discussion, the NRC determined that the question as written was technically accurate and the question had only one correct answer. No changes were made to the answer key for question 21.

Post-Examination Comment #4: Question 32, K/A 006 K6.13 Emergency Core Cooling / Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Pumps

Comment:

The question stem doesn't contain enough information to determine if a HPSI pump restart would be allowed. One attempt to restart the HPSI pump would be allowed to protect the health and safety of the public. Since the stem provides no mention of this and it is not the ROs function to determine if health and safety of the public is at risk, and there is no other information regarding why the HPSI pump tripped, it is rational that starting the HPSI pump one time would be allowed. Therefore, accept answers A and B as both correct.

Facility Licensee Recommendation:

The stem of the question does not provide enough information for the RO candidate to determine the impact on the HPSI system and associated pump. make [sic] the decision based on health and safety of the public. Therefore, the station recommends accepting A and B as correct answers.

NRC Resolution:

The licensee's recommendation was not accepted, and no changes were made to the written examination answer key.

The licensee provided insufficient information for the basis of the comments. Safety related pumps and breakers can be restarted/reclosed following a trip, provided there is no indication that there is damage to the pump, system, or electrical components. The deterministic portion of that requirement is that the restart is to protect the Health and Safety of the Public. To determine that the plant conditions provided would lead to a direct threat to the Health and Safety of the Public required assumptions to be made outside of the conditions provided within the question. The statement that this is higher than RO level knowledge was not substantiated by the station with any procedural requirements, job description, or specific training that would support this statement.

Provided in the stem of the question was enough information that the applicant was to determine that 2-EOP-99, Figure 2, Safety Injection Flow vs. RCS Pressure, was being met for Single Full Train operation. Based on this information is to be expected that a licensed operator would recognize that this meets the minimum required safety functions per 2-EOP-03, LOCA. With Safety Functions being met, there is no immediate threat to Health and Safety of the Public. Therefore, any assumption that there was a threat to the Health and Safety of the public would be false.

Based on the above discussion, the NRC determined that the question as written was technically accurate and the question had only one correct answer. No changes were made to the answer key for question 32.

Post-Examination Comment #5: Question 99, K/A G2.4.29, Knowledge of the emergency plan.

Comment:

Based on the initial conditions in the stem of the question, a loss of Offsite Power did not occur. I know this based on SBCS being used for cooldown; therefore, the main condenser is available with the circulating water pumps running. Since there was no loss of offsite power, main feedwater would have been feeding both S/G's post trip through the low power Feedwater regulating valve restoring S/G water level towards the normal program band of 60%-70%. Then, the questions stem states that, AFAS-1 and AFAS-2 actuated. This would only occur if S/G water level in both the "A" and "B" S/G reached a level of 19% NR for a total of 210 seconds. The only way for S/G water levels to reach the value for AFAS-1 and AFAS-2 actuation, would have been as a result of losing main feedwater. Due to the SGTR being in progress, a SIAS must have occurred. When SIAS occurs, main feedwater pumps will be lost due to receiving a trip signal from SIAS. At the time Main Feedwater is lost, S/G water level will be above 19% NR (AFAS-1 & AFAS-2 actuation setpoint).

The RCS Heat removal safety function per 2-EOP-04 is as follows [picture of 2-EOP-04 p. 58 included]: …

The RCS heat removal safety function requires that either, an unisolated S/G level is between 60%-70% NR with Tcold stable or lowering, or feedwater is being controlled to restore the unisolated S/G level to between 60%-70%, with Tcold stable or lowering. Once Main Feedwater is lost, the RCS heat removal safety function is not being met, and immediate action is required to restore the RCS safety function.

ADM 11.16 specifically states: "If the safety function status check acceptance criteria are NOT met for a particular safety function, the operator should immediately report this to the Unit Supervisor, then the operating crew should take appropriate contingency actions necessary to restore the safety function."

Depending on how quickly the crew starts AFW (Auxiliary Feedwater) and what S/G water levels are at, they may have to limit themselves to 150gpm for 5 minutes per the hard card. This could cause S/G levels to continue to lower for the first five (5) minutes, due to the high steaming rate in effect to cooldown the RCS to less than five hundred ten (510) degrees F for S/G isolation. The lowering level could result in S/G levels going below 19% narrow range for sufficient time to allow AFAS to time out and actuate, as stated in the stem, even after AFW is manually started by the crew.

The desk RCO will restore S/G water level IAW 2-NOP-99.07 Operations Hard Cards Attachment 5: the hard card has the operator close the steam admission valve to the "C" AFW Pump from the ruptured S/G [picture of 2-NOP-99.07 p. 18, 19, and 20 included].

Due to restoring S/G water level IAW with the operations hard card, the steam admission valve from the affected S/G to the "C" AFW pump would be closed by the time AFAS 1 and AFAS 2 would have actuated. Although, not starting the 2C AFW pump, step 3A is performed by the operator to prevent the steam admission valve from automatically opening on a subsequent AFAS. [sic] Actuation. Both motor driven AFW pumps are available, therefore, there is no reason to use the 2C AFW pump. By steaming the ruptured S/G to the main condenser and with the steam admission valve to the “C” AFW pump being closed, no release is currently in progress.

Additionally, ADM-11.16 [CAUTION statement] states: “IF Main Feedwater is lost and AFAS is left to automatically actuate, damage to the S/G feed ring will occur. Prompt restoration of Steam Generator levels using Auxiliary Feedwater is allowed in order to satisfy the Safety Function prior to formally addressing the entire RCS Hear removal Safety Function.” The allows the crew to restore Steam generator water level with Auxiliary Feedwater.

IAW EPIP-02 a release is defined as follows: “ … A Steam Generator, with primary-to-secondary leakage, due to a tube leak or rupture, is vented to atmosphere; Operating the C AFW pump with steam being supplied from a Steam Generator with primary to secondary leakage due to a tube leak or rupture. Then a RELEASE is in progress.” The ruptured S/G is not being vented to atmosphere, and the “C” AFW pump steam admission valve from the ruptured S/G would have already been closed in accordance with Plant procedures. Therefore, a release is NOT in progress.

Facility Licensee Recommendation:

PSL Station Recommendation: Based on the comment submitted by the Applicant, the Station agrees with the applicants statement that due to the information provided in the steam of question 99, the applicant, while following the guidance provided to the applicant during the NUREG- 1021, Appendix E briefing, which states: If you have any questions concerning the intent or the initial conditions of a question, do not hesitate to ask them before answering the question. Note that questions asked during the examination are taken into consideration during the grading process and when reviewing requests for informal NRC staff reviews (appeals). Ask questions of the NRC examiner or the designated facility instructor only. A dictionary is available if you need it.

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. For example, you should not assume that any alarm has activated unless the question so states or the alarm is expected to activate as a result of the conditions that are stated in the question. Similarly, you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise. Finally, answer all questions based on actual plant operation, procedures, and references. If you believe that the answer would be different based on simulator operation or training references, you should answer the question based on the actual plant.

The assumptions made by the candidate are consistent with the guidance provided.

Therefore, the station recommends accepting A and C as correct answers.

NRC Resolution:

The licensee’s recommendation was not accepted; no change was made to the answer key for question 99.

The question as written provided the following given information in bulleted format, not timeframe format:

(1) Unit 2 is experiencing a SGTR
(2) 2-EOP-04 SGTR is in progress
(3) Initial Cooldown for S/G isolation is in progress using SBCS
(4) AFAS-1 and AFAS-2 have actuated

Then, given the above information, the first part question statement asked the applicant to evaluate if "IAW EPIP-02, Duties and Responsibilities of The Emergency Coordinator, a release (is or is NOT) currently in progress." The contention therefore resolves itself into two different determinations: (1) what is meant by the term "currently" in the question statement; and then (2) based on what "currently" refers to, is the (steam-driven) 2C AFW pump being operated while supplied from a Steam Generator with primary to secondary leakage due to a tube leak or rupture. As correctly noted by the applicant contention quoting 2-EPIP-02 above, if the 2C AFW pump is running and yet-to-be-isolated from the ruptured S/G at the "currently" point of the question statement, a release is in progress; if the 2C AFW pump is not running or has been isolated from the ruptured S/G at the "currently" point of the question statement, a release is NOT in progress. We now examine these points in turn.

The question, as written, did not provide a timeline, or sequence of events associated with the plant conditions; rather, it provided an applicant a list of current plant conditions as they applied to the question, at a given point in time. The applicant is informed that a SGTR is in progress, that operators are performing procedure 2-EOP-04, STEAM GENERATOR TUBE RUPTURE SGTR, and that initial cooldown for S/G isolation is in progress using SBCS. Although we do not know a specific time for the above conditions, the information provided means that the operators performing the initial cooldown are somewhere between step 10 of EOP-04 and step 17 of EOP-04. The high-level action statements of these steps are as follows:

10. INITIATE lowering RCS Thot to less than 510 °F using SBCS.
11. DEPRESSURIZE the RCS in preparation for isolating the affected S/G:
12. WHEN permissive conditions are MET during a controlled cooldown, THEN BLOCK automatic MSIS and SIAS actuation signals as follows:
13. VERIFY circulating water flow to the main condenser.
14. STABILIZE the Secondary Plant per Appendix X, Secondary Plant Post Trip Actions, Section 2
15. IF a LOOP has occurred, THEN PERFORM the following….
16. DETERMINE the MOST affected S/G by evaluating the following indications:
17. WHEN RCS Thot is less than 510 °F, THEN ISOLATE the MOST affected S/G per Appendix R, Steam Generator Isolation.

Therefore, given the information provided in the stem, the applicant can "place" the operators as performing the initial cooldown, somewhere between steps 10 and 17 of EOP-04. This is what is meant by the term "currently" in the first part question statement.

The final bulleted information in the given statement of the question is that both AFAS-1 and AFAS-2 interlocks have actuated. Note that no other information is given concerning any additional operator actions taken to reset the AFAS signals or isolate the "C" AFW pump steam supply from the ruptured S/G. Recall that Appendix E of NUREG-1021 stated: "…you should assume that no operator actions have been taken, unless the stem of the question or the answer choices specifically state otherwise." A correct application of this section of Appendix E is that no other operator actions have been taken except those that would have been directed in steps 10-17 of EOP-04. To assert otherwise would be to be making an assumption unsupported by Appendix E requirements or the given information of question 99.

Therefore, the correct application of the given conditions of the question is that AFAS-1 and AFAS-2 actuated at some point before where the operators are now in the procedure flowpath, that is, somewhere between steps 10-17 of EOP-04. No other operator actions can be assumed to have occurred. Because AFAS-1 and AFAS-2 actuation would have automatically started the “C” AFW Pump, and because there are no operator actions in steps 10-17 to isolate the “C” AFW Pump steam admission valves from the ruptured S/G, then the technically correct determination is that a release is “currently” in progress as defined by procedure EPIP-02.

The contention above asserts that the operators would be using the “operations hard card” procedure, 2-NOP-99, and completing various actions that were not specifically directed by the EOP steps. This assertion cannot be considered correct for several reasons: (1) this is an unsupported assumption as detailed in Appendix E quoted above; (2) the EOP steps never direct the operators to take action in accordance with 2-NOP-99; and (3) moreover, EOP-04 step 17 specifically directs the operators to isolate the most affected S/G using Appendix R (not NOP-99). In other words, the EOP requires the operators to complete the initial cooldown before S/G isolation, and to conduct the isolation using Appendix R, not the “operations hard card” (2-NOP-99).

The applicant’s assumptions are contrary to the given conditions of an AFAS actuation with no other operator action. AFAS Actuation coupled with the other given plant conditions is operationally valid in that given a large enough Steam Generator Tube Rupture (SGTR) that the steam generators (SGs) would reach the AFAS setpoint and compete the 240 second timeout to actuate prior to crew action. This was observed during simulator scenarios during this exam that mirrored the conditions as given in this question. Although the NRC agrees with some of the assertions in the above comment, and disagrees with others, it is sufficient to have shown that they do not have a material bearing on the correct logical application of the NUREG-1021 requirements to this question.

To summarize again: it was a given condition in the question that AFAS-1 and AFAS2 had actuated, this is an operationally valid occurrence during a SGTR, therefore 2C AFW Pump had to be running since there were no cues in the question to indicate otherwise. With the 2C AFW Pump running and the steam admission valve from the affected SG with a SGTR still open, this is by the definition of EPIP-02, a Release to the environment and would be required to be documented as such on the State Notification Form. A release was “currently” in progress.

Since a release cannot be simultaneously in progress and not in progress, there cannot be two correct answers to the question. The NRC determined that the question, as written, was technically accurate and the question had only one correct answer.

Post-Examination Comment #6: Job Performance Measure JPM A-2S and A-2R, Perform RCS inventory Balance for the RO and SRO Candidates,

Comment

During administration of PSL L-20-1 NRC A-2S and A-2R; Perform RCS Inventory Balance for the RO and SRO Candidates, an unintended typographical error was identified which resulted in a math error in the JPM key. The station has corrected the typographical error in the calculation in the JPM.

Station Recommendation: It is the stations request to update the two JPM's. [sic]

NRC Resolution

The licensee's recommendation was accepted.

The typographical/arithmetic errors that were listed in the JPM key were corrected, and all applicants were evaluated against the correct mathematical determination of the RCS leak rate.

Post-Examination Comment #7: Job Performance Measure JPM S-7, Start Containment Purge/Respond to High Radiation- Unit 2

Comment

The JPM directs the candidate to perform a Containment Purge for Refueling Operations, in accordance with 2-NOP-06.20, beginning with Section 4.2.1, Step 5. Performance Step 3 of the JPM is identified as a critical step and states, "ensure the Purge Mode selector switch is in the Refuel position prior to fuel movement." This is contrary to 2-GOP-365, Refueling Sequencing Guidelines. The Purge Mode selector switch position, in support of refueling, is maintained/ aligned per 2-GOP-365, Refueling Sequencing Guidelines, Section 4.0, step 31. Specifically, step 31 states, "ensure the following is complete prior to performing 0-NOP-67.05, Refueling Operation. Ensure the Purge Mode Selector switch is selected to the Refuel position". During refueling operations at PSL, this Purge Mode selector switch is positioned prior to refueling in accordance with the procedure for Refueling Sequencing Guidelines.

Station Recommendation: It is the stations position that reference to the purge mode selector switch in the task standard be removed from the JPM and performance step 3 be changed to not critical.

NRC Resolution

The licensee's recommendation was accepted.

Based on 2-GOP-365 procedural guidance that would ensure the Purge Mode Selector Switch was in the proper position prior to fuel movement, the NRC concurs that step 2-NOP-06.20, Section 4.2.1, Step 5 should not be a critical step for JPM S-7. The NRC agrees that acceptable performance for JPM step 5 would either consist of positioning the Purge Mode selector switch to REFUEL, or by verbalizing that the Purge Mode selector switch would be re-positioned before commencing moving fuel in the Reactor. All applicants were evaluated against this change to the JPM answer key.

SIMULATOR FIDELITY REPORT

Facility Licensee: St. Lucie Nuclear Plant

Facility Docket No.: 05000335, 05000389

Operating Test Administered: December 7 – 12, 2020

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.