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

January 12, 2018

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, VA 23060-6711

SUBJECT: SURRY NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000280/2017301 AND 05000281/2017301

Dear Mr. Stoddard:

During the period October 30 – November 3, 2017, the Nuclear Regulatory Commission (NRC) administered operating tests to employees of your company who had applied for licenses to operate the Surry 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 November 8, 2017.

All ten applicants passed the operating test, while nine applicants passed the written examination. There were 17 post-administration comments concerning the written examination. These comments, and the NRC resolution of them, are summarized in Enclosure 2. A Simulator Fidelity Report is included in this report as Enclosure 3.

The initial written examination submitted by your staff met the guidelines for quality contained in 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: 50-280, 50-281
License Nos: DPR-32, DPR-37

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

cc: Distribution via Listserv

SUBJECT: SURRY NUCLEAR PLANT – NRC OPERATOR LICENSE EXAMINATION REPORT 05000280/2017301 AND 05000281/2017301

Distribution:
M. Donithan, RII
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J. DeMarshall, NRO

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NAME	MDonithan	PCapehart	JDeMarshall	GMcCoy		
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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket No.:	50-280, 50-281
License No.:	DPR-32, DPR-37
Report No.:	05000280/2017301 and 05000281/2017301
Licensee:	Dominion Energy
Facility:	Surry Nuclear Plant, Units 1 and 2
Location:	5850 Hog Island RD Surry, VA 23883
Dates:	Operating Test – October 30 - November 3, 2017 Written Examination – November 8, 2017
Examiners:	M. Donithan, Chief Examiner, Operations Engineer P. Capehart, Senior Operations Engineer J. DeMarshall, Operations Engineer
Approved by:	Gerald J. McCoy, Chief Operations Branch 1 Division of Reactor Safety

SUMMARY

ER 05000280/2017301, 05000281/2017301; October 30 - November 3, 2017 & November 8, 2017; Surry Nuclear Plant; Operator License Examinations.

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

Members of the Surry Nuclear Plant staff developed both the operating test and the written examination. All met the quality guidelines contained in NUREG-1021.

The NRC administered the operating test during the period October 30 – November 3, 2017. Members of the Surry Nuclear Plant training staff administered the written examination on November 8, 2017. Three Reactor Operator (RO) and seven Senior Reactor Operator (SRO) applicants passed the operating test; two ROs and seven SROs passed the written examination. Nine applicants were issued licenses commensurate with the level of examination administered.

There were 17 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 job performance measures (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 October 30 – November 3, 2017. NRC examiners evaluated three RO and seven SRO applicants using the guidelines contained in NUREG-1021. Members of the Surry Nuclear Plant training staff administered the written examination on November 8, 2017. Evaluations of applicants and reviews of associated documentation were performed to determine if the applicants, who applied for licenses to operate the Surry 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 Surry Nuclear Plant training staff developed both the operating test 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 written examination and operating test submittals were within the range of acceptable quality for a proposed examination specified by NUREG-1021.

No issues related to examination security were identified during preparation and administration of the examination.

Nine applicants passed both the operating test and written examination and were issued licenses. One RO applicant failed the written examination and was denied a license.

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 17 post-examination comments concerning the written examination, of which 10 were suggestions by the applicants to modify the answer key. The facility agreed with only 2 of these, and of those recommended changes, only 1 resulted in an answer key change after NRC review. A copy of the final written examination and answer key, with all changes incorporated, may be accessed not earlier than November 8, 2019, and a copy of the licensee's post-examination comments, may be accessed in the ADAMS system (ADAMS Accession Numbers ML17354B289, ML17354B295 and ML17354B287).

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 3, 2017 the NRC examination team discussed generic issues associated with the operating test with Fred Mladen, Site Vice President, and members of the Surry 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
Fred Mladen, Site Vice President
Terri Cuthriell, Licensing Engineer
Mike Haduck, Manager Nuclear Outage & Planning
Ron Herbert, Engineering Manager
Carl Irwin III, Supervisor Nuclear Training
Randy Johnson, Operations Manager
Rich Philpot, Training Manager
John Rosenberger, Director, Director Nuclear Safety and Licensing
Jim Shell, Assistant Operations Manager-Training
David Wilson, Maintenance Manager

FACILITY POST-EXAMINATION COMMENTS AND NRC RESOLUTIONS

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 ML17354B287.

Item #1: RO Written Exam Question #1

Post-Examination Comment

One applicant contended that there was some ambiguity with the term "shutdown margin" vice reactivity. He did not suggest that another answer should be accepted, or that the designated correct answer was not correct.

Given the following:

- A Reactor Trip occurs on Unit 1 from 100% power.
- 1-ES-0.1, Reactor Trip response, has been implemented.
- Control Bank D Rod H14 IRPI indicates 218 steps.
- All other control rods indicate 0 steps.
- T_{AVG} is 547 ^{o}F and stable.

Which ONE of the following completes the statements below?

1) One minute after the trip, shutdown margin is rising due to the decay of __(1)__.
2) In accordance with 1-ES-0.1, Reactor Trip Response, Emergency Boration __(2)__ required.

A. 1) Iodine 2) is NOT

B. 1) Iodine 2) is

C. 1) Xenon 2) is

D. 1) Xenon 2) Is NOT

Answer Key correct answer: A

The facility's position was that the question was adequate as written.

NRC Resolution

This was a Modified Bank question from the V.C. Summer 2013 NRC exam, and that question also used the term "shutdown margin". Shutdown margin in the given circumstances is a measure of how much negative reactivity is in the reactor core. As Iodine decays it becomes Xenon, a powerful neutron absorber which adds more negative reactivity to the core, increasing the margin by which the reactor is shut down. Use of the term shutdown margin is technically acceptable in this question.

No answer key change is required.

Item #2: RO Written Exam Question #8

Post-Examination Comment

An applicant asked if 50°F was a potential subset of the answer, 30°F.

The Crew is responding to a SGTR in accordance with 1-E-3, Steam Generator Tube Rupture.

- The Rapid RCS cooldown has been completed.

Which ONE of the following describes:

1) The subcooling value for RCS depressurization termination.
2) The purpose for the RCS depressurization.

A. 1) 50 °F.
   2) Minimize break flow and refill the pressurizer.

B. 1) 50 °F.
   2) Minimize Reactor Vessel stress for PTS concerns.

C. 1) 30 °F.
   2) Minimize break flow and refill the pressurizer.

D. 1) 30 °F.
   2) Minimize Reactor Vessel stress for PTS concerns.

Answer Key correct answer: C

The facility determined that there was no subset issue and the question was adequate as written.

NRC Resolution

This is not a subset issue because of the wording of procedure 1-E-3 Step 19.c: RCS depressurization is to continue until subcooling is "LESS THAN 30°F". If the depressurization were stopped when subcooling reached 50°F (and getting smaller), it would be stopped too early, and the Westinghouse Owners Group Basis strategy for the step would not have been met.

No change to the answer key is required.

Item #3: RO Written Exam Question #13

Post-Examination Comment

One applicant asked if there was a Plant Computer System (PCS) alarm to support to support the 70% distractor.

Initial Condition

- Unit 1 is operating at 100% power.
- Unit 2 is in RSD with core off-load to the SFP complete 10 hours ago.

Current Condition

- A tagging error results in isolation of SW flow to all CC HXs.
- CC surge tank level is 55% and rising.
- CC HX Disch Temp is 76°F and rising.

Which ONE of the following describes:

1) CC Surge Tank High Level alarm setpoint is ___(1)___.
2) The CC Surge Tank Vent Valve ___(2)___ automatically close on high level.

A. 1) 93%
   2) will not

B. 1) 93%
   2) will

C. 1) 70%
   2) will not

D. 1) 70%
   2) will

Answer Key correct answer: A

The facility noted that there is no PCS alarm for this function, and the question was adequate as written.

NRC Resolution

The licensee was asked whether PCS displays CC Surge Tank level. The answer was that CC Surge Tank level does not input to PCS at all.

No change to the answer key or grading is required.

Item #4: RO Written Exam Question #15

Post-Examination Comment

Three applicants commented that the question stem was confusing where under Current Conditions the 2nd bullet stated that "Unit 2 parameters are comparable."

Initial Conditions:
- Unit 1 and Unit 2 are operating at 100%.
- The plant has been notified by SOC that there are significant grid instabilities due to numerous base load plants being out of service.
- The SOC has requested maximum power generation from both units.

Current Conditions:
- The SOC reports that the 500 KV system is most impacted and the Emergency Low Limit for that network has been reached.
- The BOP reports the following Unit 1 changes (Note: Unit 2 parameters are comparable):

	GEN MWe	**GEN MVARS**	**Grid Freq**	**Gen Amps**	**Gen Volts**	**Gen H2 Press**
Initial	907	+ 10	60 Hz	23,500	22.3 KV	75 psig
Current	1020	- 210	58.0 Hz	27,500	21.6 KV	75 psig

Which ONE of the following completes the statements below?
1) Based on the information given the Reactor __(1)__ required to be tripped.
2) Which Emergency buses are most affected by the current conditions stated above?

(REFERENCE PROVIDED)

A. 1) is 2) Buses 1J and 2H.

B. 1) is not 2) Buses 1J and 2H.

C. 1) is not 2) Buses 1H and 2J.

D. 1) Is 2) Buses 1H and 2J.

Answer Key correct answer: A

The facility noted that the current conditions indicate a grid frequency below the RCP breaker trip setpoint, so the question is adequate as written

NRC Resolution

The statement was simply meant to indicate that Unit 2 conditions were similar, without having to provide a specific data table for Unit 2 to this already busy question. The conditions given for Unit 1 clearly indicate that a reactor trip should have already occurred on bus low frequency.

No grading change is warranted.

Item #5: RO Written Exam Question #18

Post-Examination Comment

One applicant noted that Lesson Plan ND-95.3-LP-41 and a Caution in procedure FR-H.1 state: “If feed flow is reduced due to operator action to minimize feed flow as instructed in these guidelines and the capability of providing the minimum feed flow is available, then a transfer from these procedures is not required and FR-H.1 should not be performed.”

Initial Conditions:

- Unit 1 reactor operating at 100% power.
- The reactor is tripped and safety injection actuated.
- A steam leak is reported in Unit 1 Safeguards.

Current Conditions:

- The Crew has entered ECA-2.1, Uncontrolled Depressurization of All Steam Generators.
- All SG NR levels are off-scale low.
- RCS Cooldown rate is 155°F/hour.

Which ONE of the following states:

1) What is the MINIMUM AFW flow (gpm) to each SG?
2) Will a transition to 1-FR-H.1 be required after AFW has been throttled?

A. 1) 60
2) No

B. 1) 100
2) No

C. 1) 60
2) Yes

D. 1) 100
2) Yes

Answer Key correct answer: C

The facility’s position was that it is correct that FR-H.1 should not be performed, but the crew is still required to transition to FR-H.1 to read the caution, and then exit FR-H.1. Furthermore, license candidates are trained to transition to the appropriate FR and then exit if appropriate, so the question was adequate as written.

NRC Resolution

Per procedure rules of usage, an Orange or Red terminus on the Critical Safety Function Status Trees must transition to a functional recovery procedure. FR-H.1 recognizes that other Emergency Operating Procedures may have purposely lowered feedwater flow, and takes this into account with the Caution prior to Step 1. But you have to be “in” FR-H.1 for that Caution to be applicable.

No change to the grading is required.

<u>Item #6</u>: RO Written Exam Question #21

<u>Post-Examination Comment</u>

1) One applicant stated that the labeling for choices A2 and C2, “Vent Vent 1 Gas” was confusing, and that operators don’t call it that.
2) A different applicant asked if flow couldn’t have been aligned to Vent Stack #1.

Given the following:

- Fuel movement is in progress in the Fuel Building.
- Fuel Handlers report an assembly has been dropped and appears to be damaged.
- 1-RM-RI-153, FUEL PIT BRIDGE, HIGH Alarm is received.
- The Crew has initiated 0-AP-22.00, Fuel Handling Abnormal Conditions.

Which ONE of the following identifies:

1) The MCR must be ISOLATED within ___(1)___ minutes.
2) The release is monitored using ___(2)____.

A. 1) 2
2) 1-VG-RI-104, Vent Vent 1 Gas

B. 1) 2
2) 1-VG-RI-131 A/B/C, Vent #2 Gas/Particulate

C. 1) 60
2) 1-VG-RI-104, Vent Vent 1 Gas

D. 1) 60
2) 1-VG-RI-131 A/B/C, Vent #2 Gas/Particulate

Answer Key correct answer: B

The facility’s position was:
1) This is how the instrument is labeled in the plant.
2) Flow could be aligned to Vent stack #1 but there was nothing in the question to indicate a different alignment than the <u>normal</u> alignment.
No change required, the question is adequate as written.

<u>NRC Resolution</u>
1) Whether “Vent Vent 1 Gas” was confusing or not, the equipment designator “1-VG-RI-104” specifies the instrument exactly.
2) Applicants are to assume <u>normal alignment</u> for all systems *unless specifically directed otherwise by the question.* This was covered in the “Policies and Guidelines for Taking NRC Examinations” brief that was held with the applicants in accordance with NUREG 1021 Appendix E, therefore the answer should be based on the normal alignment of the ventilation system.

A grading change is not justified.

Item #7: RO Written Exam Question #24

Post-Examination Comment

One applicant noted that Attachment 1, System Alignment Verification, also has direction to check the position of 1-RM-TV-100C.

Given the following: • Unit 1 is operating at 100% power. • The reactor is tripped and SI actuated. • Containment pressure is 25 psia and rising slowly. Which ONE of the following completes the statement: 1-RM-TV-100C, CTMT ATMOS RM INLET I/S TV, closes on a ___(1)___ containment isolation signal, and is checked in the required position using ___(2)___ of 1-E-0, Reactor Trip or Safety Injection. A. 1) Phase I 2) Attachment 1, System Alignment Verification B. 1) Phase I 2) Attachment 4, CLS Component Verification C. 1) Phase II 2) Attachment 4, CLS Component Verification D. 1) Phase II 2) Attachment 1, System Alignment Verification Answer Key correct answer: C

The facility's position was that while Attachment 1 step 8 Response Not Obtained (RNO) does check the positions of 1-RM-TV-100A/B/C, this is only done if step 8a is answered as "NO". The question stem puts containment pressure at 25 psia, which is above the step 8a setpoint of 23 psia. Therefore step 8 RNO would not be performed. The question was adequate as written, no change required.

NRC Resolution

Attachment 1 does check the position of 1-RM-TV-100C, but that RNO step isn't reached with the containment conditions given in the question. The question was broadly trying to test whether applicants knew that Att. 1 checks Phase I valves, while Att. 4 checks Phase II & III valves. The intent was not to be tricky by having the subject valve in both Attachments, and from the question-development documentation it doesn't appear that the facility exam writers or the Chief Examiner noticed that 1-RM-TV-100C was even in Att. 1.

Because the question is technically correct, no change to the grading is necessary.

Item #8: RO Written Exam Question #28

Post-Examination Comment

One applicant asked why T_{AVE} was lower, and couldn't it be higher instead.

Given the following:

- Unit 1 is operating at 29% power.
- 1-RC-P-1A breaker spuriously trips open on overcurrent.

With no operator actions which of the following describes:

1) How Loop 'A' Tavg will change as compared to previous Tave Loop 'A'.
2) The reason for the Loop 'A' Tavg change?

A. 1) Lower 2) due to no forced flow in the loop A.

B. 1) Lower 2) due to reverse flow in the loop A.

C. 1) Higher 2) due to reverse flow in the loop A.

D. 1) Higher 2) due to no forced flow in the loop A.

Answer Key correct answer: B

The facility noted that T_{AVE} is lower because at this power level 'B' and 'C' loops will provide load demand (the reactor would not have tripped). T_{AVE} is lower because as flow is reversed, T_{COLD} rises, T_{HOT} lowers, and T_{COLD} is greater than T_{HOT}. They ran this scenario on the classroom simulator obtained those results. Question is adequate as written, no change required.

NRC Resolution

The source for this question was the 2008 Diablo Canyon NRC exam, which had the same "T_{AVE} will be lower" answer. This is a fundamental concept for Pressurized Water Reactors, and a fairly common exam question. The facility's description of the thermodynamic behavior is correct, and is supported by the results from their glass-top simulator.

No question or grading changes are warranted.

Item #9: RO Written Exam Question #31

Post-Examination Comment

One applicant thought the stem of the question was confusing by using the term "Station Instrument Air" in the last bullet of the Initial Conditions.

Initial Conditions:

- Unit 1 is in CSD with the PRZR solid.
- 1-RC-P-1C, "C" RCP running for crud burst cleanup.
- RCS temperature is 158 °F and lowering.
- RCS pressure is 305 psig.
- Station Instrument Air is supplying containment, the CMT IA compressors are in OFF.

Current Conditions:

- Unit 1 Instrument Air header ruptures.

Which ONE of the following completes the statements below:
1) RCS pressure will ___(1)___.
2) Procedure ___(2)___ is used to mitigate the effects of this transient.

A. 1) rise
2) 1-AP-27.00, Loss of Decay Heat Removal Capability

B. 1) lower
2) 1-AP-27.00, Loss of Decay Heat Removal Capability

C. 1) rise
2) 0-AP-40.00, Non-Recoverable Loss of Instrument Air

D. 1) lower
2) 0-AP-40.00, Non-Recoverable Loss of Instrument Air

Answer Key correct answer: C

The facility stated that the question was adequate as written.

NRC Resolution

Containment Air at Surry is normally provided by dedicated air compressors adjacent to the containment building, which draw air from containment and return it to containment air systems. But as given in the Initial Conditions, Unit 1 was in Cold Shutdown and these air compressors were off, so in that situation the air supply is from outside containment.

The terminology "Station Instrument Air" is used throughout Lesson Plan ND-92.1-LP-1, Station Air Systems, specifically at D.4.(d) on page 27:

> In the event that containment instrument air compressors cannot supply the system needs, the header can be kept pressurized by the Station Instrument Air System.

This is exactly the situation described in the question. So the normal Containment Air system is not in service, and now its backup system can't maintain air pressure due to a header rupture.

No grading modification is necessary.

Item #10: RO Written Exam Question #42

Post-Examination Comment

One applicant asked if it was true that T_{AVE} would rise, and referred to IMP IN vs. IMP OUT.

Given the following:

- Unit 1 Turbine Control is in IMP IN.
- Control rods are in MANUAL.
- The following parameters are given:

Rx. Pwr	Delta T	Tave	T ref.	Gen MWe
89%	90%	569.7	569.7	784

- 1
  -MS-SOV-104, MSR Steam Supply fails CLOSED.

With no operator action which ONE of the following parameters would be higher when steady-state conditions are reached?

A. Reactor Power.
B. Generator MWe.
C. T ref.
D. Tave.

Answer Key correct answer: D

The facility re-ran the question scenario on the classroom simulator, which supported T_{AVE} rising once steady-state conditions were reached. No change in the question required.

NRC Resolution

When this question was submitted for approval, its supporting documentation included “before” and “after” data obtained from the glass-top simulator. That data shows T_{AVE} rising from 569.7°F to 574.8°F, a change of 5.1°F, which is quite significant.

No changes to the question or grading are warranted.

Item #11: RO Written Exam Question #53

Post-Examination Comment

Two applicants contended that answer choice 'B' is also correct, because the sub-steps in Abnormal Procedure 0-AP-12.00, Service Water System Abnormal Conditions, Step 4 RNO are not bulleted and therefore can be performed in any order.

Initial Conditions:

- 1-SW-P-10A, CHG Pump SW Pump, is running in HAND
- 1-SW-P-10B, CHG Pump SW Pump is in AUTO and OFF.
- 1D-G5, SW or CC PPS DISCH TO CHG PPS LO PRESS has just alarmed.

Current Conditions (2 minutes later):

- 1-SW-P-10A, CHG Pump SW Pump, continues running in HAND.
- 1-SW-P-10B, CHG Pump SW Pump is in AUTO and OFF.
- The Service Building Inside Operator reports the following readings:

Parameter	**Previous**	**Current**	**Log spec**
1-SW-DPI-100A, 1-VS-S-1A, SW Strainer D/P	0.8 psid	0.8 psid	Max: 1.5 psid
1-SW-DPI-27, 1-SW-S-2A, SW Suct. Strainer D/P	0.2 psid	1.8 psid	Max: 1.5 psid
1-SW-PI-140A, 1-SW-P-10A, Suction press	3.8 psig	0.8 psig	Min: -5 psig
1-SW-PI-26, 1-SW-P-10A, discharge pressure	27.5 psig	6 psig	Min: 15 psig

Which of the following describes the actions the operator must take per 0-AP-12.00, Service Water System Abnormal Conditions?

A. Start 1-SW-P-10B, place duplex strainer 1-SW-S-10 in service, and open 1-SW-263 Suction Cross-tie.

B. Place 1-SW-S-2A, suction strainer standby basket in service, and vent 1-SW-P-10A.

C. Start 1-SW-P-10B, stop and vent 1-SW-P-10A, place 1-SW-S-2A, suction strainer standby basket in service.

D. Place additional SW header in service, vent all three SW headers, vent 1-SW-P-10A Chg SW pump.

Answer Key correct answer: C

The facility agreed that choice 'B' is also correct, and provided the following marked-up page from the procedure, with subsequent justification:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	CHECK SW PARAMETERS - NORMAL (Continued)	☐ f) WHEN SW header is greater than 0 psig, THEN vent and start a CHG pump SW pump.
	A	IF SW header(s) NOT in a vacuum, THEN perform the following:
		☐ a) Check running or Start standby CHG pump SW pump(s).
		☐ b) Stop and vent the affected pump(s).
	B	☐ IF a CHG pump SW pump has low suction pressure due to a clogged suction strainer, THEN place standby basket In Service AND vent pump.
		☐ IF system leakage suspected, THEN inspect Aux Bldg piping.

Sub-steps (marked “A” and “B” above) are of the same level of indent and although “A” is the more appropriate action, “B” contains corrective actions for this scenario. It would be reasonable for a team, performing a scenario with the given conditions, to choose either the “A” or “B” path.

<u>NRC Resolution</u>

Placing the standby strainer basket in service would correct the issue presented by the question, and is procedurally allowed (the “B” option above). It appears that the question writers and Chief Examiner were focused on pump 1-SW-P-10B not auto-starting as designed, and the most direct and quickest way to remedy the situation would be to start the ‘B’ CHG Pump SW Pump from the main control room (the “A” option above). But locally swapping strainer baskets, while taking more time, would restore normal operating conditions to the ‘A’ pump, so the start of ‘B’ would no longer be necessary.

Therefore, Answer Choice ‘B’ is <u>also</u> correct, so **two correct answers** were accepted for Question 53: ‘B’ <u>and</u> ‘C’

<u>Item #12</u>: RO Written Exam Question #57

<u>Post-Examination Comment</u>

One applicant asked if pulling <u>input</u> fuses would also allow 1-CH-LCV-1460A&B to be opened, making 'B' also a correct answer.

Given the following:

- A steam generator tube rupture has occurred.
- Due to multiple failures, the PRZR spray and PORVs are unavailable to depressurize the RCS.
- PRZR level is offscale low.
- The Crew is attempting to re-establish Letdown flow in accordance with ECA-3.3, SGTR with loss of Pressure Control.

In accordance with 1-ECA-3.3, which ONE of the following identifies:

1) The ______ fuses for LC-1-460C and LC-1-459C are removed.
2) 1-CH-LCV-1460A and B _______.

A. 1) input
2) open automatically

B. 1) input
2) are opened manually

C. 1) output
2) open automatically

D. 1) output
2) are opened manually

Answer Key correct answer: D

The facility's position was that ECA-3.3 specifically states to pull the <u>output</u> fuses, so the question was adequate as written.

<u>NRC Resolution</u>

The Chief Examiner asked the same question during question development. The answer from the facility was that removing the <u>input</u> fuses would cause the Pressurizer low level signal to be locked in, and letdown valves 1-CH-LCV-1460A & B could not be opened, so Letdown flow could not be re-established.

And regardless of what physically happens, the question asks for which set of fuses is removed *in accordance with 1-ECA-3.3*, and Attachment 8 Step 1 clearly directs removing the <u>output</u> fuses (with "OUTPUT" being capitalized).

Furthermore, the facility advised during question development that this action is trained during both Classroom and Simulator training, and that there is a Job Performance Measure covering the fuse-removal task, with failure warranted if the applicant incorrectly removes the <u>input</u> fuses.

No answer key change is required for this question.

Item #13: RO Written Exam Question #62

Post-Examination Comment

One applicant asked if choice "C" would also be correct, because at this vacuum wouldn't steam dumps be lost.

Given the following:

- Unit 1 is operating at 100%.
- Condenser Vacuum is 27.6 inches Hg and is lowering 0.4 inches Hg/minute.
- Turbine 1 and 2 operators dispatched to look for leaks.
- Generator Megawatts are 890 Mwe and lowering rapidly.
- Annunciator 1A-G1, Traveling Screens Hi Diff Lvl is Lit.

Which of the following completes the following statements:

1) __(1)__ is required to be entered.
2) 1-E-0 is required to be entered if Main Condenser Vacuum lowers to __(2)__.

A. 1) 0-AP-12.01, Loss of Intake Canal Level
   2) 25.0 in-Hg

B. 1) 0-AP-12.01, Loss of Intake Canal Level
   2) 22.5 in-Hg

C. 1) 1-AP-14.00, Loss of Main Condenser Vacuum
   2) 25.0 in-Hg

D. 1) 1-AP-14.00, Loss of Main Condenser Vacuum
   2) 22.5 in-Hg

Answer Key correct answer: D

The facility's position was that 1-AP-14.00 specifically states that if Condenser vacuum lowers to 22.5 in – Hg, the turbine must be tripped (if power is > 10%), so the question is adequate.

NRC Resolution

The correct answer is definitively supported by 1-AP-14.00 Step 2 (if condenser vacuum is not >22.5" Hg, trip the reactor). The question asks when 1-E-0 is required to be entered, so per this step in AP-14.00, tripping at 25.0" Hg would be much too soon, and would bypass steps that might restore condenser vacuum and thus avoid an unnecessary reactor trip.

But the applicant's comment seems to ask if it would ever be correct to trip the reactor at 25 in-Hg, specifically citing loss of the Steam Dumps at 25 in-Hg. It is true that one of the Steam Dump permissives is that condenser vacuum be "available", with a setpoint of 25" Hg (ref. Lesson Plan ND-93.3-LP9, Steam Dumps – Handouts, Slide 19). There is no requirement to trip the reactor simply because the steam dumps would not actuate if called upon to do so. In the event that steam release were necessary from the Steam Generators, the one Power Operated Relief Valve and five Code Safety Valves per Steam Generator have sufficient capacity to meet design-basis heat removal requirements without reliance on the steam dumps [ref. Surry Technical Specification 3.6-5 Basis].

No change to the answer key for this question is warranted.

Item #14: RO Written Exam Question #63

Post-Examination Comment

One applicant asked why choice "A" wouldn't also be correct, because isn't 1-SD-LCV-106 the divert valve.

> Unit 1 is operating at 100% when the Reactor Operator notices that Main Feed Pump suction pressure has lowered on both Main Feed pumps and is now reading 400 psig (20 psig lower than before).
> Which one of the following identifies a possible cause for this lower reading.
> A. 1-SD-LCV-106, HP Heater Drain Pump Level Control valve fails OPEN.
> B. 1-CN-FCV-107, Condensate Recirculation valve inadvertently opened.
> C. 1-CP-MOV-100, Condensate Polisher Bypass valve fails OPEN.
> D. 1-CN-126, Condensate bypass for LP FW Heaters 2, 3 and 4 inadvertently opened.
>
> Answer Key correct answer: B

The facility responded that 1-SD-LCV-106 is the normal Discharge LCV to the Feed Pump Suctions. 1-SD-LCV-107 is the divert valve. Opening 1-SD-LCV-106 further will cause a rise in Feed pump suction pressure. No change is required, the question is adequate.

NRC Resolution

The noun name for 1-SD-LCV-106 given in the question stem is "HP Heater Drain Pump Level Control Valve." Simulator Drawing SD1C provided to the NRC as a technical basis for the question indicates that 1-SD-LCV-106 is in the main flowpath from the Heater Drain Pumps to the Main Feed Pumps. This valve failing open (from its normal partly-throttled position) would cause more of the Heater Drain Pump's discharge pressure to be applied to the suction of the Main Feed Pumps, making their suction pressure higher.

No change to the answer key is warranted.

Item #15: RO Written Exam Question #67

Post-Examination Comment

One applicant contended that answer choice "C" is also correct per GOP-1.4, Attachments 6 and 7.

Which ONE of the following completes the statement from 1-GOP-1.4, Unit Startup, HSD to 2% Reactor Power, concerning the Reactivity Plan provided by Reactor Engineering? The plan shall contain recommendations for control of Delta Flux, rod height and/or Boron adjustments, and _____. A. expected Xenon transient B. startup rate limitations C. RCS temperature control D. source range counts at doubling points Answer Key correct answer: A

The facility agrees that choice "C" is also correct, because 1-GOP-1.4 also references the distractor "RCS temperature control" in attachment 7. The following supporting material was provided:

Supporting Facts:

- This question references step 5.1.3 to support the correct answer:

5.1.3 Check that a Reactivity Plan has been provided by Reactor Engineering. The plan shall contain recommendations for control of core parameters:

- Delta Flux Control
- Recommendations for rod height and/or RCS Boron adjustments
- Expected Xenon transient

And in this reference, Tave is not listed, but upon review of attachment 7 (Reactivity Control and Monitoring During Startup), the following is given:

(Page 1 of 3)

Attachment 7

REACTIVITY CONTROL AND MONITORING DURING STARTUP

_____ 1. Begin logging data on Attachment 7 (page 2 of 3) at a maximum interval of 15 minutes. Use multiple sheets as required.

_____ 2. Begin logging reactivity manipulations on Attachment 7 (page 3 of 3) as applicable. Use multiple sheets as required.

_____ 3. Maintain Tave and Tref approximately matched and Delta Flux in band (use Control Rods, Boration and/or Dilution) as discussed during the pre-job brief. Use the Reactivity Plan as a guide. (**Reference 2.4.13**)

The statement "Use the Reactivity Plan as a guide" infers that the plan contains this information.

- Upon review of an engineering supplied reactivity plan, Tave is a parameter given as a reference:

S1C28 RXMANP03.3		DATA LOADED VERSION OK		S1C28 BOC-Online and 30% Hold					
Number of cases =	22	HFP Depletion Rate (ppm/efpd):	2.32	Maximum (+) Ramp Rate (% / min)	0.80	Maximum (-) Ramp Rate (% / min)	0.00	Minimum margin to the RIL (steps):	106
Cycle Burnup (MWD/MTU)	0	RCS Boron (ppm)	1488	RCS B10/B ratio	0.2000	BAST Boron (% or ppm)	14350	BAST B10/B (Ratio)	0.2
RCS Leak Rate (gpm)	0.1	Boration Dilution Filter (ppm)	0.0	HFP ARO A/O Bias (%)	0.0				

S1C28 BOC-Online and 30% Hold

Prepared: Date: 11/10/17 Reviewed: Date:

Date	Elapsed Time (Hours)	Time Interval (Hours)	Core Power (%)	D-Bank (Steps)	RCS Tave (°F)	Boration Rate (gals/min)	Boration Step (gallons)	Boration Total (gallons)	Dilution Rate (gals/min)	Dilution Step (gallons)	Dilution Total (gallons)	Critical Boron (ppm)	Δ-Flux Estimate (%)	Xe Worth (pcm)	
10/23/17 15:00	0.00	0.00	6.0	149	548.6	N/A	0	0	N/A	0	0	1488	-0.2	-491	
10/23/17 15:15	0.25	0.25	6.0	149	548.6	0.0	0	0	2.5	38	38	1487	-0.2	-491	
10/23/17 15:30	0.50	0.25	18.0	163	551.7	0.0	0	0	37.7	565	603	1471	0.4	-488	
10/23/17 15:45	0.75	0.25	30.0	177	554.8	0.0	0	0	35.3	529	1133	1456	2.1	-482	
10/23/17 16:00	1.00	0.25	30.0	177	554.8	0.1	1	1	0.0	0	1133	1456	2.1	-479	
10/23/17 16:15	1.25	0.25	30.0	177	554.8	0.1	1	2	0.0	0	1133	1457	2.1	-478	
10/23/17 16:30	1.50	0.25	30.0	177	554.8	0.0	1	3	0.0	0	1133	1457	2.1	-477	
10/23/17 16:45	1.75	0.25	30.0	177	554.8	0.0	0	3	0.0	0	1133	1457	2.1	-477	
10/23/17 17:00	2.00	0.25	30.0	177	554.8	0.0	0	3	0.2	3	1135	1457	2.1	-478	
10/23/17 17:15	2.25	0.25	30.0	177	554.8	0.0	0	3	0.5	7	1142	1457	2.0	-480	
10/23/17 17:30	2.50	0.25	30.0	177	554.8	0.0	0	3	0.7	11	1154	1456	2.0	-483	
10/23/17 17:45	2.75	0.25	30.0	177	554.8	0.0	0	3	1.0	15	1169	1456	2.0	-486	
10/23/17 18:00	3.00	0.25	30.0	177	554.8	0.0	0	3	1.2	19	1187	1455	2.0	-491	
10/23/17 18:15	3.25	0.25	30.0	177	554.8	0.0	0	3	1.5	22	1209	1455	2.0	-495	
10/23/17 18:30	3.50	0.25	30.0	177	554.8	0.0	0	3	1.7	25	1234	1454	1.9	-501	
10/23/17 18:45	3.75	0.25	30.0	177	554.8	0.0	0	3	1.9	28	1262	1453	1.9	-507	
10/23/17 19:00	4.00	0.25	30.0	177	554.8	0.0	0	3	2.1	31	1293	1452	1.9	-514	
10/23/17 19:15	4.25	0.25	30.0	177	554.8	0.0	0	3	2.2	33	1327	1451	1.9	-521	
10/23/17 19:30	4.50	0.25	30.0	177	554.8	0.0	0	3	2.4	36	1362	1450	1.8	-529	
10/23/17 19:45	4.75	0.25	30.0	177	554.8	0.0	0	3	2.5	38	1400	1449	1.8	-537	
10/23/17 20:00	5.00	0.25	30.0	177	554.8	0.0	0	3	2.7	40	1441	1448	1.8	-546	
10/23/17 20:15	5.25	0.25	30.0	177	554.8	0.0	0	3	2.8	42	1483	1447	1.8	-555	

Conclusion:
- Question 67 contains two correct answers as the reference given (GOP-1.4) contains supportive information for “A” and “C”.

NRC Resolution

NRC does not agree that “C” is also a correct answer. The question specifically asks which answer choice “**completes the statement from 1-GOP-1.4** …” *[emphasis added]* That “statement,” as submitted by the facility during exam development as the explanation for the correct answer, is Step 5.1.3 on p. 12 of 1-GOP-1.4, Rev. 60, which the licensee included above. The third bullet is the correct answer. Note that RCS temperature is not included in the list.

Furthermore, 1-GOP-1.4 in three places (Step 5.2.5.m, Step 5.4.15, and Attachment 4) provides direction to maintain RCS temperature at approximately 547°F. There is no need to infer that the Reactivity Plan is controlling the desired temperature.

Additionally, later when the main generator is connected to the grid, the main turbine impulse chamber pressure provides a signal for calculation of the “reference” RCS temperature, T_{REF}. GOP-1.5, “Unit Startup, 2% Reactor Power to Max Allowable Power,” gives direction in 4 places (5.3.22, 5.5.6, & 5.5.43, and Attachment 9 Step 4) to “maintain T_{AVE} close to T_{REF}.” So again, there’s no need to look to a Reactivity Plan for guidance on how to maintain T_{AVE}: it’s to be maintained “close to” T_{REF}.

Finally, even though the Reactivity Plan has a column titled “RCS T_{AVE},” this is essentially the value of T_{REF} at the corresponding Core Power level. Reactor Engineering creates a Reactivity Plan to take the reactor from shutdown to some power level, typically 100%. The “waypoints” (typically at 15-minute intervals) are chosen based on nuclear fuel constraints related to rate of power rise, heat removal capabilities, fission product poisons, and other factors. To calculate the amount of control rod withdrawal and/or dilution water or boric acid to specify, the engineer has to consider the *current* power level and the *next* desired power level. Since these two power levels will have different T_{REF} values, the engineer has to take into account the expected change in T_{REF}, and will provide reactivity variables (rod height, dilution water) **that will match T_{AVE} with T_{REF}** at the new power level. So while T_{AVE} values are listed on the Reactivity Plan, they’re not the driver for maintaining a particular T_{AVE}. That comes from the GOP guidance to maintain T_{AVE} “close to” T_{REF}. This column could be left off of the Reactivity Plan and not impact the crew’s ability to operate, because they are **always** expected to maintain T_{AVE} near T_{REF}.

No change to the answer key is justified.

Item #16: RO Written Exam Question #69

Post-Examination Comment

One applicant contended that choice "B" is also correct since VCT level is used in the calculation for identified leakage.

According to surveillance procedure 1-OPT-RC-10.00, Reactor Coolant Leakage-Computer Calculated, which of the following parameters is an input into the calculation for IDENTIFIED RCS leakage? A. Pressurizer Level. B. VCT Level. C. Pressurizer Relief Tank Level. D. Containment Sump Level. Answer Key correct answer: C

The facility noted that VCT level and Pressurizer level are used to calculate Total Leakage, but that only PRT and PDTT levels are used to calculate Identified Leakage. Their position was that the question is adequate as written.

NRC Resolution

RCS Leakage is categorized as Total, Identified, or Unidentified.

To determine TOTAL RCS leakage, RCS temperature and Pressurizer level are held constant, and the decrease in VCT level in gallons divided by the duration time of the test gives the TOTAL leak rate. This is the basis for Distractor 'B', VCT Level.

IDENTIFIED leakage is determined by measuring leakage "that is captured and conducted to collection systems" (terminology from Technical Specifications). The Pressurizer Relief Tank is one such collection system; if level increases in this tank then it can only be because of RCS leakage from a known set of components. This is the support for the correct answer, 'C'. Note that VCT level is not needed for this calculation; only the change in PRT level is required.

UNIDENTIFIED leakage is simply the amount of TOTAL leakage that is not IDENTIFIED, and is obtained by subtracting IDENTIFIED leakage from TOTAL leakage.

One does not need to know TOTAL leakage (derived from VCT level change) to calculate IDENTIFIED leakage, therefore 'B' is not a correct answer. No change to the answer key is necessary.

Item #17: RO Written Exam Question #75

Post-Examination Comment

One applicant commented that the "CAP" page of 1-ES-1.1 affirms the answer of D.

Given the following:

0130 Unit 1 was at 100% power when a spurious Safety Injection occurred.
0131 The crew enters 1-E-0, Reactor Trip or Safety Injection.
0145 The crew transitions to 1-ES-1.1, SI Termination.
0155 The RCS is solid and PRZR PORV,1-RC-PCV-1456 begins to cycle.
0158 HHSI flow has been isolated and charging placed in service.
0212 While placing letdown in service 1-RC-PCV-1456 opens and will not close.
0217 The block valve for 1-RC-PCV-1456 will not close.

- RCS pressure is 1000 psig and lowering rapidly.
- All steam generators are at 900 psig and slowly lowering.

Which ONE of the following describes the correct actions and procedure implementation?

A. Initiate Safety Injection and go to 1-E-0, Reactor Trip or Safety Injection.

B. Transition to 1-E-1, Loss of Reactor or Secondary Coolant, and use 1-E-1 guidance to reinitiate Safety Injection.

C. Manually start charging pumps and align HHSI flow path, and go to 1-ES-1.2 Post LOCA Cooldown and Depressurization.

D. Manually start charging pumps and align HHSI flow path, and go to 1-E-1, Loss of Reactor or Secondary Coolant.

Answer Key correct answer: D

The facility agreed.

NRC Resolution

The intent of this comment is not clear, but the Continuous Actions Page (the "CAP" page) for 1-ES-1.1 was the basis for the correct answer that was given in the question development package. No changes to the question or answer key are required.

SIMULATOR FIDELITY REPORT

Facility Licensee: Surry Nuclear Plant

Facility Docket No.: 050000280, 050000281

Operating Test Administered: October 30 – November 3, 2017

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.