<u>RO Question 24</u>

Given the following conditions for Unit 2:

- A LOCA has occurred.
- While performing 2-EOP-LOCA 1 "LOSS OF REACTOR COOLANT" 22 RHR pump motor seizes and power is lost to 21 RHR pump.
- The crew enters 2-EOP-LOCA-5, "LOSS OF EMERGENCY RECIRCULATION".
- A cooldown has been initiated as directed in 2-EOP-LOCA-5.
- During the cooldown, the crew restores power to 21 RHR pump.
- RWST Level is 30'

Based on current plant conditions, which of the following describes the mitigation strategy?

a. Continue with the cooldown and start 21 RHR pump when directed in 2-EOP-LOCA-5.
b. Return to 2-EOP-LOCA-1 and continue recovery actions with the step previously in effect.
c. Start 21 RHR pump and continue actions of 2-EOP-LOCA-5 until the RWST LO Level alarm actuates.
d. Start 21 RHR pump and transition to 2-EOP-LOCA-3, Transfer to Cold Leg Recirculation, to verify recirculation flowpath.

Correct answer: b

References: 2-EOP-LOCA-5, Loss of Emergency Recirculation
2-EOP-LOCA-1, Loss of Reactor Coolant
S2.OP-TM.ZZ-0002, Tank Curves

Facility comment: Both a and b are correct.

- Stem conditions places candidate at the RCS Cooldown to Cold Shutdown step 10 of EOP-LOCA-5.

- In the stem of the question, given that 21 RHR pump is restored, the candidate has to make a decision on the appropriate procedure flowpath. The CAS of step 6.1 says… "**<u>IF</u>** emergency recirculation capability is restored during this procedure **<u>THEN</u>** return to step 6". Step 6 directs operator to return to procedure in effect <u>if</u> at least one train of emergency recirculation in Table B is available. Table B requires Containment Sump level >62%, a RHR pump, and its associated SJ44. With RWST at 30' there would not be sump level of > 62%. If ALL the conditions of Table B are not met, the candidate would continue on with the cooldown in EOP-LOCA-5 until such time that Table B conditions were ALL met.

- Per S-C-A900-MDC-0082 page 13 shows that the minimum sump volume for RHR pump suction is 160248 gallons.

- Per S2.OP-TM.ZZ-0002 the total volume of water from 40.5ft to 30 is 89500 gallons, which is not enough volume to cause containment sump level to be at 62%.
    - Flowpath #1: The candidate will continue the cooldown in step 10. If the RWST Lo-Lo lvl CAS is not performed at step 12 they will continue on to reduce SI flow to one train, isolate the RWST from the containment sump in step 15, stop/start all but one RCP, evaluate minimum SI flow for RHR, align normal charging, ensure adequate RCS makeup, depressurize the RCS to minimize subcooling and at step 26, with RHR entry conditions being met, the candidate will place the available RHR loop in service when directed by the TSC.
    - Flowpath #2. At step 12 a CAS is encountered that if the RWST Lo-Lo level alarms then go to step 30. The ensuing steps stop pumps with suction from the RWST, perform RCS alternate makeup, depressurize SG's to inject accumulators, isolate the accumulators amd continue RCS depressurization. Step 38 RHR initiation criteria would have the candidate place the available RHR loop in service when directed by TSC.

The containment sump level will reach 62% prior to reaching 15.2' in the RWST, as the level selected (15.2') will ensure that there will be adequate containment sump level to allow transfer to cold leg recirc at this time. When this condition is present, ALL the conditions in LOCA-5 Table B will be met, and the crew will transition back to procedure in effect (LOCA-1). However, because the stem states that a LOCA has occurred but does not indicate its size. i.e., does not state that it is a LBLOCA, a LOCA of indeterminate size could cause slow drawdown of the RWST during performance of LOCA-1, and continued slow drawdown as operators perform actions of LOCA-5, which could indicate that substantial actions would be taken in LOCA-5, long enough for the crew to reach the step to initiate RHR when directed by the TSC. Based on the assumption of a smaller then Large Break LOCA, it would be plausible to continue in LOCA-5 and have time to reach the step where RHR initiation would be directed by the TSC before 62% sump level was achieved. It would also be plausible, if assuming the LOCA was of significantly Larger size, to assume that containment sump level would reach 62% in a very short time, knowing that the 62% would be somewhere between RWST level of 30' (from stem) and 15.2'.

Facility recommends accepting a and b as correct.

Since the stem does not provide enough information as to the size of the LOCA, the facility believes both Choice a and Choice b can be correct based on differing assumptions of LOCA size.

NRC Response

The facility's recommendation is partially accepted. The NRC examiners determined the correct answer is only choice A; choice B is not also correct.

The question asks the candidate to make a decision whether to remain in LOCA-5, or return to LOCA-1. That decision is based on whether recirculation capability has been restored AND whether containment sump level is >62%. The candidate can accurately conclude recirculation

capability has been restored since a stem condition states 21 RHR pump has been restored. Regarding sump level, the candidate can accurately conclude it is NOT >62%. That conclusion is justified because a stem condition states RWST level is 30'. RWST level correlates to sump level, and thus is a key piece of information needed to remain in LOCA-5. During the LOCA, ECCS systems have pumped down the RWST from its full capacity (approximately 40.5 feet) to the level given in the stem: 30 feet. Such a volume transfer would result in about 90,000 gallons of refueling water being pumped into the sump. That volume of water is not nearly enough volume to cause sump level to exceed 62%. Thus, even though recirculation capability is restored, the requisite sump level condition has not been met. So, per answer choice A, the candidate must continue the cooldown, and start 21 RHR pump when that procedure directs.

In conclusion, the answer to this question is only choice A. The key has been revised accordingly.

RO Question 46

Given the following conditions:

- Operators are responding to a LBLOCA IAW 2-EOP-LOCA-3, Transfer to Cold Leg Recirculation.
- RWST level is 8.9' and lowering as expected.

Which of the following describes the effect if 21CS2 Containment Spray pump discharge valve experienced a short and motored closed?

Containment Spray Header flow will...

a. lower to 0 gpm.
b. lower but remain > 0 gpm.
c. be unaffected since 21CS36 RHR CS STOP VALVE is open supplying all spray flow
d. be unaffected since 22CS36 RHR CS STOP VALVE is open supplying all spray flow.

Correct answer: a

References: 2-EOP-LOCA-3, Transfer to Cold Leg Recirculation
NOS05ECCS00-09 EMERGENCY CORE COOLING SYSTEM Lesson Plan

Facility comment: Both a and b are correct.

- This question assumes that 22 Containment Spray pump is secured. This may not be the case. A crew may not have reached step 8 at this point. This would result in the continued operation of 22 Containment Spray pump.
- The stem of the question states that operators are transferring to 2-EOP-LOCA-3 due to a large break loss of coolant accident. The candidate can assume that all emergency core cooling system pumps are operating in run-out conditions. Per the basis document for 2-EOP-LOCA-3, it assumed that the operator reaches step 5 of 2-EOP-LOCA-3 and have the 2SJ69 close within 3.7 minutes of the receipt of the RWST low level alarm.
- NC.EP-EP.ZZ-0201(Q) – REV. 14 states run-out flows as follows: 2 Centrifugal Charging Pumps at 550 GPM each,2 Safety Injection Pumps at 660 GPM each, and 2 Residual Heat Removal Pumps at 5250 GPM each. FSAR section 6.2.2.1.1 states each containment spray loop is capable of providing a minimum of 2600 GPM per loop. When all ECCS and CS pumps are running concurrently the total flow would be a minimum of 18,120 gallons per minute.
- In accordance with tank capacity data of S2.OP-TM.ZZ-0002 Rev. 8, the unit 2 refueling water storage tank is at approximately 150,000 gallons at 15.2 feet of level and between 90,000 and 100,000 gallons at the level of 8.9 feet stated in the stem. Using the total flow

from the pumps calculated above, the time to reach the conservative value of 90,000 gallons (corresponding to 8 feet of level) would take 3.3 minutes. If the operators performing 2-EOP-LOCA-3 accomplished step 5 at the critical action time of 3.7 minutes, at this time RWST volume would be 82,956 gallons, or 7.2 feet.

- This leads 2 proposed answers to be correct depending on the assumption of the candidate on the time the crew would take to reach step 8 of 2-EOP-LOCA-3 to stop 22 containment Spray pump. Prior to step 8 of EOP-LOCA-3, both containment spray pumps are assumed to be running during a LBLOCA accident. If 21 containment spray pump's outlet valve were to shut prior to stopping 22 containment spray pump at step 8, then flow would lower to about half with 22 CS pump still running making answer (b) correct, however if the crew performing the procedure was operating at a faster speed than required by the basis document for EOP-LOCA-3, the possibility of stopping 22 CS pump could have already happened, causing flow to lower to 0 GPM spray flow, also making answer (a) correct.

Facility recommends accepting a and b as correct.

Using the Critical Action Time described above as the bounding time available for performance of steps in EOP-LOCA-3, Transfer to Cold Leg Recirculation, the time available between the RWST lo lvl alarm at 15.2' and the conservative level of 8' in the RWST is <u>**3.6 minutes**</u> meets the Critical Action Time using design flows as described in Lesson Plan NOS05ECCS00-09. RHR pump flow is limited to 4500 gpm by way of a flow restricting orifice. This means that it is assumed that the 2SJ69 Close PB would be depressed 3.6 minutes after the RWST reached 15.2'. The 2SJ69 close stroke time is ~25 seconds as shown at EOP-LOCA-3 step 5. After 2SJ69 is shut, the crew will perform SI reset actions at step 7.

Based on the assumption of time taken (and allowed by the assumed Critical Action Time), it is plausible that both Containment Spray pumps could be running when the fault in the stem (21CS2 going shut) occurs. Conversely, if the crew were to perform LOCA-3 actions in the most expedient manner, 22 CS pump could be assumed to be stopped at step 8.

<u>NRC Response</u>

Comment accepted: A and B are both correct answers.

The question does not clearly define how quickly the crew progresses through LOCA-3. Consequently, there are two correct answers, and each answer depends on how fast the crew steps through the procedure. If they quickly progress through LOCA-3, then it is plausible the crew would have stopped 22 containment spray pump before the electrical fault occurs. This sequence – stop 22 CS pump; then later experience the electrical fault - results in zero header flow, and is the basis for original correct answer A. However, if the candidate assumes the crew executes LOCA-3 at a pace consistent with Critical Action Times – a pace *slower* than previously described - then it is plausible that both containment spray pumps would still be running at the time the electrical fault occurs. This sequence – experience the electrical fault;

both CS pumps still running - results in diminished (but not zero) header flow, and is consistent with proposed alternate answer B. Salem training staff ran both sequences on their plant-reference simulator and proved the sequences produced the results described in both answer choices.

The NRC review team also considered the facility's recommendation in light of the guidance specified in NUREG-1021, ES-403, Section D.1.c, which states:

> "If it is determined that there are two correct answers, both answers will be accepted as correct. If, however, both answers contain conflicting information, the question will likely be deleted. For example, if part of one answer states that operators are required to insert a manual scram, and part of another answer states that a manual scram is not required, then it is unlikely that both answers will be accepted as correct, and the question will probably be deleted."

Choice A and B contain slightly conflicting information regarding header flow: choice A states header flow lowers to zero; choice B states header flow lowers but remains > zero. However, each answer is correct for the different assumptions a candidate could reasonably make regarding how fast the crew implements LOCA-3.

In summary, the NRC review team chose to keep the question and accept two correct answers on the basis that the question does not make clear how fast the crew executed LOCA-3.

Choices A and B are both correct. The answer key has been revised accordingly.

RO Question 64

Which of the following describes the effect if BOTH Containment Phase A and Phase B isolations occur during a LBLOCA, and when will they be reset when operating in the EOP network?

_________ containment penetrations not supporting ECCS functions will be isolated.

Phase A and Phase B will be reset in ________.

a. ONLY NON-ESSENTIAL. EOP-TRIP-1 Reactor Trip or Safety Injection.

b. ONLY NON-ESSENTIAL. EOP-LOCA-1 Loss of Reactor Coolant.

c. ALL. EOP-TRIP-1 Reactor Trip or Safety Injection.

d. ALL. EOP-LOCA-1 Loss of Reactor Coolant.

Correct answer: d

References: EOP-TRIP-1 Reactor Trip Response

Candidate comment: Both b and d are correct.

Choice B uses the terminology of "non-essential"

The basis document of EOP-TRIP-1 Step 10 states the purpose to ensuring valve groups in table B (phase A, FWI, CVI and SI) are in their proper position is "To ensure non-essential containment penetrations are isolated". A Phase B isolation will isolate CCW to the RCP. The RCP CCW cooling valves (header) are clearly labeled as the Non-Essential header in CCW drawing # 205231-SIMP and 205331-SIMP. Since all the containment penetrations isolated by Phase A and Phase B are considered non-essential by above mentioned documents and drawings, recommend accepting b and d as correct answers.

Facility recommends accepting both b and d as correct..

The two CCW system drawings 205231-SIMP and 205331-SIMP show that the headers from which RCP CCW flow is supplied are labeled "Non-Essential Headers". Additionally, for a Design Basis Accident at Salem, no credit is given for forced flow circulation upon the Loss of Offsite power coincident with a LOCA. Therefore, there is no differentiation to be made between a penetration which is considered "non-essential" and ALL penetrations which do not affect ECCS. This means that ALL penetrations isolated by Phase A and Phase B are considered non-essential, and the two choices, b and d, which contain the correct procedural direction where the isolations will be reset, are correct.

NRC Response

Comment accepted. Based on information from the facility, there is no differentiation between a penetration which is considered “non-essential” and penetrations which do not affect ECCS. Thus, ALL penetrations isolated by Phase A and Phase B are considered non-essential. Therefore, choices B and D, which both contain the correct procedural direction, are correct.

The answer key has been changed accordingly.

SRO Question 5

Given the following conditions:

- Unit 1 is operating at 100% power.
- 21, 24 and 26 SW pumps are in service.
- 21 and 22 SW header pressures are 108 psig.
- The following OHAs annunciate sequentially in this order:
  - B-13, 21 SW HDR PRESS LO
  - B-14, 22 SW HDR PRESS LO
  - B-15, TURB AREA SW HDR PRESS LO.
  - B-48, SW VLV RM FLOODED.

The standby SW pump starts automatically, and OHAs B-13, B-14, and B-15 clear.

Which of the following describes where a SW leak could be located which would produce these alarms, and the procedurally directed actions which would mitigate the event?

a. On a CFCU supply piping in the 78' Mechanical Penetration Area. S2.OP-AB.SW-0001, Loss of Service Water Header Pressure, will direct isolating multiple CFCUs IAW Attachment 5 CFCU leaks, until header pressure stabilizes and sump pump runs stop.

b. Upstream of the 2ST901, TURB LO CLR ST RET VLV. S2.OP-AB.SW-0002, Loss of Service Water-Turbine Header, will direct operators to adjust 2ST901 TURB LO CLR ST RET V, and 2ST1 TG AREA SW PRESS CONT VLV, to compensate for the SW leak.

c. On a CFCU supply piping in the 78' Mechanical Penetration Area. S2.OP-AB.SW-0001, Loss of Service Water Header Pressure, will direct isolating a single CFCU which would be readily identifiable from the control room.

d. Upstream of the 2ST901, TURB LO CLR ST RET VLV. S2.OP-AB.SW-0002, Loss of Service Water-Turbine Header, will direct removing the Main Turbine from service in preapration for isolating the TGA header.

Correct answer: c

References: S2.OP-AB.SW-0001 Loss of Service Water Header Pressure
Drawing 205342 Sheet 6
S2.OP-AR.ZZ-0002, Overhead Annunciators Window B

Facility recommends deleting question based on no correct answer

In the stem of this question, OHA B-48, SW VLV RM FLOODED, annunciates in addition to the three associated service water headers pressure low alarms. This corroborates a service water leak in the service water piping room(s) in the mechanical penetration area.

Distractors "b" and "d" are incorrect because they indicate a service water leak "upstream of the 2ST901." This would be a leak in the turbine building and OHA B-48, SW VLV RM FLOODED would not have annunciated.

Distractor "a" is incorrect because it specifies using Attachment 5, CFCU Leaks, of S2.OP-AB.SW-0001, Loss of Service Water Header Pressure, for isolating multiple CFCUs. Attachment 5 is used only for isolating CFCUs when directed at Step 3.15 which asks, "Is there a Service Water leak in Containment?" If the leak is in containment, Attachment 5 directs actions to isolate the affected CFCU(s) based on whether the leak is identifiable to a single CFCU or multiple CFCUs if the leak is suspected to be very large in an effort to minimize containment sump pumps runs.

When performing S2.OP-AB.SW-0001, operators would rule out leaks in the Service Water Bays (Step 3.2) and Turbine Area (Step 3.5), then be directed to Attachment 3, Guidelines for Locating Service Water Header Leak/Valve Malfunction. Attachment 3 initiates walkdowns of affected areas, including the Service Water piping compartment/tunnels, and at this time the leak could be identified definitively as being in the SW Valve Room. **Because the SW Supply header does not split into individual SW supply lines to each CFCU until it has entered the SW Valve Room, if the leak were on the header supply piping only visual observation of the actual leak location could identify if the leak was on a single CFCU supply line.** This means that the correct answer cannot be correct, since the indication in the control room COULD show lowered SW flow to the two CFCUs supplied from the header which could be leaking in the SW Valve Room.

Per drawing 205342 Sheet 6, the common supply piping for the Service Water (SW) Containment Fan Coil Units (CFCUs) is also in the 78' Mechanical Penetration Area. There are individual flow measurement orifices for each CFCU (FA3539-3543). The stem of the question does not provide any detail on the specific location of the leak (upstream or downstream of the flow measurement orifice). Per the drawing, depending on the location of the leak, flow could drop on at least two CFCUs if the leak was upstream of the orifice. Without additional information provided in the stem, it would be impossible to conclusively state that this would be readily identifiable from the control room.

Because the correct answer states that ..." a single CFCU which would be readily identifiable from the control room." there is no correct answer because if the leak were on the SW header supply piping it would NOT be identifiable to a single CFCU, yet still result in the indications provided in the stem.

<u>NRC Response</u>

Comment accepted. The stem of the question does not provide sufficient detail regarding the leak's location, i.e., the candidate cannot determine whether the leak is upstream or downstream of the flow measurement orifice. If the leak is upstream of the orifice, then flow could drop on *at least two* CFCUs. This condition rules out the proposed correct answer C, which directs the crew to isolate *a single* CFCU. The question does not have a correct answer, and so has been deleted. The answer key has been revised accordingly.

SRO Question 14

Given the following conditions:

- Unit 2 is operating at 100% power.
- Intermediate Range Nuclear Instrument (IRNI) Channel I (2N35) failed yesterday.
- The CRS entered TSAS 3.3.1.1, action 3, and the crew removed the channel from service IAW S2.OP-SO.RPS-0001, Nuclear Instrumentation Channel Trip / Restoration.

- Subsequently, the RO reports IRNI Channel II (2N36) just started indicating erratically, oscillating between 1x10-11 amps and 1x10-5 amps.
- The reactor remains at power.

The CRS shall...

a. Enter S2.OP-AB.NIS-0001, Nuclear Instrumentation Malfunction. Remove IRNI Channel 2N36 from service. Power operation may continue with no restrictions.
b. Enter S2.OP-AB.NIS-0001, Nuclear Instrumentation Malfunction. Remove IRNI Channel 2N36 from service. Initiate actions within one hour to place the unit in Hot Standby within the next 6 hours.
c. Direct the RO to trip the reactor based on an ATWT. BOTH Source Range Nuclear Instrument Channels must be manually energized in EOP-TRIP-2, Reactor Trip Response, if the Rx trip is successful, or 2-EOP-FRSM-1, Response to Nuclear Power Generation, if the reactor does not trip.
d. Direct the RO to trip the reactor based on an ATWT. ONLY Source Range Nuclear Instrumentation Channel II (2N32) will automatically energize. SRNI Channel I (2N31) must be manually energized in EOP-TRIP-2, Reactor Trip Response, if the Rx trip is successful, or 2-EOP-FRSM-1, Response to Nuclear Power Generation, if the reactor does not trip.

Correct answer: b

References: S2.OP-AB.NIS-0001, Nuclear Instrumentation System Malfunction
Salem Tech Spec 3.3.1.1 Reactor Trip System Instrumentation,
(Functional Unit 5, Intermediate Range, Neutron Flux, Action 3)

S2.OP-SO.RPS-0001, Nuclear Instrumentation Channel Trip/Restoration
Salem Initial License Class 11-01 NRC Exam Scenario Guide ESG-4
Salem Initial License Class 13-01 Certification Exam Scenario Guide ESG-1

Facility recommends deleting question based on Salem Training material and previous NRC examination material endorsing an interpretation of Tech Specs not in agreement with correct answer for this question.

The question presents the loss of the second of two intermediate range detectors while Salem Unit 2 is at 100%. Technical Specification 3.3.1.1 contains Action Statements for the loss of a single intermediate range detector (Action 3). Action 3 states the following:

With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below P-6, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint.
b. Above P-6, but below 5% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.
c. Above 5% of RATED THERMAL POWER, POWER OPERATION may continue.
d. Above 10% of RATED THERMAL POWER, the provisions of Specification 3.0.3 are not applicable.

Interpretation of this Action Statement by the Salem Training and Operations Staff has been that Tech Spec 3.0.3 also does not apply to a condition in which two IRNIs are inoperable. There are no additional actions with two IRNI's inoperable > P-10, (10% Reactor Power) than there are with one inoperable IRNI, and Tech Spec 3.0.3 is not applicable >10% power.

When responding IAW S2.OP-AB.NIS-0001, the operators are directed to remove the failed channel IAW S2.OP-SO.RPS-0001. This procedure does not caution against removing two channels from service UNLESS power is below P-10, where a loss of IRNI instrument power could result in a reactor trip, and which aligns with LCO 3.3.1.1 Action 3 which states above 10% power TS 3.0.3 does not apply.

This action, (to allow continued power operation above P-10) was reinforced by the NRC Operating exam (ESG-4) approved for Salem ILOT class 11-01 NRC exam administered in December of 2013, as well as using this scenario as part of Salem ILOT class 13-01 during its Audit examination in November of 2014. During this ESG, one IRNI is initially O/S, and the second becomes erratic. The Expected Plant/Student Response states in the ESG's that... "CRS refers to Tech Spec 3.3.1.1, Action 3. There is no additional action with 2 IRNI's O/S >P-10, than there is with one IRNI, and Tech Spec 3.0.3 is not applicable > 10% power."

An additional indication that candidates were presented this Tech Spec application in this fashion (TS 3.0.3 not applicable in this situation) is further reinforced by the fact that all candidates chose distractor 'A' as the correct answer.

As the Training and Operations Department have endorsed this interpretation (TS 3.0.3 does not apply), as well as the fact that this material was approved for use in the above describes NRC and Audit Exam scenarios, the company requests this question should be removed from the examination.

<u>NRC Response</u>

The comment is partially accepted. Contrary to the facility's proposed recommendation, the NRC reviewers determined the correct answer should be A, and that A is the only correct answer.

The position of both Salem Training and Operations staff is that above 10% reactor power, operation may continue without restrictions. That is, TS 3.0.3 actions (to initiate actions within one hour to place the Unit in Hot Standby within the next six hours) do not apply. The NRC exam team reviewed Salem Unit 2 TS, and confirmed this position. The correct actions for two inoperable IRNI are accurately described in choice A: enter the abnormal procedure for a nuclear instrument malfunction and remove the IRNI channel from service. The other choices are wrong, including the original proposed answer choice, B.

The correct answer is A, and the answer key has been revised accordingly.