K/A Number: 004G2.2.22, Chemical and Volume Control / 1 & 2, Knowledge of limiting conditions for operations and safety limits, (CFR: 41.5 / 43.2 / 45.2)

Level: SRO   Tier #: 2   Group #: 1   IR – RO: 4   IR-SRO: 4.7

Proposed Question:

Current Unit Conditions:

- Units 1 and 2 are operating at 100% power.
- CVCS is in a NORMAL configuration.
- Valve number 1-CH-83, Unit 1 Boric Acid Filter Outlet Valve, is found to be plugged with boron.

The Maintenance Department has estimated a repair time of eight (8) hours to cut out and replace **1-CH-83**. Tagout has been hung and verified.

Which ONE of the following is correct concerning:

1) The operability of the Boron inject flowpaths as a result of the tagout.
2) The actions required to exit the applicable LCO.

(Reference Provided)

A. 1) One inoperable.
   2) Bypass the Boric Acid Filter.

B. 1) Two inoperable.
   2) Bypass the Boric Acid Filter.

C. 1) One inoperable.
   2) Ensure two flowpaths are operable within the required time, or place the unit in HSD within 6 hours.

D. 1) Two inoperable.
   2) Ensure two flowpaths are operable within the required time, or place the unit in HSD within 6 hours.

Proposed Answer: C. 1) One inoperable. 2) Ensure two flowpaths are operable within the required time, or place the unit in HSD within 6 hours.

Explanation: With this valve (1-CH-83) plugged, 1 boric acid flow for Unit 1 has been lost (from BAST to CH pump suction) and the flowpath from the BASTs to the blender. Under these conditions given, one (1) flowpath remains operable from the RWST to the CH pump suction. In order to replace the valve, the piping section containing 1-CH-83 must be tagged out, which also removes the ability to bypass the piping section.

Technical Reference: TS 3.2.C and TS 3.2.B.

Reference Provided to Applicant: Yes, Drawing For LTS00068.TIF, (11448-FM-88A, SH 1 of 4)

Learning Objective: ND-88.3-LP-2, D, Describe the Technical Specifications associated with the CVCS System, including for SRO candidates, the basis behind these specifications.

Question Source: Modified Bank (LTS00068. Modified)

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.2 / 45.2): 10 CFR 55.43 (b) (2)

Comments:

Distractor Analysis:

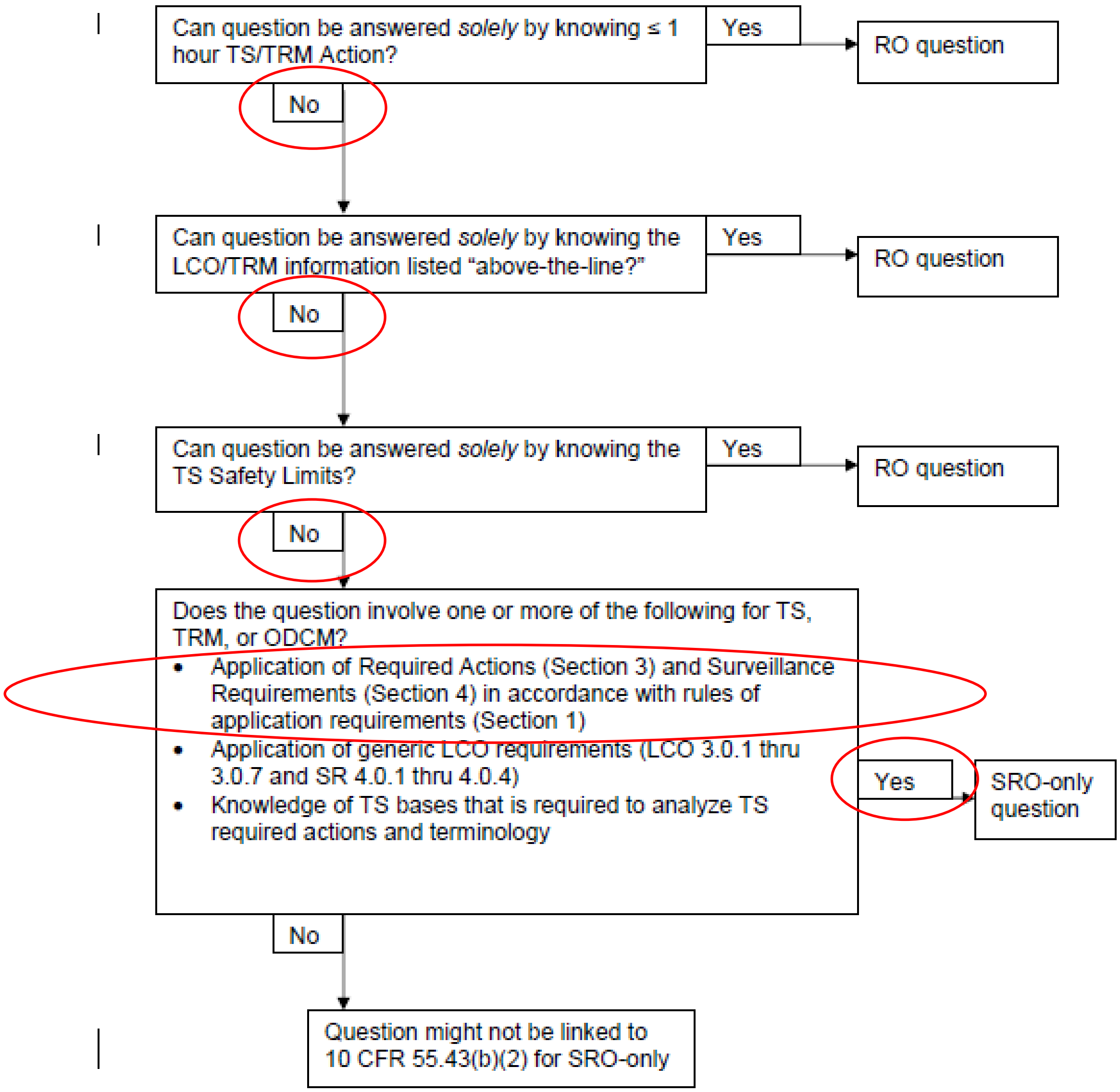
A. 1) Incorrect – Part 1 is correct, part 2 is incorrect. Candidate misconception of tagout requirements for 1-CH-83. In order to cut out and replace valve, the piping section containing 1-CH-83 must be tagged out, which also removes the ability to bypass the piping section.

B. 1) Incorrect – Both Part 1 and part 2 is incorrect. Candidate misconception of what constitutes an operable Boric acid flowpath. There is only 1 inoperable flow path as the RWST is still available. Also, candidate misconception of tagout requirements for 1-CH-83. In order to cut out and replace valve, the piping section containing 1-CH-83 must be tagged out, which also removes the ability to bypass the piping section.

C. 1) Correct – One of two required boric acid flowpaths is inoperable. Restoration of this flowpath will require tagout of valve and bypass line.

D. 1) Incorrect – Part 1 is incorrect, part 2 is correct. Candidate misconception of what constitutes an operable Boric acid flowpath. There is only 1 inoperable flow path as the RWST is still available.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2)
(Tech Specs)



10 CFR 55.43 (b) (2)

TS 3.2-1
05-31-95

3.2 CHEMICAL AND VOLUME CONTROL SYSTEM

Applicability

Applies to the operational status of the Chemical and Volume Control System.

Objective

To define those conditions of the Chemical and Volume Control System necessary to ensure safe reactor operation.

Specification

A. When fuel is in a reactor, there shall be at least one flow path to the core for boric acid injection. The minimum capability for boric acid injection shall be equivalent to that supplied from the refueling water storage tank.

B. The reactor shall not be critical unless:

1. At least two boron injection subsystems are OPERABLE consisting of:

   a. A Chemical and Volume Control subsystem consisting of:

      1. One OPERABLE flow path,

      2. One OPERABLE charging pump,

      3. One OPERABLE boric acid transfer pump,

      4. The common OPERABLE boric acid storage system with:

         a. A minimum contained borated water volume of 6000 gallons per unit,

         b. A boron concentration of at least 7.0 weight percent but not more than 8.5 weight percent boric acid solution, and

         c. A minimum solution temperature of 112°F.

         d. An OPERABLE boric acid transfer pump for recirculation.

b. A subsystem supplying borated water from the refueling water storage tank via a charging pump to the Reactor Coolant System consisting of:

1. One OPERABLE flow path,

2. One OPERABLE charging pump,

3. The OPERABLE refueling water storage tank with:

a. A minimum contained borated water volume of 387,100 gallons,

b. A boron concentration of at least 2300 ppm but not more than 2500 ppm, and

c. A maximum solution temperature of 45°F.

2. One charging pump from the opposite unit is available with:

a. the pump being OPERABLE except for automatic initiation instrumentation,

b. offsite or emergency power may be inoperable when in COLD SHUTDOWN, and

c. the pump capable of being used for alternate shutdown with the opening of the charging pump cross-connect valves.

C. The requirements of Specification 3.2.B may be modified as follows:

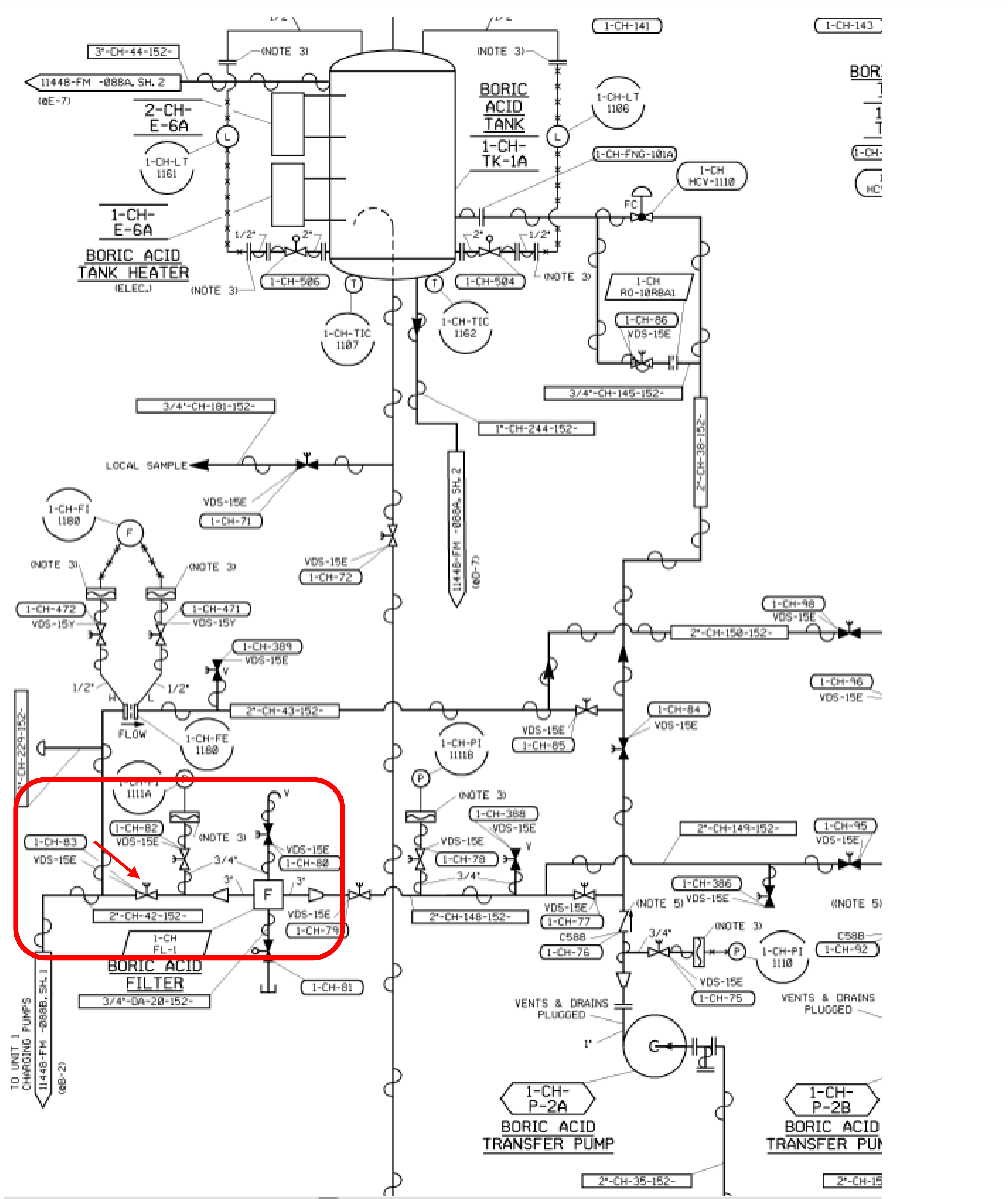
1. With only one of the boron injection subsystems OPERABLE, restore at least two boron injection subsystems to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 6 hours.

2. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or place the reactor in HOT SHUTDOWN within the next 6 hours.

a. For conditions where the RWST is inoperable due to boron concentration or solution temperature not being within the limits of Specification 3.3.A.1, restore the parameters to

within specified limits in 8 hours or place the reactor in HOT SHUTDOWN within the next 6 hours.

3. With no charging pump from the opposite unit available, return at least one of the opposite unit's charging pumps to available status in accordance with Specification 3.2.B.2 within 7 days or place the reactor in HOT SHUTDOWN within the next 6 hours.


BORIC ACID TANK
1-CH-TK-1A
BORIC ACID TANK HEATER
(ELEC.)
2-CH-E-6A
1-CH-E-6A
1-CH-LT 1161
1-CH-LT 1106
1-CH-TIC 1107
1-CH-TIC 1162
LOCAL SAMPLE
1-CH-FI 1180
1-CH-FE 1180
1-CH-PI 1111A
1-CH-PI 1111B
1-CH FL-1
BORIC ACID FILTER
1-CH-P-2A
BORIC ACID TRANSFER PUMP
1-CH-P-2B
BORIC ACID TRANSFER PUMP
TO UNIT 1 CHARGING PUMPS
VENTS & DRAINS PLUGGED
1-CH-PI 1110
1-CH HCV-1110
1-CH RO-10RBA1
2"-CH-42-152-
2"-CH-43-152-
2"-CH-148-152-
2"-CH-149-152-
2"-CH-150-152-
2"-CH-35-152-
3/4"-DA-20-152-
3/4"-CH-181-152-
3/4"-CH-145-152-
1"-CH-244-152-
3"-CH-44-152-
1-CH-83
1-CH-82
1-CH-80
1-CH-79
1-CH-81
1-CH-78
1-CH-388
1-CH-389
1-CH-71
1-CH-72
1-CH-471
1-CH-472
1-CH-84
1-CH-85
1-CH-86
1-CH-98
1-CH-96
1-CH-95
1-CH-386
1-CH-77
1-CH-76
1-CH-75
1-CH-92
1-CH-504
1-CH-506
1-CH-141
1-CH-143
1-CH-FNG-101A
VDS-15E
VDS-15Y
(NOTE 3)
(NOTE 5)
FLOW

K/A Number: 009EA2.39, Small Break LOCA / 3, Ability to determine or interpret the following as they apply to a small break LOCA: (CFR 43.5 / 45.13) Adequate core cooling

Level: SRO Tier #: 1 Group #: 1 IR – RO: 4.3 IR-SRO: 4.7

Proposed Question:

Initial Unit 1 Conditions:

- A small break LOCA has occurred.
- RCS Pressure is 770 psig and stable.
- Wide Range T-Cold indication is 446°F and slowly lowering.
- Average 5 high CETC temperatures indication is 452°F and slowly lowering.
- Containment Pressure is 21.4 psia and slowly rising.
- Containment Radiation level by CHRRMS (1-RM-RMS-127/128) indicate 1R/Hr, and are unchanged from time of event initiation.

In accordance with the above conditions, which one of the following states:

1) The status of RCP Trip criteria.
2) The required procedural flow path is E-0 (Reactor Trip or SI), E-1 (Loss of Reactor or Secondary Coolant), and ________.

A. 1) RCP Trip criteria is met.
   2) ES-1.1, SI Termination.

B. 1) RCP Trip criteria is NOT met.
   2) ES-1.2, Post LOCA Cooldown and Depressurization.

C. 1) RCP Trip criteria is met.
   2) ES-1.2, Post LOCA Cooldown and Depressurization.

D. 1) RCP Trip criteria is NOT met.
   2) ES-1.1, SI Termination.

Proposed Answer: C. 1) RCP Trip criteria is met. 2) ES-1.2, Post LOCA Cooldown and Depressurization.

Explanation: Calculated subcooling based on parameters given is 64°F (Tsat for 770 psig is 516 °F, 516 °F – 452 °F = 64 °F); the Containment is adverse therefore RCP Trip criteria is subcooling < 85 ^{o}F. The required procedural flowpath is E-0 Reactor Trip, E-1 Loss of Reactor or Secondary Coolant, and ES-1.2 Post LOCA Cooldown and Depressurization. E-1, step 20 directs entry into ES-1.2 if RCS pressure is greater than 250 psig [400 psig].

Technical Reference: 1-ES-1.2, Post LOCA Cooldown and Depressurization, Revision 43. Step 22, Page 14.

Reference Provided to Applicant: NO

Learning Objective: ND-95.3-LP-9, ES-1.2; Objective B, Given a copy of ES-1.2, Post-LOCA Cooldown and Depressurization, explain the basis of each procedural step.

Question Source: Modified Bank , Surry, 2004, Question 78

Question History: Last NRC Exam: NO

Question Cognitive Level: Analysis

10 CFR Part 55 Content: (CFR 43.5 / 45.13) 10 CFR 55.43(b)(5)

Comments:

Distractor Analysis:

A. Incorrect – RCP trip criteria is met (correct). ES-1.1 (incorrect). ES-1.1 is SI Termination and is not correct for these conditions. This is plausible because if non-adverse subcooling numbers are used (30 ^{o}F) then ES-1.1 would be correct per step 6 of 1-E-1.
B. Incorrect – RCP trip criteria is NOT met (incorrect). This is plausible because if non-adverse subcooling is used then RCP trip criteria would not be met. ES-1.2 (correct).
C. Correct – RCP trip criteria is met (correct), and E-0, E-1, ES-1.2 (correct).
D. Inorrect - RCP trip criteria is NOT met (incorrect). This is plausible because if non-adverse subcooling is used then RCP trip criteria would not be met. ES-1.1 is SI Termination and is not correct for these conditions. This is plausible because if non-adverse subcooling numbers are used (30 ^{o}F) then ES-1.1 would be correct per step 6 of 1-E-1.

References for part 1) correct answer:

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. Charging Pumps - AT LEAST ONE RUNNING AND FLOWING TO RCS

   b. RCS Subcooling - LESS THAN 30°F [85°F]

References for part 2) correct answer:

NUMBER	PROCEDURE TITLE	REVISION
1-E-1	LOSS OF REACTOR OR SECONDARY COOLANT	40
		PAGE 21 of 28

20. ___ CHECK IF RCS COOLDOWN AND DEPRESSURIZATION IS REQUIRED:

☐ a) RCS pressure - GREATER THAN 250 PSIG [400 PSIG]

☐ b) GO TO 1-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION

☐ a) IF LHSI pump flow greater than 1000 gpm, THEN GO TO Step 21.

References for incorrect answers:

NUMBER	PROCEDURE TITLE	REVISION
1-E-1	LOSS OF REACTOR OR SECONDARY COOLANT	40
		PAGE 5 of 28

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*6.___	CHECK IF SI FLOW SHOULD BE REDUCED:	
	☐ a) RCS subcooling based on CETCs - GREATER THAN 30°F [85°F]	☐ a) GO TO Step 7.
	b) Secondary heat sink:	☐ b) GO TO Step 7.
	☐ • Total feed flow to INTACT SGs - GREATER THAN 350 GPM [450 GPM]	
	OR	
	☐ • Narrow range level in at least one intact SG - GREATER THAN 12% [18%]	
	☐ c) RCS pressure - STABLE OR INCREASING	☐ c) GO TO Step 7.
	☐ d) PRZR level - GREATER THAN 22% [50%]	☐ d) Try to stabilize RCS pressure with normal PRZR spray. GO TO Step 7.
	☐ e) GO TO 1-ES-1.1, SI TERMINATION	

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing immediate operator actions?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?
- Yes → RO question
- No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?
- Yes → RO question
- No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- ~~Knowledge of when to implement attachments~~ and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

- Yes → SRO-only question
- No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

K/A Number: 013A2.03, Engineered Safety Features Actuation / 2, Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations;, (CFR: 41.5 / 43.5 / 45.3 / 45.13) Rapid depressurization

Level: SRO Tier #: 2 Group #: 1 IR – RO: 4.4 IR-SRO: 4.7

Proposed Question:

Unit 1 was operating at 100% power when the following events occur:

- Seismic activity 10 miles northwest of Richmond causes multiple Power Station trips, grid instability and loss of offsite power.
- Both Units automatically trip, and #3 EDG starts and trips on differential current.
- Unit 1 experiences a LBLOCA.
- When Containment pressure exceeds 23 psia, Hi-Hi CLS alarms are received. 1-CS-P-1A fails to start due to mechanical damage.
- The Team has just reached E-0, Reactor Trip or Safety Injection, Step 11, “Check RCS intact inside Containment”.

Based on the above conditions:

1) Which ONE of the following states the correct procedural implementation?
2) Classify the event using the Emergency Plan.

(Reference Provided)

A. 1) Immediately Transition to FR-Z.1, Response to Containment High Pressure.
2) Alert FA1.1.

B. 1) Transition to FR-Z.1 on Transition to 1-E-1, Loss of Reactor or Secondary Coolant.
2) SAE FS1.1.

C. 1) Immediately Transition FR-Z.1, Response to Containment High Pressure.
2) SAE FS1.1.

D. 1) Transition to FR-Z.1 on Transition to 1-E-1, Loss of Reactor or Secondary Coolant.
2) Alert FA1.1.

Proposed Answer: B

Explanation: FR-Z.1, Response to Containment High Pressure, will be indicated when the Containment Status Tree becomes ORANGE; this occurs when CTMT pressure is >23 psia and NO CTMT Spray pumps are operating. With the loss of 1J 4160 V bus (“B” CS pump has no power), the failure of 1-CS-P-1A to start on Hi-Hi CLS (23 psia in CTMT), and CTMT pressure >23 psia, FR-Z.1

ORANGE path is indicated. Immediate transition to FR-Z.1 does not comply with the rules of usage; FR procedures do not apply until E-0 has been exited. The classification of SAE is correct due to 1) LOCA inside CTMT (Loss of RCS Barrier), and 2) CTMT pressure is not responding as expected (Loss of Containment Barrier) for LOCA condition.

Technical Reference: 1-FR-Z.1, Response to Containment High Pressure. Surry EAL Table, Revision 4.

Reference Provided to Applicant: Yes

Learning Objective: ND-95.3-LP-48, FR-Z.1, Objective C, Given a copy of FR-Z.1, Response to Containment High Pressure, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition. ND-95.5-LP-2, SEM, Objective C, Using EPIP-1.01, Emergency Manager Controlling Procedure, assess/classify plant situations by accessing the EAL charts (both HOT and COLD conditions).

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13) 10 CFR 55.43 (b) (5)

Comments:

Distractor Analysis:

A. Incorrect – 1) Transition to FR-Z.1 at Step 11 of E-0 does not comply with “Rules of Usage”; the transition is made upon exit from E-0 to E-1. Step 14 of E-0 is the step that would allow direct transfer to FR-Z.1. 2) Classification of Alert is not complete, the Candidate must assess that the Unit is experiencing a LOCA, and CTMT pressure > 23 psia with no CS pump running.

B. Correct – The transition to FR-Z.1 is appropriate once E-0 has been exited, and the SAE classification is appropriate for the conditions existent.

C. Incorrect – 1) Transition to FR-Z.1 at Step 11 of E-0 does not comply with “Rules of Usage”; the transition is made upon exit from E-0 to E-1. Step 14 of E-0 is the step that would allow direct transfer to FR-Z.1. 2) Classification is correct.

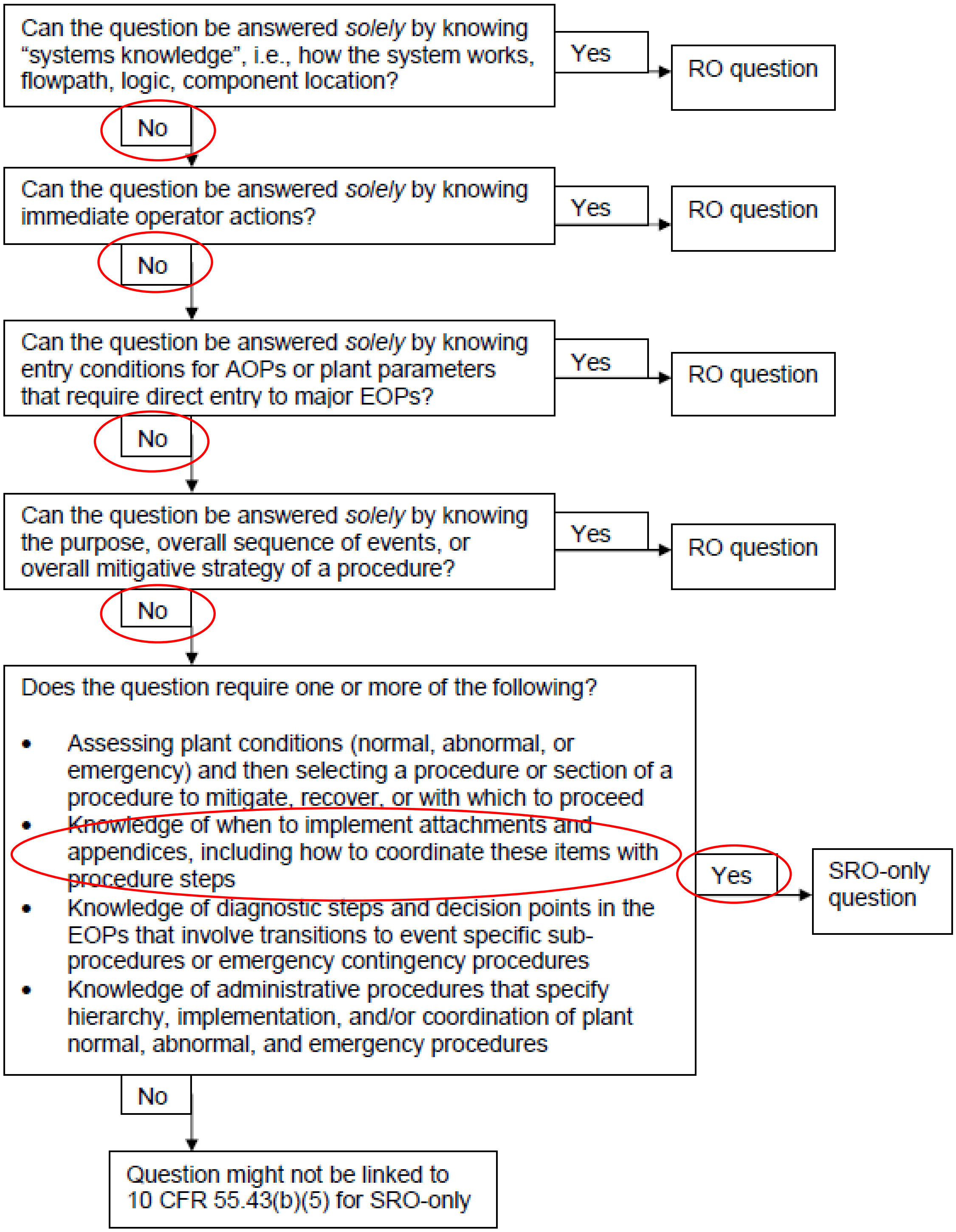
D. Incorrect – 1) Transition is appropriate. 2) Classification of Alert is not complete, the Candidate must assess that the Unit is experiencing a LOCA, and CTMT pressure > 23 psia with no CS pump running.

References for correct answer: OP-AP-104, Emergency and Abnormal Operating Procedures

Procedure User 3.6.1 **MONITOR** Critical Safety Status carefully.

a. Monitoring and implementation of the CSF Status Trees shall begin when directed by the initial emergency response procedure, or when a transition is made to another emergency procedure unless otherwise directed.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



10 CFR 55.43 (b) (5)

K/A Number: 017G2.4.47, in-core Temperature Monitor / 7, Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material., (CFR: 41.10 / 43.5 / 45.12). (NRC Reviewed)

Level: SRO Tier #: 2 Group #: 2 IR – RO: 4.2 IR-SRO: 4.2

Proposed Question:

Given the following:

- Unit 1 is currently at 371 °F and is heating up to HSD following a refueling outage.
- Annunciator 1D-B5, ICCM Failure is received.
- Team implements ARP 1D-B5.

The following is the current Quadrant II CETC indications for Train A and Train B:

CETCs Train A		**CETC Train B**	
B10	2300°F, stable	A09	40°F, stable
C08	40°F, stable	C12	40°F, stable
E11	40°F, stable	D10	368°F, rising slowly
F09	370°F, rising slowly	F11	783°F, rising 1°/30 seconds
F13	40°F, stable	G09	815°F, rising 1°F/15 seconds
H11	369°F, rising slowly	G14	2300°F, Stable

Which ONE of the following identifies

1) The number of FAILED CETCs in Train A and Train B. AND
2) The Tech Spec clock in effect.

(Reference Provided)

A. 1) Train A: **4**; Train B: **5**.
2) 7 day clock.

B. 1) Train A: **3**; Train B **2**.
2) 7 day clock.

C. 1) Train A: **4**; Train B: **5**.
2) 30 day clock.

D. 1) Train A: **3**; Train B **2**.
   2) 30 day clock.

Proposed Answer: C

Explanation: Tech Spec 3.7.E and TS Table 3.7-6, Accident Instrumentation, require at least 2 CETC per quadrant operable per train to consider the train operable. A CETC indication of 40 °F indicates that the CETC has been disabled by I&C department. CETC F11 and G09 are considerably above current RCS temperature and rising at a rate greater that RCS temperature. By count of disabled and failing CETCs in Quadrant 2: A train has 4 failed CETCs (B10, C08, E11, and F13); B Train has 5 failed CETCs (A09, C12, F11, G09, and G14). B CETC Train has fewer than 2 CETC operable in this Quadrant therefore B train is inoperable.

Technical Reference: TS 3.7.E and Table 3.7-6.

Reference Provided to Applicant: Yes (TS 3.7.E)

A. Learning Objective: ND-93.4-LP-3, Objective A, Describe the operation of the Inadequate Core Cooling Monitor (ICCM) System, including the following: CETC inputs and outputs.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12) 10 CFR 55.43 (b)(2)

Comments:

A. Incorrect – 1) CETC at 40 °F have been disabled and the CETS at 2300 have also failed therefore, Train "A" =4 (correct). Train B has 2 CETCs at 40 °F, 1 CETC at 2300 °F and 2 CETCs that are 783 °F and 815 °F and rising rapidly. Therefore Train "B" =5 (correct). 2) With the indication provided, train "B" has fewer than 2 Operable CETCs, Train "A" has a sufficient number of Operable CETCs to call the Train Operable. TS 3.7.E 30 day clock would be in effect vice a 7 day clock (incorrect).
B. Incorrect – 1) Train 'A' has 4 failed CETCs (not 3). Plausible if the operator doesn't include the CETC indicating 2300 °F which once disabled by I&C will indicate 40 °F (incorrect). Also B train has 5 failed CETCs not 32. 2) TS 3.7.E 7 day clock would in effect if <u>Both</u> Trains had less than the required number of Operable CETCs (2). Part 2 is also incorrect.
C. Correct – 1) CETCs at 40 °F have been disabled, and the CETS at 2300 °F has also failed. Train "A" =4, and Train "B" =5 (correct). 2) IAW TS 3.7.E, Train "B" has less than 2 Operable CETCs, thus a 30 day clock is in effect (correct).
D. Incorrect – 1) Train 'A' has 4 failed CETCs (not 3). Plausible if the operator doesn't include the CETC indicating 2300 °F which once disabled by I&C will indicate 40 °F (incorrect). Also

B train has 5 failed CETCs not 2. 2) With the “B” Train having less than 2 CETCs Operable, a 30 day clock is in effect (correct).

References for Student:

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability

Applies to reactor and safety features instrumentation systems.

Objectives

To ensure the automatic initiation of the Reactor Protection System and the Engineered Safety Features in the event that a principal process variable limit is exceeded, and to define the limiting conditions for operation of the plant instrumentation and safety circuits necessary to ensure reactor and plant safety.

Specification

A. The Reactor Protection System instrumentation channels and interlocks shall be OPERABLE as specified in Table 3.7-1.

B. The Engineered Safeguards Actions and Isolation Function Instrumentation channels and interlocks shall be OPERABLE as specified in Tables 3.7-2 and 3.7-3, respectively.

C. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 3.7-4.

D. The explosive gas monitoring instrumentation channel shown in Table 3.7-5(a) shall be OPERABLE with its alarm setpoint set to ensure that the limits of Specification 3.11.A.1 are not exceeded.

   1. With an explosive gas monitoring instrumentation channel alarm setpoint less conservative than required by the above specification, declare the channel inoperable and take the action shown in Table 3.7-5(a).

2. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the action shown in Table 3.7-5(a). Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission (Region II) to explain why the inoperability was not corrected in a timely manner.

E. Prior to the Reactor Coolant System temperature and pressure exceeding 350°F and 450 psig, respectively, the accident monitoring instrumentation listed in Table 3.7-6 shall be OPERABLE in accordance with the following:

1. With one required channel inoperable, either restore the inoperable channel to OPERABLE status within 30 days or submit a report to the NRC within the next 14 days. The report shall outline the cause of inoperability and the plans and schedule for restoring the inoperable channel to OPERABLE status.

2. With two required channels inoperable, either:

   a. Restore an inoperable channel(s) to OPERABLE status within 7 days or initiate the preplanned alternate method of monitoring the appropriate function and submit a report to the NRC within the next 14 days. The report shall outline the preplanned alternate method of monitoring the function, the cause of inoperability, and the plans and schedule for restoring an inoperable channel to OPERABLE status.

   b. If no preplanned alternate method of monitoring the function is available, restore an inoperable channel(s) to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 6 hours and be less than 350°F and 450 psig within the following 12 hours.

F. Two manual actuation trains of the Main Control Room/Emergency Switchgear Room (MCR/ESGR) Envelope Isolation Actuation Instrumentation shall be OPERABLE whenever:

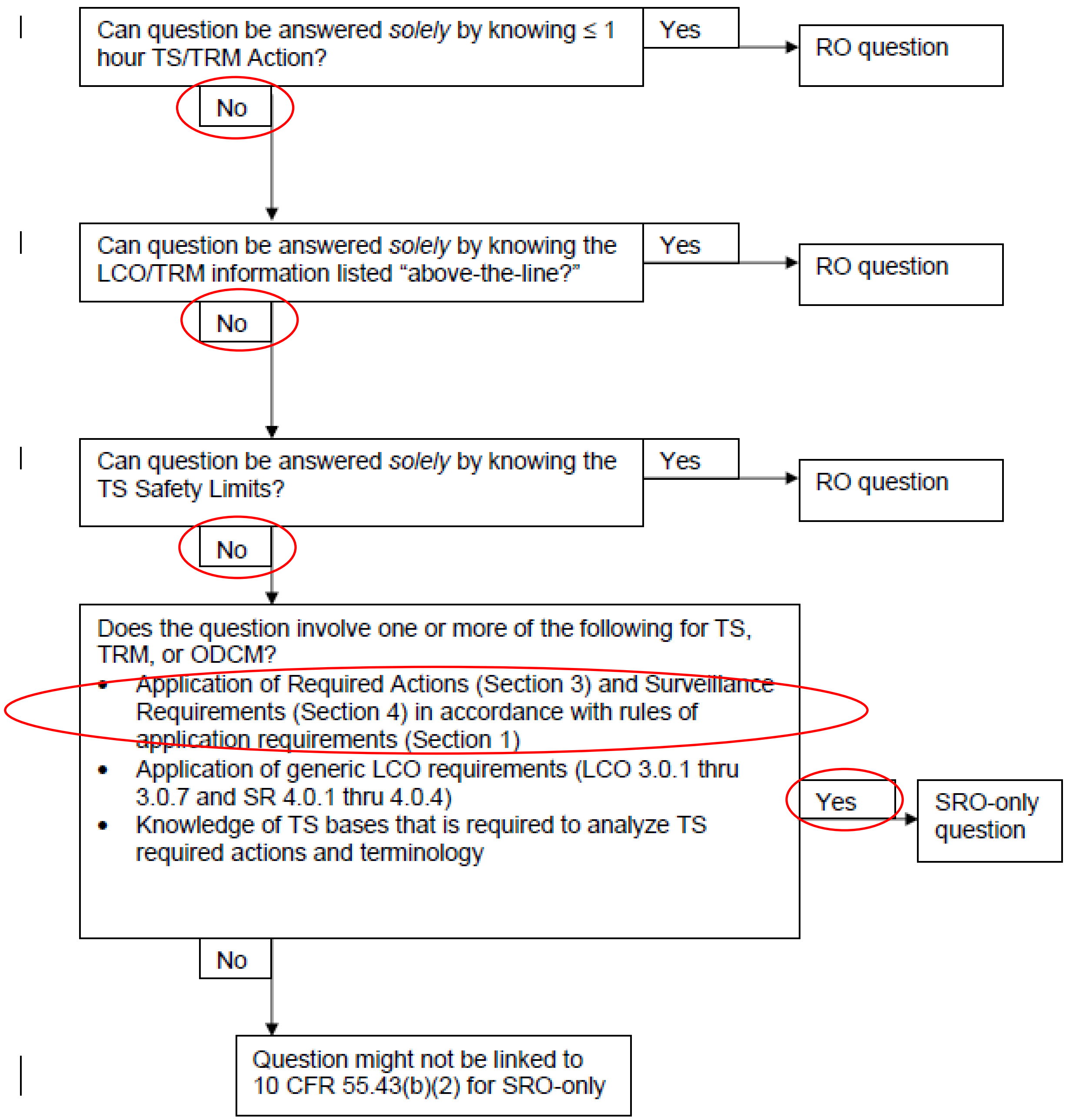
- T_{avg} (average Reactor Coolant System (RCS) temperature) exceeds 200°F, or
- During movement of irradiated fuel.

Note: Automatic actuation of the MCR/ESGR Envelope Isolation Actuation Instrumentation is addressed as part of the Safety Injection Instrument Operating Conditions included in TS Table 3.7-2, "Engineered Safeguards Action Instrument Operating Conditions," Functional Unit No. 1.

1. For unit operation when T_{avg} exceeds 200°F:

   a. With one train inoperable, isolate the MCR/ESGR envelope normal ventilation within seven (7) days or be in at least HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: 022AA2. 01, Loss of Rx Coolant Makeup I 2, Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: (CFR 43.5/ 45.13) Whether charging line leak exists.

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.2 IR-SRO: 3.8

Proposed Question:

Unit 1 is operating at 100% when the following alarms and/or indications are noted:

- Pressurizer level is 53% and lowering slowly.
- Annunciator 1D-E5, CHG PP TO REGEN HX HI-LO FLOW is LIT.
- Annunciator 1D-F3, REGEN HX LETDOWN LINE HI TEMP is LIT.
- 1-RM-RR-131A, Vent Stack #2 RM, is in Alert.
- CHG Line Flow, 1-CH-FI-1122, indicates 110 gpm and rising.
- REGEN HX CHG Outlet Temp, 1-CH-TI-1123, indicates 458 °F and rising.
- Letdown Line Flow, 1-CH-FI-1150, oscillating between 90 and 110 gpm.
- Seal Injection flow to "A", "B", and "C" RCPs indicate 7.5 gpm and slowly lowering.
- Containment Sump Level indicates 23% and stable.

Answer the following questions:

1) Where is the location of the leak in the Chemical and Volume Control System?

2) Make an immediate determination of operability of the High Head Safety Injection system in accordance with OP-AA-102, Operability Determination.

A. 1) Downstream of 1-CH-FCV-1122, Charging flow control valve.
   2) Operable.

B. 1) Downstream of 1-CH-HCV-1186, Seal Injection flow control valve.
   2) Operable.

C. 1) Downstream of 1-CH-FCV-1122, Charging flow control valve.
   2) Inoperable.

D. 1) Downstream 1-CH-HCV-1186, Seal injection flow control valve.
   2) Inoperable.

Proposed Answer: A 1) Downstream of 1-CH-FCV-1122, Charging flow control valve. 2) Operable

Explanation: OP-AA-102 directs the Shift Manager (SRO) to make an immediate determination of operability if enough information exists. This determination is made in a controlled manner (OP-AA-102, step 3.2.2). Based on indications, the heat sink for the RHX has been lost (Normal Charging), leading to increased temperatures and flashing of letdown. Change in seal injection flows are insufficient to cause the magnitude of indication changes seen. High Head SI is operable as the leak location is

downstream of the HHSI flow path and is automatically isolated on a Safety Injectiion. Therefore the HHSI flow path is not affected.

Technical Reference: 1-OP-CH-006, Shifting or Increasing/Decreasing Letdown Flow. OP-AA-102, Operability Determination.

Reference Provided to Applicant: No

Learning Objective: ND-88.3-LP-2, Charging and Letdown, Objective B, Using a one-line diagram drawn from memory, describe the charging and letdown system flowpaths including all interconnections with other systems.

Question Source: Modified Bank (McGuire, 2005)

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

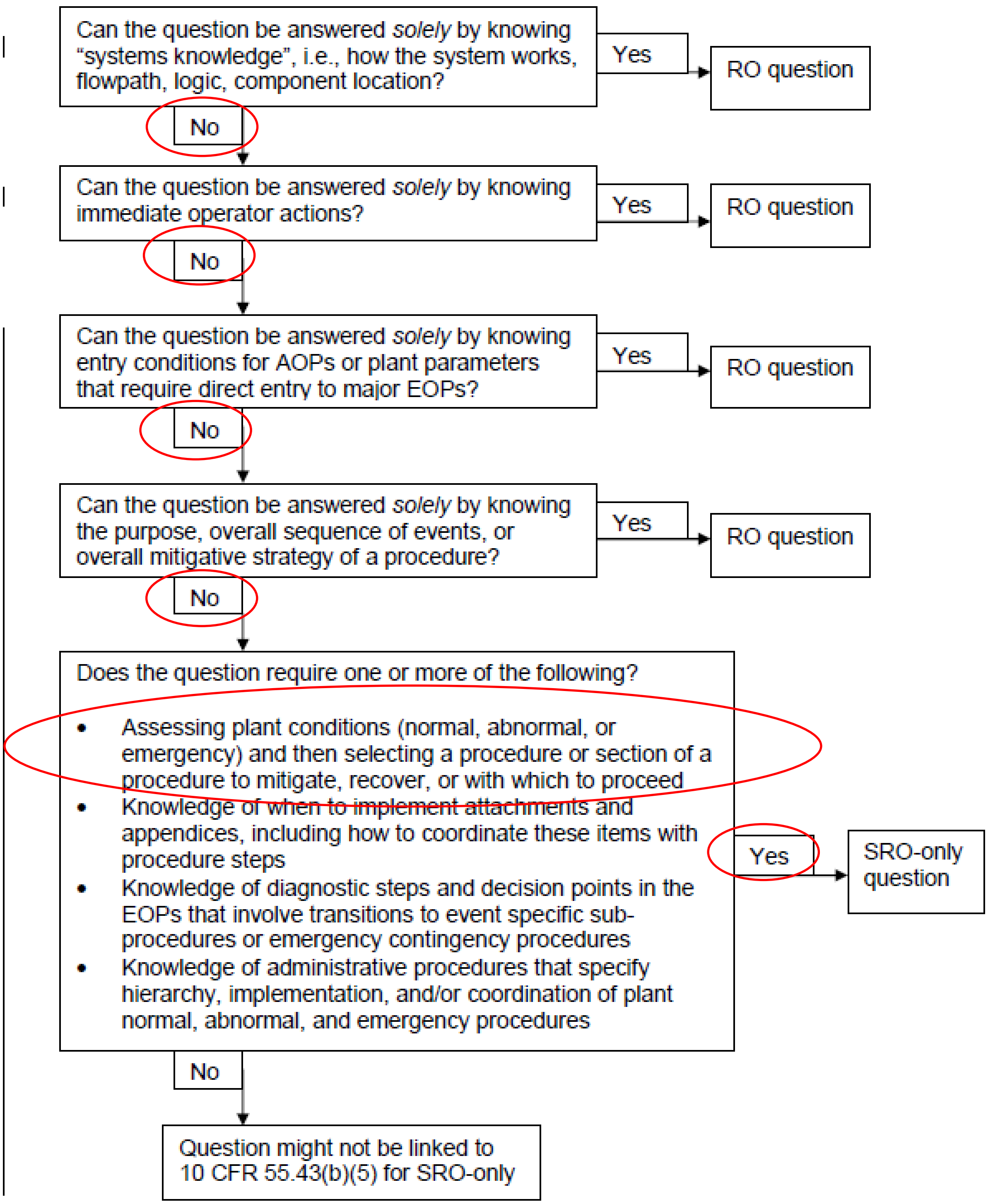
10 CFR Part 55 Content: (CFR 43.5/ 45.13) 10CFR55.43(b)(5)

Comments:

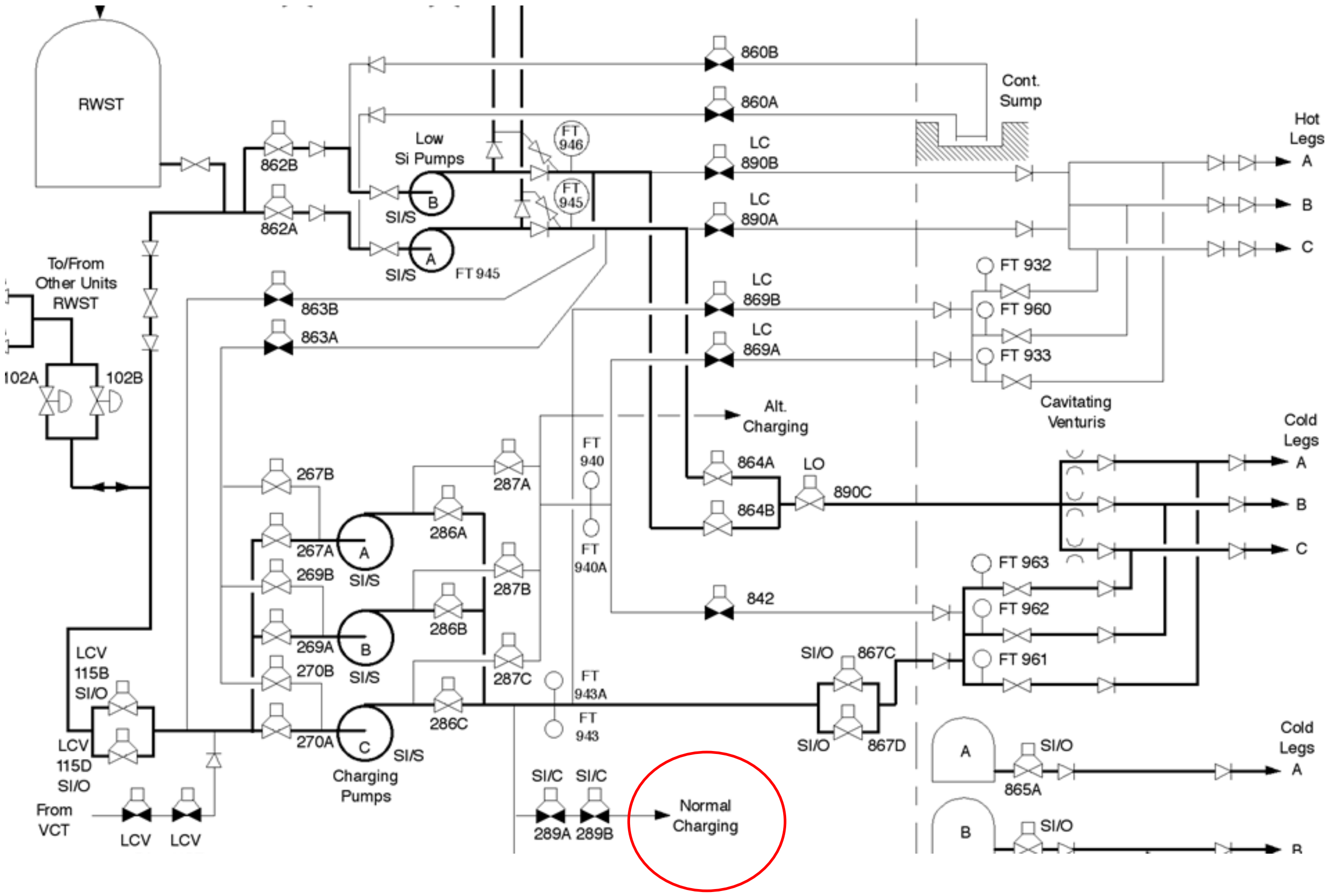
Distractor Analysis:

A. Correct – based on indications, the leak must be downstream of FCV-1122 to see the magnitude of the changes. High Head SI is operable because the flow path is not affected.
B. Incorrect- A change of 1.5 gpm (total) in seal injection will not cause the changes seen. The second part, High Head SI is operable is correct.
C. Incorrect – The location of the leak is correct but HHSI is not inoperable.
D. Incorrect – A change of 1.5 gpm (total) in seal injection will not cause the changes seen. The second part, High Head SI is inoperable is also incorrect.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



References: As can be seen by drawing of SI System; a leak downstream of Normal Charging line does not affect the SI Flow path therefore HHSI is operable.



K/A Number: 024AG2.1.23, Emergency Boration/I 1, Ability to perform specific system and integrated plant procedures during all modes of plant operation.(CFR: 41.10 / 43.2 / 45.6)

Level: SRO Tier #: 1 Group #: 2 IR – RO: 4.3 IR-SRO: 4.4

Proposed Question:

Initial Conditions for Unit 1:

- Unit 1 at 100% power.
- Delta Flux is at -2.4% with a target of -1%.
- Spurious Turbine Runback occurs causing Tave to increase rapidly.

Current Conditions:

- Reactor Power is 91% and stable.
- Delta Flux is at -16%.
- Tave is 571.5 ^{o}F, Tref is 571.0 ^{o}F.
- Annunciator 1E-E3, Delta Flux Deviation is lit.
- Annunciator 1G-H8, Rod Bank D Extra Lo Limit.

The SRO directs Emergency Boration to be performed per 1-AP-3.00, Emergency Boration. When the RO attempted to open 1-CH-MOV-1350, Emergency Borate MOV; the MOV immediately thermalled.

Based on the current conditions, which ONE of the following states:

1) The **next** actions to be taken to start Emergency Boration in accordance with 1-AP-3.00.
2) The **most restrictive** LCO, and basis for this CONDITION?

A. 1) Manually open 1-CH-FCV-1113A, and locally open 1-CH-228.
   2) Insertion limits to ensure adequate shutdown margin on reactor trip.

B. 1) Manually align Charging pump suction to the RWST.
   2) Insertion limits to minimize the effects of Xenon redistribution during load-follow maneuvers.

C. 1) Manually open 1-CH-FCV-1113A, and locally open 1-CH-228.
   2) Delta flux to minimize the effects of Xenon redistribution during load-follow maneuvers.

D. 1) Manually align Charging pump suction to the RWST.
   2) Delta flux to ensure adequate shutdown margin on reactor trip.

Proposed Answer: C. 1) Manually open 1-CH-FCV-1113A, and locally open 1-CH-228. 2) Minimize the effects of Xenon redistribution during load-follow maneuvers.

Explanation: 1) 1-AP-3.00 specifies manually opening 1-CH-FCV-1113A and 1-CH-228 if 1-CH-MOV-1350 does not open. If neither valve can be open then the procedure directs aligning charging pump

suction to the RWST. 2) The Delta Flux LCO is not met with Delta flux at -9.0. With power above 90% the requirement is to restore axial flux to the band within 15 minutes or be below 90%. The basis for the Delta Flux LCO is to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.

Technical Reference: 1-AP-3.00, Emergency Boration, Tech Specs 3.12.

Reference Provided to Applicant: No

Learning Objective: ND-93.2-LP-4, Power Range Nuclear Instruments, Objective D, Explain the meaning of the "Delta Flux" indication, including the limitations imposed upon it by Technical Specifications.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.2 / 45.6)

Comments:

Distractor Analysis:

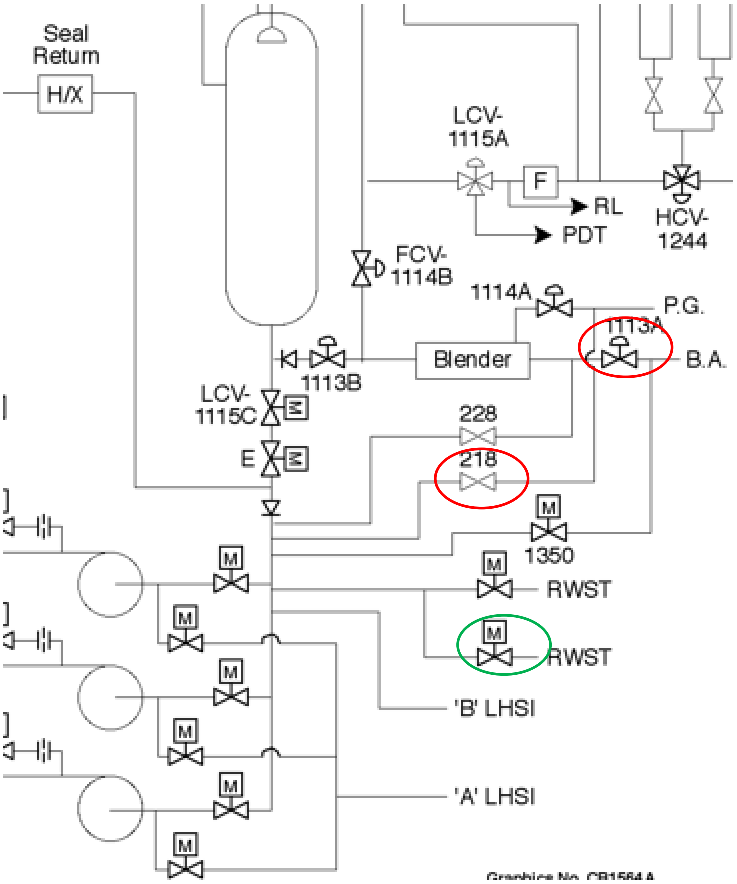
A. Incorrect –. Part 1 is correct. Part 2 is incorrect. Ensure adequate Shutdown Margin is the basis for Control Rod Insertion limits. This is plausible because the conditions given do imply Control Bank is below the insertion limit but this LCO has a 1-hour time limit to verify shutdown margin and a 2-hour limit to restore control banks to within limits. The SRO candidate must prioritize the situation, and determine the most limiting LCO and basis for that LCO. The Delta Flux LCO has a 15 minute limit therefore it is the most limiting LCO.
B. Incorrect –. Part 1 is incorrect because 1-AP-3.00 states that 1-CH-FCV-1113A and 1-CH-228 should be open next. This path is a redundant path that uses the same source for Boric Acid as 1-CH-MOV-1350. The RWST path is only used if *neither* path is available. This is plausible because this is a viable path in the procedure but the question is asking for the NEXT action. Part 2 is incorrect. Insertion limits to minimize the effects of Xenon redistribution during load-follow maneuvers. This is the wrong LCO and the Basis is also not the basis for Insertion limits. This is plausible because the conditions given do imply Control Bank is below the insertion limit but this LCO has a 1-hour time limit to verify shutdown margin and a 2-hour limit to restore control banks to within limits. The SRO candidate must prioritize the situation, and determine the most

limiting LCO and basis for that LCO. The Delta Flux LCO has a 15 minute limit therefore it is the most limiting LCO.

C. Correct.

D. Incorrect: – Part 1 is incorrect because 1-AP-3.00 states that 1-CH-FCV-1113A and 1-CH-228 should be open next. This path is a redundant path that uses the same source for Boric Acid as 1-CH-MOV-1350. The RWST path is only used if *neither* path is available. This is plausible because this is a viable path in the procedure but the question is asking for the NEXT action. Part 2 is also incorrect because the LCO Basis is not the correct Basis for Delta flux.

References for Part 1. Figure below shows flowpath for Emergency Boration. AP-3.00 shows if 1-CH-MOV-1350 cannot be opened then next step is to open 1-CH-FCV-1113A, and 1-CH-228. If neither valve can be opened then align charging pump suction to RWST.



2. ___ START EMERGENCY BORATION

☐ a) Transfer the in-service BATP to FAST	a) Manually align CHG pump suction to the RWST:
	☐ 1) Open 1-CH-MOV-1115B and D.
	☐ 2) Close 1-CH-MOV-1115C and E.
	☐ 3) GO TO Step 5.
☐ b) Open 1-CH-MOV-1350	☐ b) Locally open 1-CH-MOV-1350.
	IF 1-CH-MOV-1350 can NOT be opened, THEN do the following:
	☐ 1) Manually open 1-CH-FCV-1113A.
	☐ 2) Locally open 1-CH-228.
	☐ 3) Monitor Boric Acid flow on FR-1-113 (red trace).
	☐ 4) GO TO Step 3.
	☐ IF neither valve can be opened, THEN manually align CHG pump suction to the RWST AND GO TO Step 5.

References for part 2, Tech spec basis for Delta Flux (Tech Specs page 3.12-18).

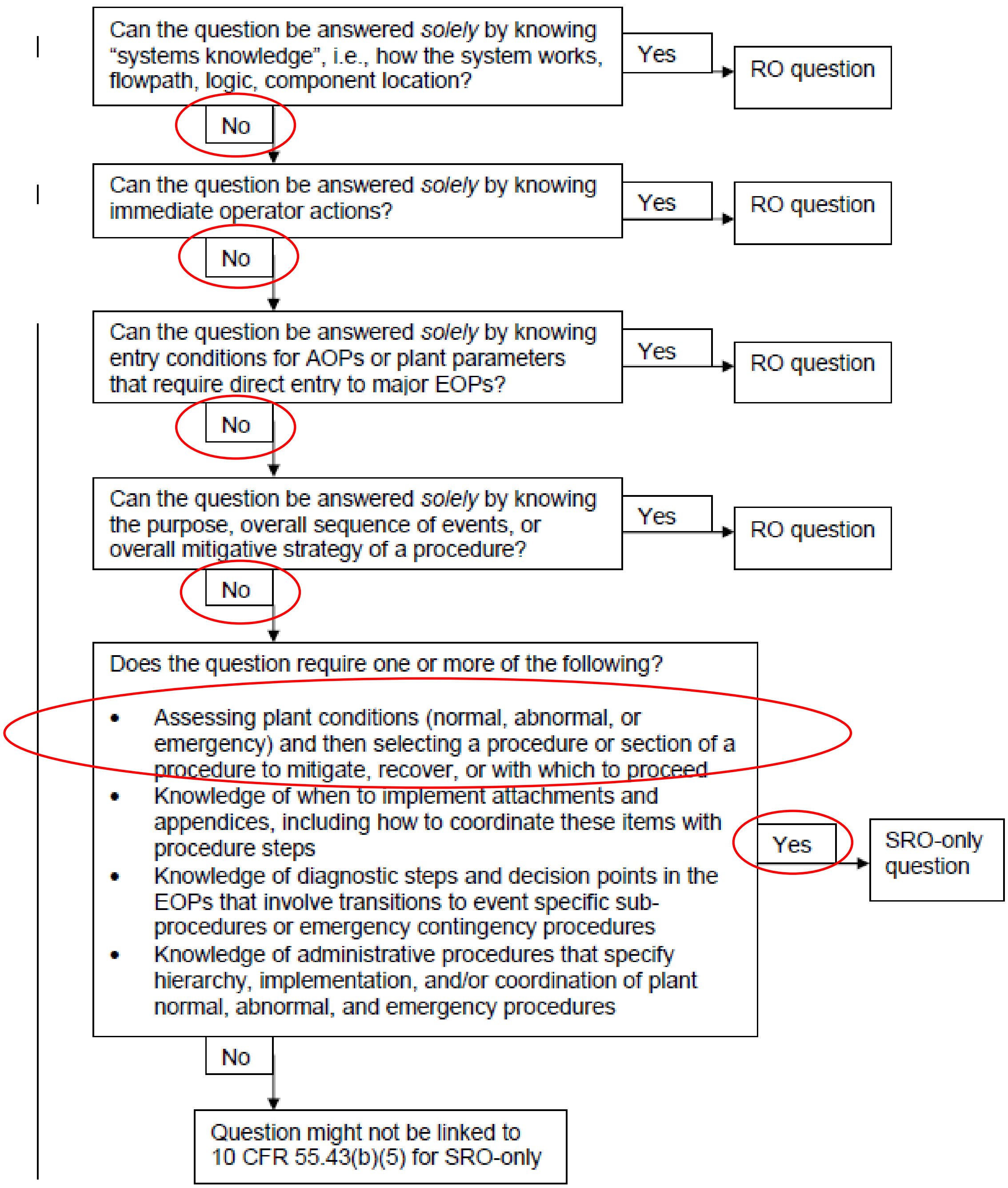
The procedures for axial power distribution control are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers. Basically, control of flux difference is required to limit the difference between the current value of flux difference (ΔI) and a reference value which corresponds to the full power equilibrium value of axial offset (axial offset = ΔI/fractional power). The reference value of flux difference varies with power level and burnup, but expressed as axial offset it varies only with burnup.

References for part 2, distractor; Tech Spec basis for Control Bank insertion limit.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to COLD SHUTDOWN) are compensated for by changes in the soluble boron concentration. During POWER OPERATION, the shutdown control rod assemblies are fully withdrawn and control of power is by the control banks. A reactor trip occurring during POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit startup and operating data to provide a more realistic limit which will allow for more flexibility in unit operation and still assure compliance with the shutdown requirement.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: 027AA2.10, Pressurizer Pressure Control System Malfunction / 3, Ability to determine and Interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:, (CFR: 43.5 / 45.13) PZR heater energized/de-energized condition

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.3 IR-SRO: 3.6

Proposed Question:

Unit 1 is operating at 100% Power and Normal Operating Pressure.

“C” and “D” pressurizer heater banks control switches are in ON; “A”, “B”, and “E” pressurizer heater control switches are in “AUTO” with the Green breaker open indicating light LIT.

The Master Pressure Controller malfunctions, causing a step change in the **setpoint** to 2310 psig which causes the Master Pressure Controller output to decrease to 0%.

The response of the “A”, “B”, and “E” pressurizer heater to this event is to ____(1)___. During power operation at least 125 KW of heaters must be supplied electrical power from an emergency bus in order to provide assurance that these heaters will remain energized _____(2)_____.

A. 1) energize
   2) to ensure the Reactor Coolant System will not be solid when criticality is achieved

B. 1) energize
   2) during a loss of offsite power to maintain natural circulation at HOT SHUTDOWN

C. 1) remain deenergized
   2) during a loss of offsite power to maintain natural circulation at HOT SHUTDOWN

D. 1) remain deenergized
   2) to ensure the Reactor Coolant System will not be solid when criticality is achieved

Proposed Answer: B

Explanation: When controller setpoint changes to 2310 psig, Master Pressure Controller output will decrease to 0%. This causes A/B/E pressurizer heater banks which are in AUTO to energize, and the C heater bank to go to maximum output. Per TS 3.1.A.5 basis, the requirement for 125 KW of operable heaters powered from the emergency bus it to provide assurance that these heaters will remain energized during a loss of offsite power to maintain natural circulation at HOT SHUTDOWN. The A, and E heaters are powered from emergency buses (J, H). The requirement for a bubble in the pressurizer with pressurizer heaters and spray available is to ensure the reactor is not critical with the pressurizer solid.

Technical Reference: TS 3.1.A basis.

Reference Provided to Applicant: No

Learning Objective: ND-93.3-LP-5, PZR Press Control, Objective B, Using a one-line diagram, drawn from memory, describe all design characteristics of the Pressurizer Pressure Control System, including setpoints, controls, interlocks, inputs, and outputs. ND-88.1-LP-9, Tech Specs, Objective H, Describe the RCS Tech Specs, including for the SRO candidate, the basis behind each specification.

Question Source: Modified Bank , PPC00017 Surry

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13) 10 CFR 55.43(b)(2)

Comments:

Distractor Analysis:

A. 1) Correct, A/B/E heaters will energize because they are in AUTO.

2) Incorrect – Response linked to TS-3.1.A basis regarding requirement to maintain reactor subcritical by 1% until a steam bubble is established. This basis relates to pressurizer heaters in general and this distractor is plausible if the operator confuses the basis for maintaining a bubble with requirement for 125 KW.

B. 1) Correct

2) Correct – Linked to TS-3.1.A basis

C. 1) Incorrect – All Pzr heaters will energize. Common misconception when Pressure setpoint is confused with *actual pressure.* Plausible because if operator confuses actual pressure with pressure setpoint then it is true that A/B/E pressurizer heaters will de-energize.

2) Correct – linked to TS 3.1.A basis

D. 1) Incorrect - All Pzr heaters will energize. Common misconception when Pressure setpoint is confused with actual pressure. Plausible because if operator confuses actual pressure with pressure setpoint then it is true that C pressurizer heaters will go to minimum.

2) Incorrect – Response linked to TS-3.1.A basis regarding requirement to maintain reactor subcritical by 1% until a steam bubble is established. This basis relates to pressurizer heaters in general and this distractor is plausible if the operator confuses the basis for maintaining a bubble with requirement for 125 KW.

References for correct answer: If setpoint fails high then output will go low causing Proportional heaters to be full ON and Backup heaters (A,B,E) to be ON.

PT-444 Master Controller Output Pressure Control

% Controller Output	* Ref. Pressure, psig*	Function
80	2335	PCV-1455C PORV Operation
67.5	2310	High Pressure Alarm
67.5**	2305	Spray Valves Open
42.5**	2255	Spray Valves Start Open
37.5	2250	Proportional Heaters Off
30	2235	NOP
22.5	2220	Proportional Heaters Full On
17.5	2210	Backup Heaters On
20	2215	Backup Heaters Off

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the Reactor Coolant System average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two OPERABLE steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for Reactor Coolant System temperatures of greater than 350°F.

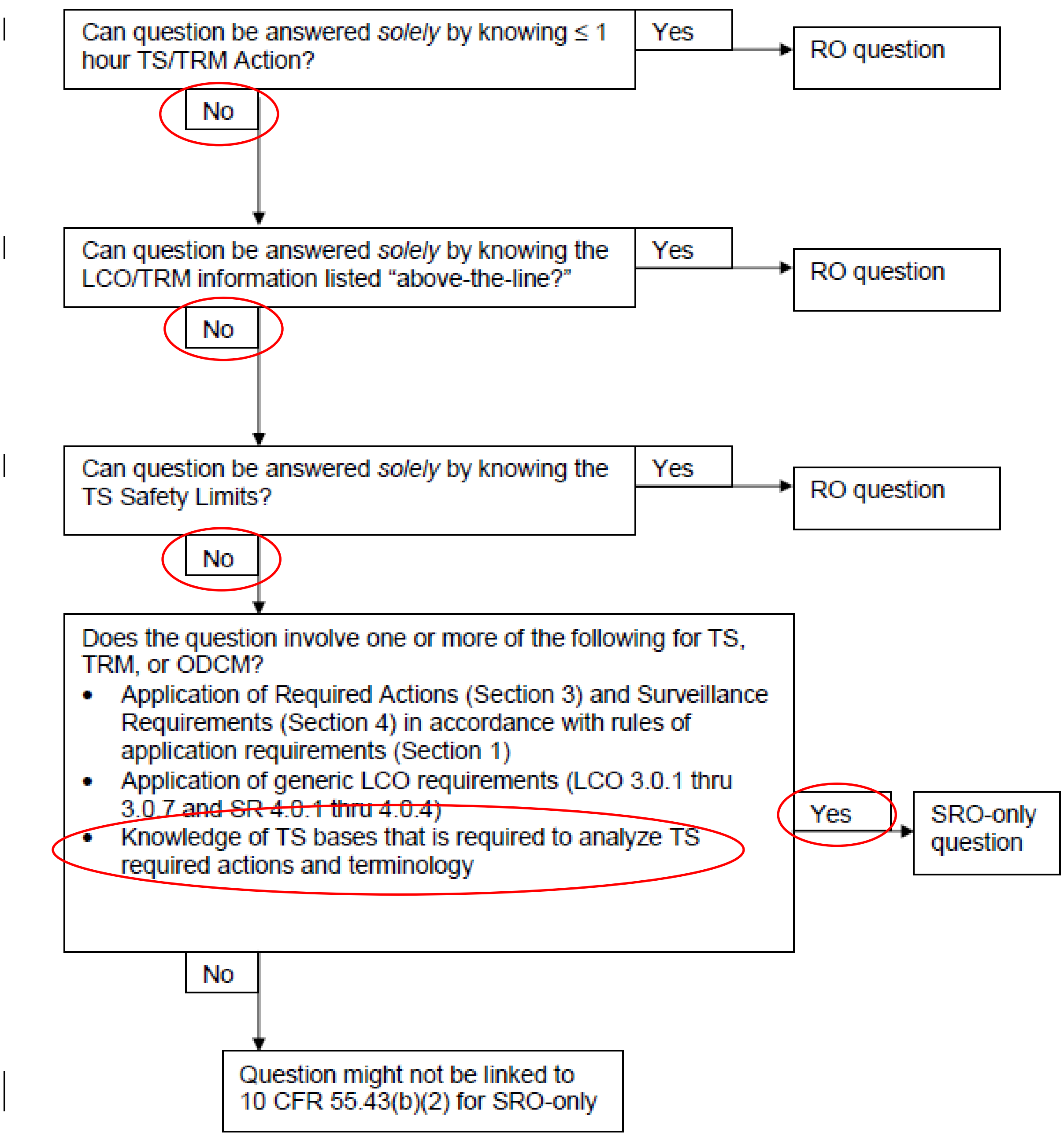
Each of the pressurizer safety valves is designed to relieve a minimum of 293,330 lbs. per hr. of saturated steam at the valve setpoint. Three safety valves have a combined capacity greater than the maximum surge rate resulting from UFSAR analyzed accidents.[2]

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

The requirement for steam bubble formation in the pressurizer when the reactor passes 1% subcriticality will ensure that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that 125 Kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT SHUTDOWN.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: 033AG2.2.44, Loss of Intermediate Range NI / 7, Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions, (CFR: 41.5 / 43.5 / 45.12)

Level: SRO Tier #: 1 Group #: 2 IR – RO: 4.2 IR-SRO: 4.4

Proposed Question:

Given the following:

- Unit 1 is performing a plant startup in accordance with 1-GOP-1.5, Unit Startup, 2% Reactor Power to Max Allowable Power.
- The Main Generator has been placed on line.
- Reactor power is 13% and rising.
- Permissive status light B2, P-7 NIS PWR RGE and TURB PWR < 10%, is NOT lit.
- Permissive status light B1, NIS INT RNG RX TRIP and ROD STOP BLOCKED is lit.
- Permissive status light C1, NIS PWR RNG LO SP TRIP BLOCKED is lit.
- Annunciator 1E-G9, NIS INT RNG HI FLUX has just alarmed.
- Reactor Operator reports that N-36 has failed HIGH.
- The SRO enters 1-AP-4.00 Nuclear Instrument Malfunction.

Which ONE of the following identifies:

1) The actions required to be taken for this event in accordance with 1-AP-4.00.
2) Determine Tech Spec actions (if any).

A. 1) Place N36 Level Trip Switch in the Bypass position, remove Control Power Fuses.
   2) The LCO is not met. Verify by permissive status light, that the interlock is in required state within one hour.

B. 1) Place N36 Level Trip Switch in the Bypass position
   2) The LCO is not met. Verify by permissive status light, that the interlock is in required state within one hour.

C. 1) Place N36 Level Trip Switch in the Bypass position, remove Control Power Fuses.
   2) The LCO is met.

D. 1) Place N36 Level Trip Switch in the Bypass position.
   2) The LCO is met.

Proposed Answer: B

Explanation: 1) With the indications provided, N36 channel is Inoperable. AP-4.00 would be initiated and the Level Trip Bypass Switch placed in Bypass, and TS Table 3.7-1, items 3 and 13 referenced. Pulling the fuses is performed in AP-4.00 for a Power Range Channel Failure, not an IR channel failure. Table 3.7-1 item 3, Intermediate Range Hi Flux is met as indications given provide an acceptable bypass operation, however Table 3.7-1 item 20, Reactor Trip System interlocks are not met until the

interlocks are verified. This requires a detailed SRO knowledge as to what functions are required in Table 3.7-1 and what actions need to be taken.

Technical Reference: TS Table 3.7-1, Item 3, and item 20. Operator Action 3 and 13. 1-AP-4.00, Step 10-14, and Attachment 2.

Reference Provided to Applicant: No

Learning Objective: ND-93.2-LP-3, Objective D, Explain the operation of the Intermediate Range detection system during both normal and abnormal operating conditions. ND-93.2-ST-3.1, Objective C, Given the loss of one or both IR channels above or below P-10, describe the use of AP-4.00 to address the failure(s).

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.12) 10 CFR 55.43 (b)(2)

Comments:

Distractor Analysis:

A. Incorrect – 1) Incorrect, Power range requires fuses removed, not Intermediate Range. 2) Correct.

B. Correct (see explanation)

C. Incorrect – 1) Incorrect, Power range requires fuses removed, not Intermediate Range. 2) Incorrect because Table 3.7-1, item 20 Reactor Trip Interlocks require action 13 which specifies checking interlocks in required state within 1 hour. This is plausible because item 3 is satisfied by the P-10 bypass conditions being met.

D. Incorrect – 1) Correct. 2) Incorrect because Table 3.7-1, item 20 Reactor Trip Interlocks require action 13 which specifies checking interlocks in required state within 1 hour. This is plausible because item 3 is satisfied by the P-10 bypass conditions being met.

References for part 1)

NUMBER 1-AP-4.00	ATTACHMENT TITLE INTERMEDIATE RANGE FAILURE	ATTACHMENT 2
REVISION 30		PAGE 1 of 1

NOTE: 1-OPT-RP-001, Check Of Permissive Status Lights P-6, P-7, P-8 AND P-10, should be used to aid in Permissive Status Light verification.

AVG POWER GREATER THAN 11%

☐ 1. Continue unit operation.

NOTE: If the POWER ABOVE P-6 light is LIT, and stays LIT after placing LEVEL TRIP switch for the failed channel(s) in BYPASS, AND a plant shutdown or trip occurs, the Source Range NIs will not automatically energize.

☐ 2. Place the LEVEL TRIP switch for the failed channel(s) in BYPASS.

☐ 3. Refer to Tech Spec Table 3.7-1, Item 3. (Operator Action 3)

☐ 4. Refer to Tech Spec Table 3.7-1, Item 20. (Operator Action 13)

NUMBER 1-AP-4.00	ATTACHMENT TITLE POWER RANGE FAILURE	ATTACHMENT 1
REVISION 30		PAGE 2 of 4

NOTE: Annunciator NIS PWR RNG HI STPT (1E-E5, 1E-F5, 1E-G5, or 1E-H5) for the channel being placed in trip, NIS PWR RNG LOSS OF DET VOLT (1G-C3), and NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC (1G-H1) will alarm when the instrument power fuses are pulled.

If Reactor power is less than 10%, annunciator NIS PWR RNG LO STPT HI FLUX (1E-D5) will alarm when the instrument power fuses are pulled.

2. Place the failed Power Range channel in trip IAW the following:

___ a. At the Power Range drawer, remove the INSTRUMENT POWER fuses.

___ b. At the Power Range drawer, put the POWER RANGE TEST switch in the TEST position.

___ c. Check annunciator 1G-H1, NIS DROPPED ROD FLUX DECREASE > 5% PER 2 SEC - LIT.

___ d. Check annunciator 1G-C3, NIS PWR RNG LOSS OF DET VOLT - LIT.

___ e. IF Reactor power less than 10%, THEN check annunciator 1E-D5, NIS PWR RNG LO STPT HI FLUX - LIT.

References for part 2. Item 3 is met therefore distractors “LCO is met” Is plausible

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
1. Manual	2	2	1		1
2. Nuclear Flux Power Range*	4	3	2	Low trip setting at P-10	2
3. Nuclear Flux Intermediate Range*	2	2	1	P-10	3

b. Above the P-6 (Block of Source Range Reactor Trip) setpoint, but below 11% of RATED POWER, within 24 hours, decrease power below P-6 or increase THERMAL POWER above 11% of RATED POWER.

TABLE 3.7-1
REACTOR TRIP
INSTRUMENT OPERATING CONDITIONS

Functional Unit	Total Number Of Channels	Minimum OPERABLE Channels	Channels To Trip	Permissible Bypass Conditions	Operator Action
20. Reactor Trip System Interlocks - Note D					
a. Intermediate range neutron flux, P-6	2	2	1		13

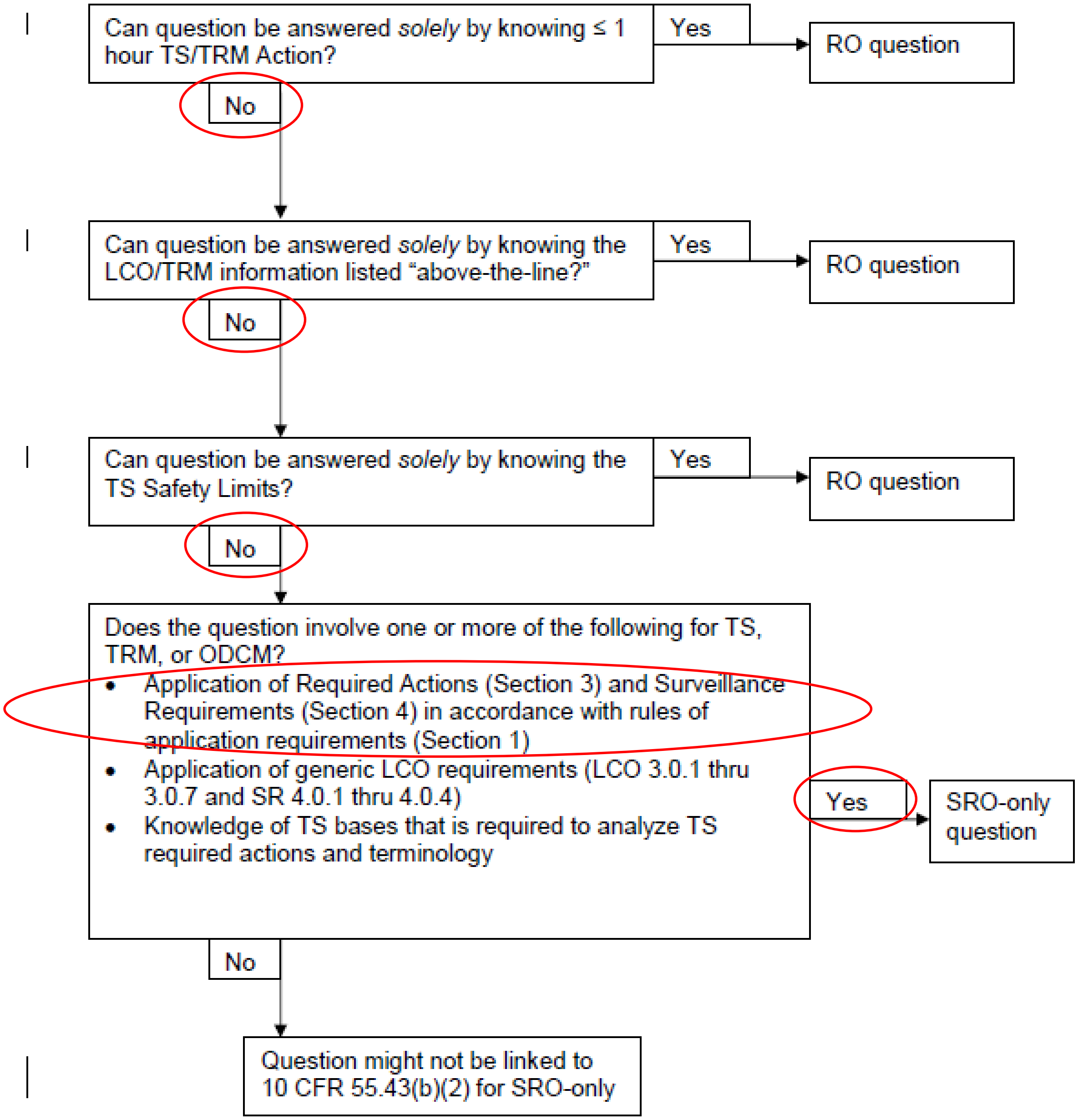
TABLE 3.7-1 (Continued)

TABLE NOTATION

ACTION STATEMENTS

ACTION 13. With the number of OPERABLE channels less than the Minimum OPERABLE Channels requirement, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or be in at least HOT SHUTDOWN within the next 6 hours.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: 014A2.05, Rod Position Indication / 1, Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip

(CFR: 41.5 / 43.5 / 45.3 / 45.13)

Level: SRO Tier #: 2 Group #: 2 IR – RO: 3.9 IR-SRO: 4.1

Proposed Question:

Unit 1 was tripped due to the loss of the “A” MFP at 100% power.

- The Team has completed E-0, Reactor Trip or Safety Injection; and ES-0.1, Reactor Trip Response.
- The Team has withdrawn the Shutdown Banks IAW 1-OP-RX-005, Rod Control System Withdrawal of the Shutdown Banks.
- The Team is preparing to withdraw Control Banks in accordance with 1-OP-RX-006, Withdrawal of the Control Banks to Critical Conditions, when the RO notes that “D” Bank Group Step Counters Indicate “14” on Group 1, and “0” on Group 2.
- The Team has entered 0-AP-1.00, Rod Control Malfunction, for the “D” Bank Group Step Counter.

Which ONE of the following identifies:

1) The maximum deviation between the Group Step Counters in accordance with Tech Specs.
2) The basis for Rod Position Indication.

A. 1) ± 2 Steps.
   2) Core peaking factors.

B. 1) ± 12 Steps.
   2) Core power distribution and Shutdown Margin.

C. 1) ± 2 Steps.
   2) Core power distribution and Shutdown Margin.

D. 1) ± 12 Steps.
   2) Core peaking factors.

Proposed Answer: C

Explanation: 1) Per TS 3.12.E.c; from movement of control banks to achieve criticality and Reactor Critical, the Bank Demand Position Indication System shall be operable and capable of determining group demand position to within 2 steps. 2) TS Basis states: “The requirements on the rod position indicators and the group step demand counters are only applicable from the movement of control banks to achieve criticality and with the REACTOR CRITICAL, because these are the only conditions in which

the rods can affect **core power distribution** and in which the rods are relied upon to provide required **shutdown margin**." (TS Basis page TS 3.12-14)

Technical Reference: Technical Specifications

Reference Provided to Applicant: NO

Learning Objective: None

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.3)

Comments:

A. 1) ± 2 Steps. This is correct.

2) Core peaking factors. This is incorrect and plausible if operator confuses basis for Rod position and Bank demand with Basis for Rod insertion limits which states that the Rod insertion limits provide a limit on maximum inserted rod worth in event of rod ejection and provide for acceptable peaking factors.

B. 1) ± 12 Steps. This is incorrect. If power is > 50% Rod Position Indication System shall be capable of determining rod position with 12 steps of bank demand. This is plausible if operator confuses requirement for moving rods to criticality with requirement for power > 50%.

2) Core power distribution and Shutdown Margin. This is correct.

C. 1) ± 2 Steps. This is correct (see explanation).

2) Core power distribution and Shutdown Margin. This is correct (see explanation).

D. 1) ± 12 Steps. This is incorrect. If power is > 50% Rod Position Indication System shall be capable of determining rod position with 12 steps of bank demand. This is plausible if operator confuses requirement for moving rods to criticality with requirement for power > 50%.

2) Core peaking factors. This is incorrect and plausible if operator confuses basis for Rod position and Bank demand with Basis for Rod insertion limits which states that the Rod insertion limits provide a limit on maximum inserted rod worth in event of rod ejection and provide for acceptable peaking factors.

References: For correct answer.

E. Rod Position Indication System and Bank Demand Position Indication System

1. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, rod position indication shall be provided as follows:

   a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

   b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise.

   c. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, the Bank Demand Position Indication System shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

The requirements on the rod position indicators and the group step demand counters are only applicable from the movement of control banks to achieve criticality and with the REACTOR CRITICAL, because these are the only conditions in which the rods can affect core power distribution and in which the rods are relied upon to provide required shutdown margin. The

References for distractors:

E. Rod Position Indication System and Bank Demand Position Indication System

1. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, rod position indication shall be provided as follows:

    a. Above 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 12 steps of their respective group step demand counter indications.

    b. From movement of control banks to achieve criticality up to 50% power, the Rod Position Indication System shall be OPERABLE and capable of determining the control rod assembly positions to within ± 24 steps of their respective group step demand counter indications for a maximum of one hour out of twenty-four, and to within ± 12 steps otherwise.

    c. From movement of control banks to achieve criticality and with the REACTOR CRITICAL, the Bank Demand Position Indication System shall be OPERABLE and capable of determining the group demand positions to within ± 2 steps.

POWER OPERATION will place the reactor into HOT SHUTDOWN. The control rod assembly insertion limits provide for achieving HOT SHUTDOWN by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit on the maximum inserted control rod assembly worth in the unlikely event of a hypothetical assembly ejection and provide for acceptable nuclear peaking factors. The limit may be determined on the basis of unit

K/A Number: 057AG2.1.28, Loss of Vital AC Inst. Bus/ 6, Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

Level: SRO Tier #: 1 Group #: 1 IR – RO: 4.1 IR-SRO: 4.1

Proposed Question:

Initial Conditions (0700):

- Unit 1 is operating at 100% power.
- UPS 1A1 Regulating Line Conditioner (RLC) is tagged out for maintenance and is scheduled to be returned to service at 1900.

Current Conditions (0720):

- Fault occurs during maintenance on UPS 1A1 inverter resulting in the following indications:
  - Annunciator 1K-A7, BATT SYSTEM 1A TROUBLE is lit.
  - Annunciator 1K-B8, UPS SYSTEM 1A TROUBLE is lit.
  - Annunciator 1K-G7, 480V BUS MCC BKR TRIP is lit.
  - Vital Bus I volts indicate 0.0 volts.
  - Operator reports that UPS 1A1 Inverter Output breaker has tripped.
- Operator reports that Breaker 14H13, MCC 1H1-2 Supply breaker has tripped open and has an overcurrent flag.
- Operator reports that "Alternate Source Supplying load light" is lit on the Static Switch Section for UPS 1A2.
- Crew has entered 1-AP-10.01, Loss of Vital Bus I.

Based on the conditions given, answer the following questions.

1) With the conditions given **Vital Bus III** is powered from the UPS 1A2 ___________?
2) What is the latest time **Vital bus I** must be re-energized, in order to comply with 1-AP-10.01?

A. 1) Battery Charger.
   2) 1320.

B. 1) Battery Charger.
   2) 0920.

C. 1) RLC.
   2) 1320.

D. 1) RLC.
   2) 0920.

Proposed Answer: D 1) RLC. 2) 0920.

Explanation: 1) With the Indications reported, Vital Bus I is de-energized and Vital Bus III has lost the power supply from its Battery Charger (1H1-2). Therefore the 1A2 Inverter that is normally supplied

from Battery Charger will shift to the RLC. 2) Per 1-AP-10.01 Note prior to Step 15: “A de-energized AC Vital Bus shall be re-energized within 2 hours OR the unit must be placed in Hot Shutdown within the next 6 hours. This requirement is not a Technical Specification (even though it sounds like one), but a commitment. (Ref: NRC GL 91-11)

Technical Reference: ARP 2K-A8, UPS System 1A Trouble. Step 4, Caution and RNO. 1-AP-10.01, Loss of Vital Bus I. Step 15 Note.

Reference Provided to Applicant: No

Learning Objective: ND-90.3-PP-5, Objective B, Describe the components and indications associated with an Uninterruptible Power Supply (UPS). SOER 83-03, Recommendation 11; and Objective E, Given a loss of a Vital or Semi-Vital bus, describe the actions taken IAW AP-10.01, 10.02, 10.03, 10.04, and/or 10.05 to address this loss. SOER 83-03, Recommendation 11 and SOER 81-02, Recommendation 5.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.7) 10 CFR 55.43 (b)(5)

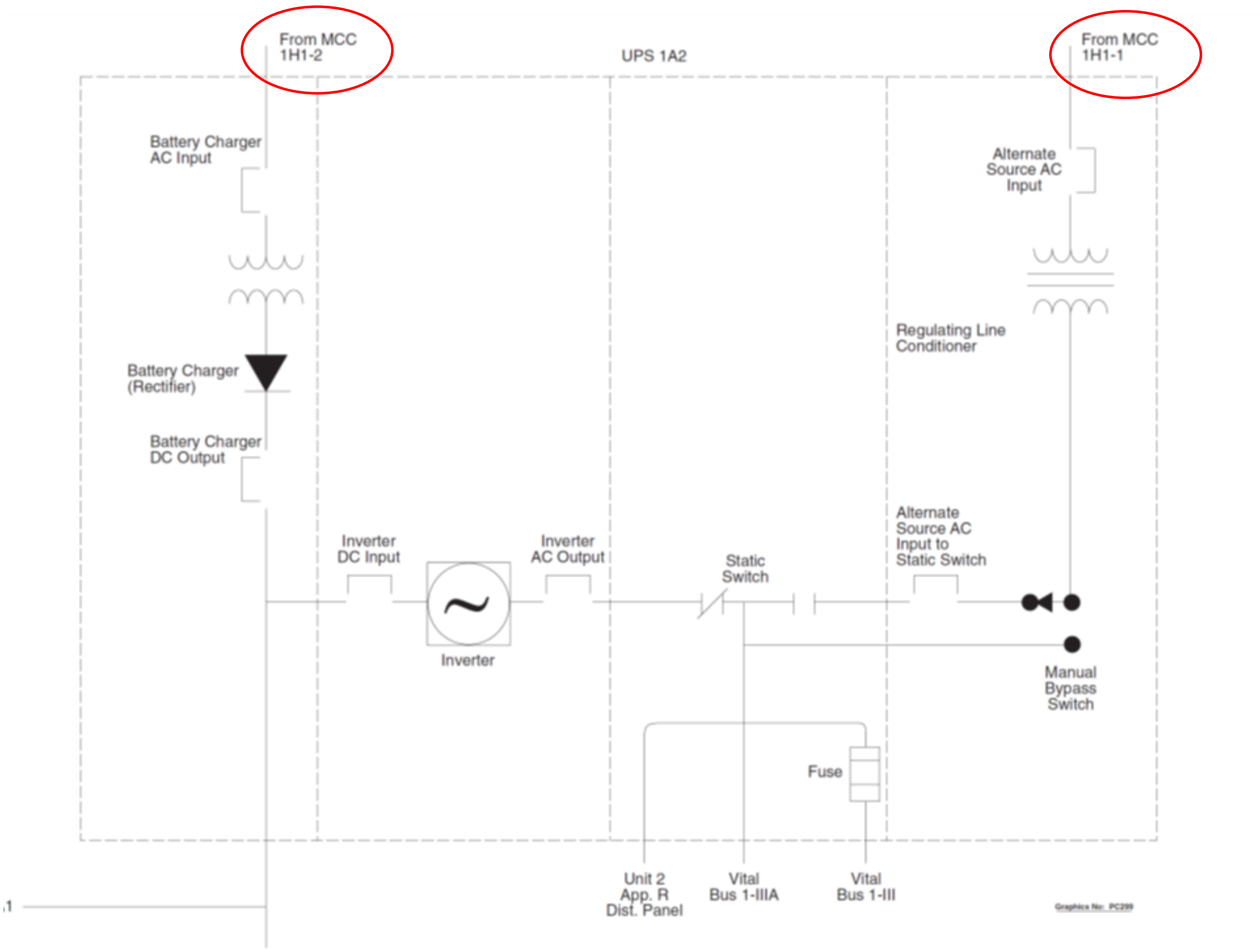
Comments:

Distractor Analysis:

A. Incorrect – 1) Incorrect. The Alternate Source is supplying the Static Switch as indicated by the Alternate Source to Load light being lit, making the power source incorrect. 2) Incorrect. The GL 91-11 clock to restore power to the Vital Bus should be 2 hours (0920).
B. Incorrect – 1) Incorrect. The Alternate Source is supplying the Static Switch as indicated by the Alternate Source to Load light being lit, making the power source incorrect. 2) Correct. 2 Hour GL 91-11 clock is correct.
C. Incorrect – 1) Power for the Vital Bus is correct. 2) Incorrect. The GL 91-11 clock to restore power to the Vital Bus should be 2 hours (0920).
D. Incorrect – 1) Power source listed is correct. 2) Correct. 2 Hour GL 91-11 clock is correct.

10 CFR 55.43 (b)(5)

References for part 1. MCC 1H1-2 is lost therefore Alternate source will supply Vital Buss through static switch.



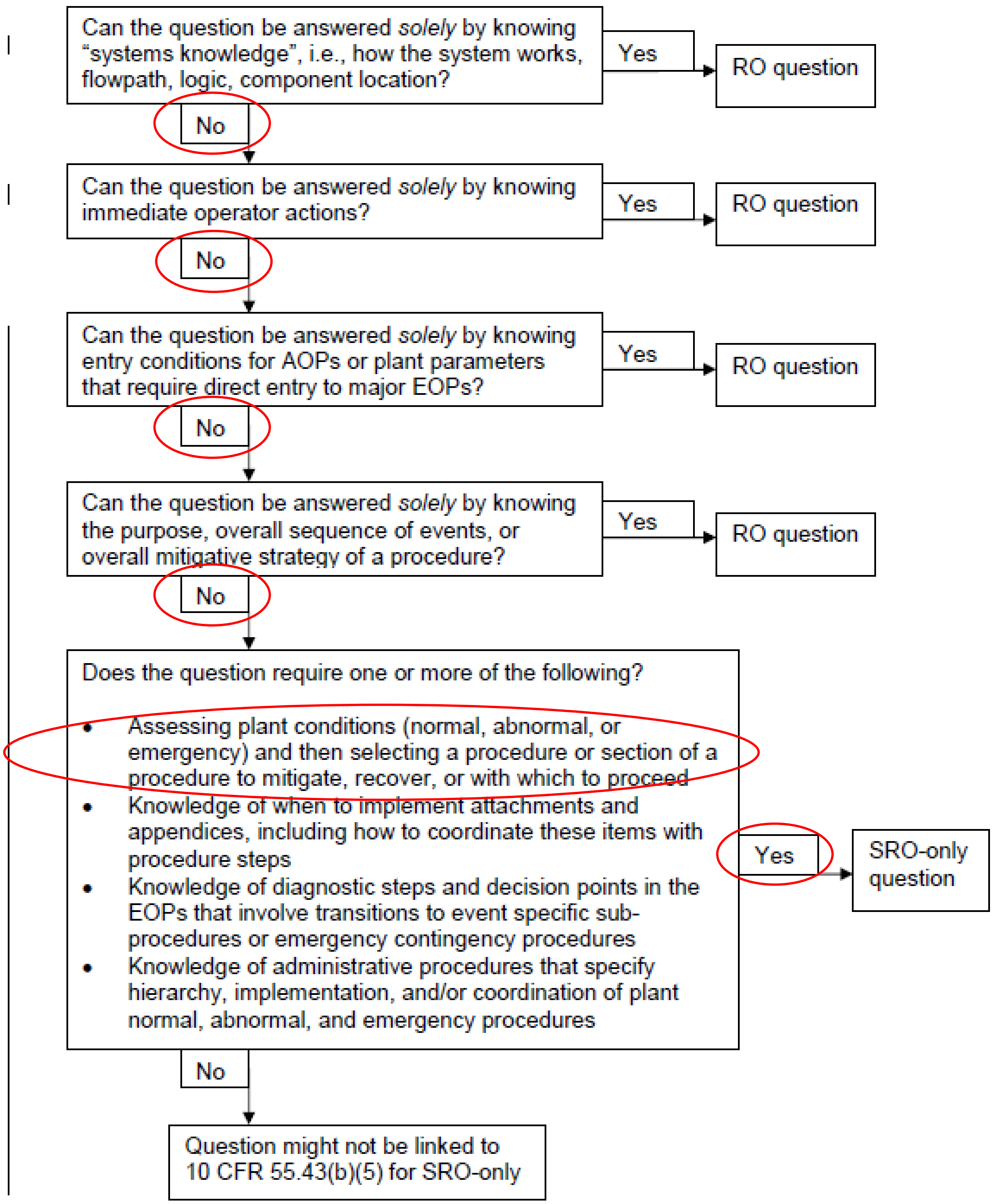
References for part 2.

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.01	LOSS OF VITAL BUS I	19
		PAGE 6 of 11

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- A de-energized AC Vital Bus shall be re-energized within 2 hours OR the unit must be placed in Hot Shutdown within the next 6 hours.
- Vital Bus 1-IA voltage can be read on PCS (ERF if not removed) Computer point V1VB002A.
- All Vital Bus voltages can be read on Group Review 25.
- Loss of Vital Bus 1-IA deenergizes ICCM Train A.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



10 CFR 55.43 (b)(5)

K/A Number: 059A2.01, Main Feedwater / 4S, Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:, (CFR: 41.5 / 43.5 / 45.3 / 45.13) Feedwater actuation of AFW system

Level: SRO   Tier #: 2   Group #: 1   IR – RO: 3.4   IR-SRO: 3.6

Proposed Question:

Unit 1 is currently at 61% power following a ramp down from 100% power so that 1-FW-P-1B, "B" Main Feed Pump can be secured for maintenance.

Per 1-OP-FW-004, Main Feedwater System Operation; 1-FW-MOV-150B, 1-FW-P-1B Discharge MOV, has just been closed.

Which ONE of the following identifies:

1) The Tech Spec Clock Duration that is entered when 1-FW-MOV-150B is closed.
2) The reason for the Tech Spec. clock.

A. 1) 72 hours.
   2) The TD AFW pump will not auto start if 1-FW-P-1A tripped under this condition.

B. 1) 48 Hours.
   2) The TD AFW pump will not auto start if 1-FW-P-1A tripped under this condition.

C. 1) 72 Hours.
   2) The MD AFW pumps will not auto start if 1-FW-P-1A tripped under this condition.

D. 1) 48 Hours.
   2) The MD AFW pumps will not auto start if 1-FW-P-1A tripped under this condition.

Proposed Answer: D

Explanation: When the discharge MOV on a running MFP is closed, the MFP supplying feed flow to the SGs could trip and the auto start matrix for the MD AFW pumps (1/2 breakers on 2/2/ MFPs) would not make up. The Unit would experience a loss of feed event with no AFW auto initiation signal – operator action would be required. 1-OP-FW-004 procedurally requires 48 hour TS clock prior to closing discharge valve of a Main Feed Pump (1-OP-FW-004, step 5.8.5).

Technical Reference: Technical Specification Table 3.7-2, Engineered Safeguards Action, Item 3.e, Action 24.

Reference Provided to Applicant: No

Learning Objective: ND-94-SP-1, Objective A, Given a condition where Tech Spec requires, make notifications and commence a Unit shutdown IAW GOP-2.1, Unit Shutdown, Power Decrease from Max Allowable Power to less than 30% Reactor Power.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13) 10 CFR 55.43(b)(5)

Comments:

Distractor Analysis

A. Incorrect – 1) Plausible if operator confuses AFW pump start criteria with FRV criteria. 72 hours would relate to FRV on Jack admin requirement per Table 3.7-3. 2) TDAFW auto start not impacted by MFP trip. Plausible if operator confuses TDAFW pump start with MDAFW pumps start.
B. Incorrect – 1) 48 hour clock correct. 2) TDAFW pump start incorrect. Plausible if operator confuses TDAFW pump start with MDAFW pumps start.
C. Incorrect – 1) Plausible if operator confuses AFW pump start criteria with FRV criteria. 72 hours would relate to FRV on Jack admin requirement per Table 3.7-3.
D. Correct – Both 48 hour clock requirement and auto start of MDAFW pump correct.

References:

DOMINION
Surry Power Station

1-OP-FW-004
Revision 19
Page 36 of 46

NOTE:
- Both breakers for 1-FW-P-1B, FW PUMP 1B, must be in TEST and closed to complete the logic to permit opening the Feed Pump Discharge MOV.
- AFW auto start interlocks are affected when both breakers for a Main Feedwater pump are closed in TEST or CONNECT and the discharge mov is closed. **(Reference 2.4.3)**

______ 5.8.5 IF the unit is greater than 350°F,

AND

both breakers on any MFW Pump are closed in CONNECT OR TEST,

WITH

the pump discharge valve closed,

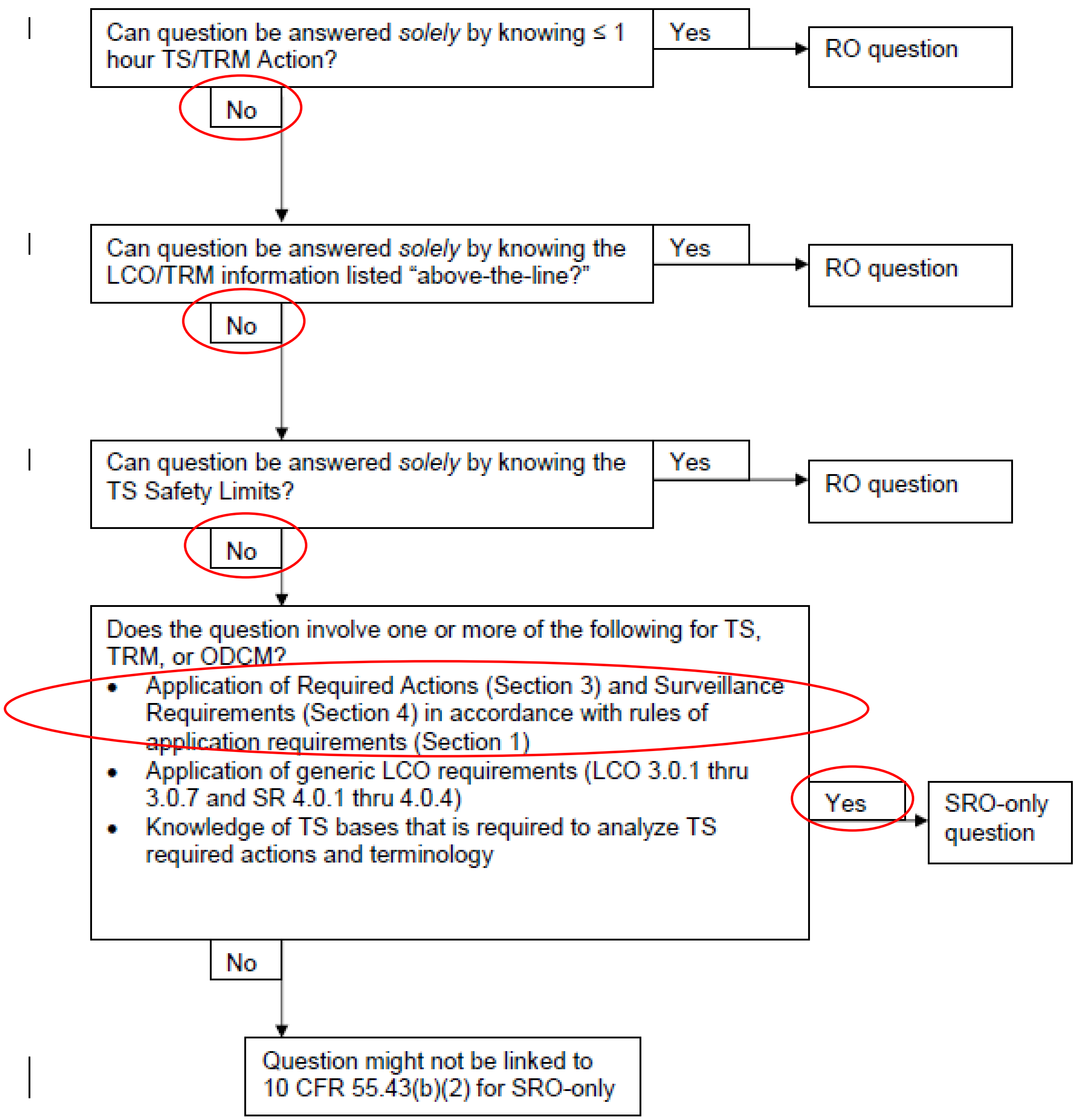
OR

the other MFW Pump is in service

THEN

Enter a 48 Hour T.S. CLOCK IAW T.S. Table 3.7-2.
Otherwise, enter N/A.

Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)



K/A Number: 061G2.4.47, Auxiliary/Emergency Feedwater, Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material., (CFR: 41.10 / 43.5 / 45.12)

Level: SRO Tier #: 2 Group #: 1 IR – RO: 3.4 IR-SRO: 3.7

Proposed Question:

Initial Conditions:

- Unit 1 was at 100% when a Seismic event occurred.
- Security reports a large volume of steam coming from U1 Safeguards louvers.
- A Loss of offsite Power has occurred, with a failure of EDG-3 to start.
- Crew has been unable to isolate steam flow from the Steam Generators.
- Efforts are in progress to close Main Steam Trip Valve/Non-Return isolation valve.

Current Conditions (10 minutes later):

- RCS Pressure 1200 psig and lowering.
- RCS Tave 510 °F and lowering in all 3 loops.
- SG Pressure: S/G“A” - 480 psig/ lowering, S/G“B” - 541 psig/ rising, S/G“C” - 490 psig/ lowering.
- AFW Flow: S/G“A” – 170 gpm, S/G“B” – 120 gpm, S/G“C” – 160 gpm.
- SG Levels are: S/G “A” – 44% WR/lowering, S/G“B” – 38% WR/rising, S/G“C” – 43% WR/lowering.
- Containment Pressure is 10.5 psia and steady.
- Crew is just starting step 2 in ECA 2.1, Uncontrolled Depressurization of all S/Gs; “Control FF to minimize RCS cooldown.”

Given the above information determine:

1) The correct procedural sequence. AND
2) Feed Flow directions the SRO is required to give.

A. 1) Continue in ECA-2.1, transition to FR-H.1 (Loss of Sec Heat Sink), then back to ECA-2.1.
2) Maintain a minimum of 60 gpm to each S/G.

B. 1) Continue in ECA-2.1, transition to FR-H.1 (Loss of Sec Heat Sink), then back to ECA-2.1.
2) Isolate FW to S/G A and S/G C, maintain minimum of 350 gpm to S/G B.

C. 1) Go to E-2 (Faulted S/G Isolation), then transition to E-1 (Loss of Rx. or Sec. Coolant), and then to ES-1.1 (Si Term.).
2) Maintain a minimum of 60 gpm to each S/G.

D. 1) Go to E-2 (Faulted S/G Isolation), then transition to E-1 (Loss of Rx. or Sec. Coolant), and then to ES-1.1 (Si Term.).
2) Isolate FW to S/G A and S/G C, maintain minimum of 350 gpm to S/G B.

Proposed Answer: D. 1) Go to E-2, then transition to E-1, and then to ES-1.1.
2) Isolate FW to S/G A and S/G C, maintain minimum of 350 gpm to S/G B.

Explanation: With conditions given, S/G B has been isolated. E-2 Transition Criteria (Continuous action no. 3) in ECA-2.1 states the following “Go to E-2, Faulted Steam Generator Isolation, if any S/G Pressure increases at any time except while performing SI Termination in Steps 13 to 23.” The SRO should decide to go to E-2 immediately because there is now an intact S/G. After performing E-2 to isolate S/G A and S/G C the crew will transition to E-1 to stabilize the plant and then ES-1.1 to reduce Safety Injection Flow. The SRO should decide to feed S/G B at best available rate to fully establish S/G B as a heat sink.

Technical Reference: 1-ECA-2.1, 1-E-2.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-2, ECA-2.1, Objective B, Given a copy of ECA-2.1, Uncontrolled Depressurization of All Steam Generators, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12) 10 CFR 55.43 (b)(5)

Comments:

Distractor Analysis:

A. Incorrect – E-2 Transition criteria is met therefore E-2 is the correct procedure to go to. Plausible because if the SRO doesn’t perform E-2 the procedure sequence will be as written with the sequence in and out of FR-H.1 after throttling to 60 gpm. 60 gpm is the correct flow rate based on cooldown rate from conditions given (approx. 40 ^{o}F/10min equates to > 200 ^{o}F/hr.).
B. Incorrect – E-2 Transition criteria is met therefore E-2 is the correct procedure to go to. Also Feed Flow rate incorrect if staying in ECA-2.1. Plausible because after step 2 is performed total Aux Feed will be < 350 gpm which would satisfy FR.H.1 criteria.
C. Incorrect – E-2 Transition criteria is correct. The flow rate is out of step 2 of ECA-2.1 which is incorrect. Plausible if operator confuses ECA-2.1 flow requirements with E-2 flow requirements.
D. Correct – The SRO should decide to go to E-2 immediately since E-2 transition criteria is met. After performing E-2 to isolate S/G A and S/G C the crew will transition to E-1 to stabilize the plant and then ES-1.1 to reduce Safety Injection Flow. Feed Flow should be maintained at best available rate to S/G B.

Attachment 1

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: A minimum of 60 gpm [100 gpm] feed flow must be maintained to each SG with a narrow range level less than 12% [18%].

* *

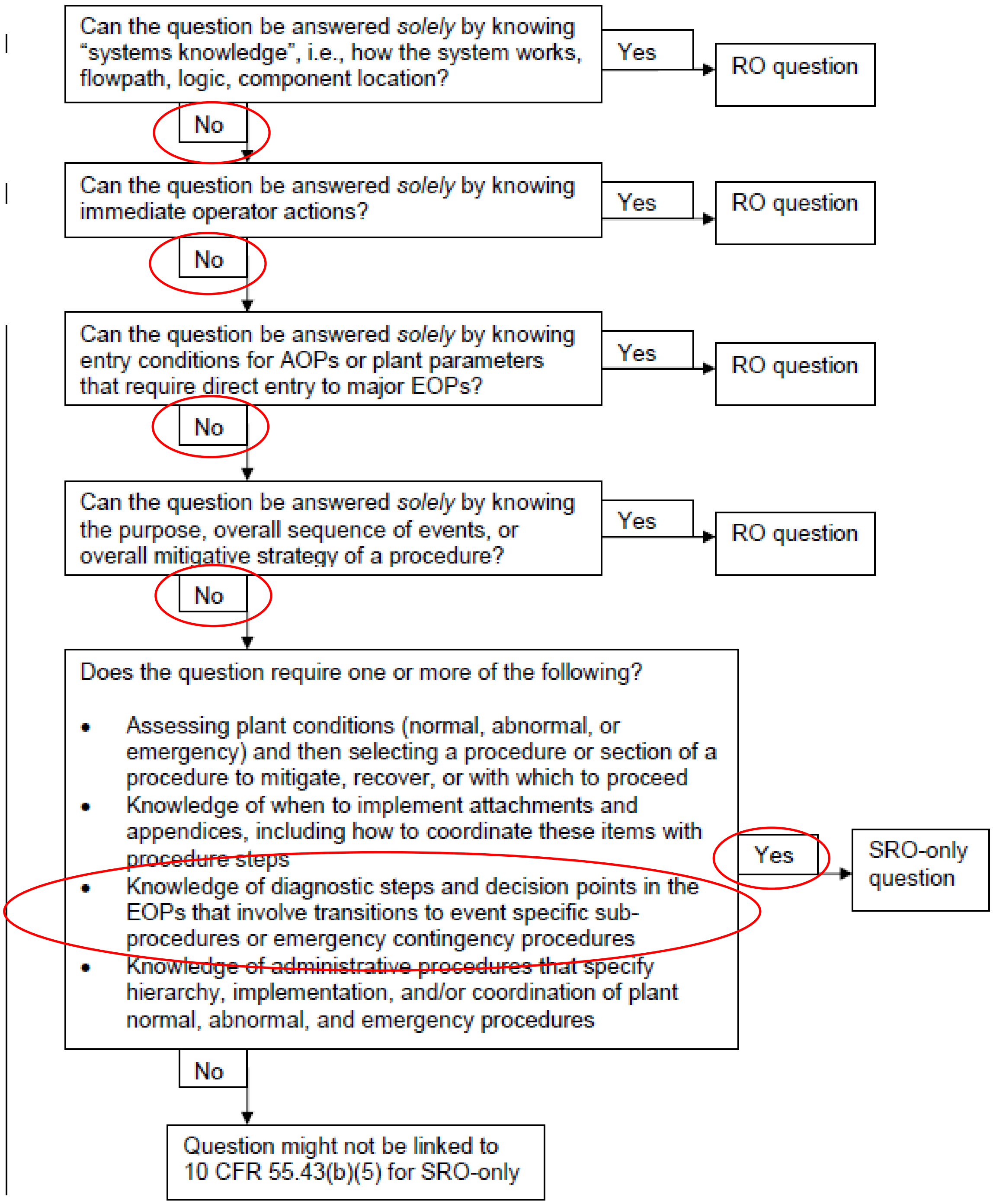
NOTE: Shutdown Margin should be monitored during RCS cooldown.

2. ___ CONTROL FEED FLOW TO MINIMIZE RCS COOLDOWN:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a) Check cooldown rate in RCS cold legs - LESS THAN 100°F/hr	☐ a) Lower feed flow to 60 gpm [100 gpm] to each SG. GO TO Step 2c.
☐ b) Check narrow range level in all SGs - LESS THAN 50%	☐ b) Control feed flow to maintain narrow range level less than 50% in all SGs.
☐ c) Check RCS hot leg temperatures - STABLE OR DECREASING	☐ c) Control feed flow or dump steam to stabilize RCS hot leg temperatures.

NOTE: Seal injection flow should be maintained to all RCPs.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



References for correct answer: ECA-2.1 Continuous Action directs going to E-2.

CONTINUOUS ACTIONS PAGE FOR 1-ECA-2.1

1. SI REINITIATION CRITERIA

Following SI termination or SI flow reduction, manually start SI pumps as necessary if EITHER condition listed below occurs:

- RCS subcooling based on CETCs - LESS THAN 30°F [85°F]
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 22% [50%]

2. ADVERSE CONTAINMENT CRITERIA

Use Adverse Containment setpoints if EITHER condition listed below occurs:

- Containment Pressure - GREATER THAN 20 PSIA
- Containment Radiation - GREATER THAN 1.0E5 R/HR

3. E-2 TRANSITION CRITERIA

GO TO 1-E-2, FAULTED STEAM GENERATOR ISOLATION, if any SG pressure increases at any time, except while performing SI Termination in Steps 13 to 23.

References for distractor A and B: If SRO decides not to transition to E-2 then ECA-2.1 step 2 is next step.

* *

CAUTION: A minimum of 60 gpm [100 gpm] feed flow must be maintained to each SG with a narrow range level less than 12% [18%].

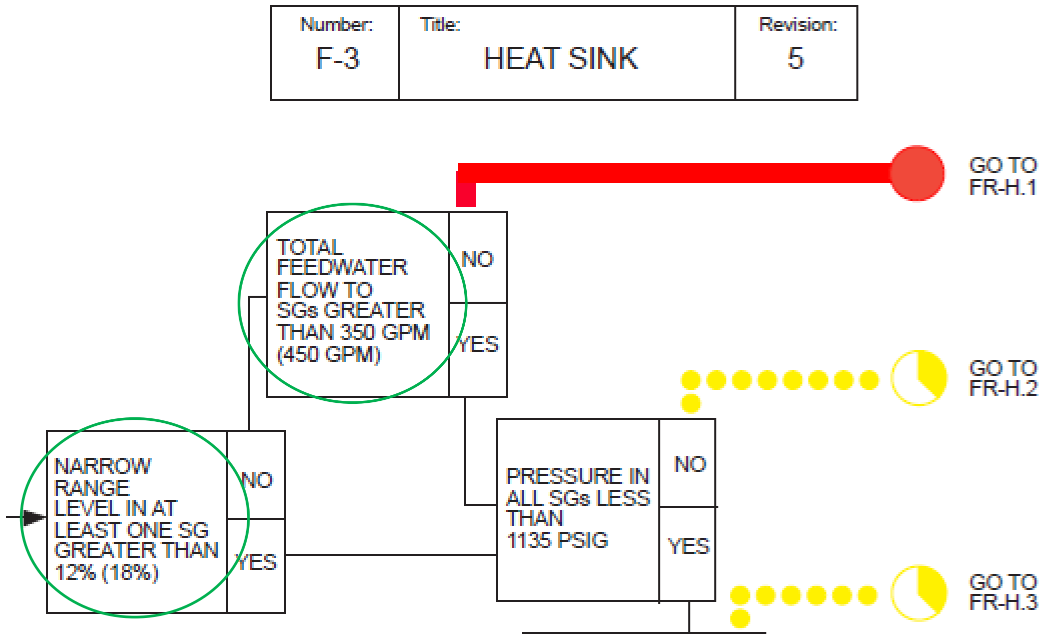
* *

NOTE: Shutdown Margin should be monitored during RCS cooldown.

2. ___ CONTROL FEED FLOW TO MINIMIZE RCS COOLDOWN:

☐ a) Check cooldown rate in RCS cold legs - LESS THAN 100°F/hr	☐ a) Lower feed flow to 60 gpm [100 gpm] to each SG. GO TO Step 2c.

After ECA-2.1 step 2 performed then FR-H.1 will be met. (No SG > 12% and Feed Flow <350 gpm)



References for Distractor C: If SRO throttles flow to 60 gpm before transitioning to E-2. Procedure usage dictates going to E-2 first.

* *

CAUTION: A minimum of 60 gpm [100 gpm] feed flow must be maintained to each SG with a narrow range level less than 12% [18%].

* *

NOTE: Shutdown Margin should be monitored during RCS cooldown.

2.___ CONTROL FEED FLOW TO MINIMIZE RCS COOLDOWN:

☐ a) Check cooldown rate in RCS cold legs - LESS THAN 100°F/hr	☐ a) Lower feed flow to 60 gpm [100 gpm] to each SG. GO TO Step 2c.

K/A Number: 075G2.2.44, Circulating Water / 8, Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions, (CFR: 41.5 / 43.5 / 45.12)

Level: SRO Tier #: 2 Group #: 2 IR – RO: 4.2 IR-SRO: 4.4

Proposed Question:

Initial Conditions:

- Unit 1 and 2 are operating at 100% power.
- A Loss of Offsite Power causes an Automatic Reactor Trip on Both Units.
- #3 EDG fails to start.

Current Conditions:

- Intake Canal Level is decreasing.
- Crew is performing immediate actions of E-0, Reactor Trip and SI.

Which ONE of the following identifies

1) The appropriate procedure flowpath to regain control of Intake Canal level.
2) The minimum intake canal level required per TS 3.14 (Circulating and SW) Bases to provide adequate design flow through RS SW heat exchangers.

A. 1) E-0, 1(2)-AP-10.07, Loss of Unit 1(2) Power, 0-AP-12.01, Loss of Intake Canal Level.
2) 17.2 feet.

B. 1) E-0, 0-AP-12.01, Loss of Intake Canal Level, 1(2)-AP-10.07, Loss of Unit 1(2) Power.
2) 23.5 feet.

C. 1) E-0, 1(2)-AP-10.07, Loss of Unit 1(2) Power. 0-AP-12.01, Loss of Intake Canal Level.
2) 23.5 feet.

D. 1) E-0, 0-AP-12.01, Loss of Intake Canal Level, 1(2)-AP-10.07, Loss of Unit 1(2) Power.
2) 17.2 feet.

Proposed Answer: A

Explanation: As described in Stem, AP-10.07, Loss of Unit 1(2) Power is initiated from Step 3 of E-0, RNO. AP-10.07 directs response to the loss of power event in order of priority to control the plant and restore power. AP-12.01, Loss of Intake Canal level would be initiated FROM AP-10.07. Candidate may assess “Intake Canal Level decreasing rapidly” as requiring a higher level priority than responding to the loss of power. Canal Level minimum level described in TS 3.14, Circulating and Service Water Systems Basis.

Technical Reference: 1(2)-AP-10.07, Loss of Unit 1(2) power.

Reference Provided to Applicant: No

Learning Objective: ND-90.3-LP-7, SS and Emer Dist and Control, Objective G, Given copies of AP-10.07, Loss of Power, and AP-10.08 Station Power Restoration, explain the steps necessary to operate the station following a total or partial loss of power.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

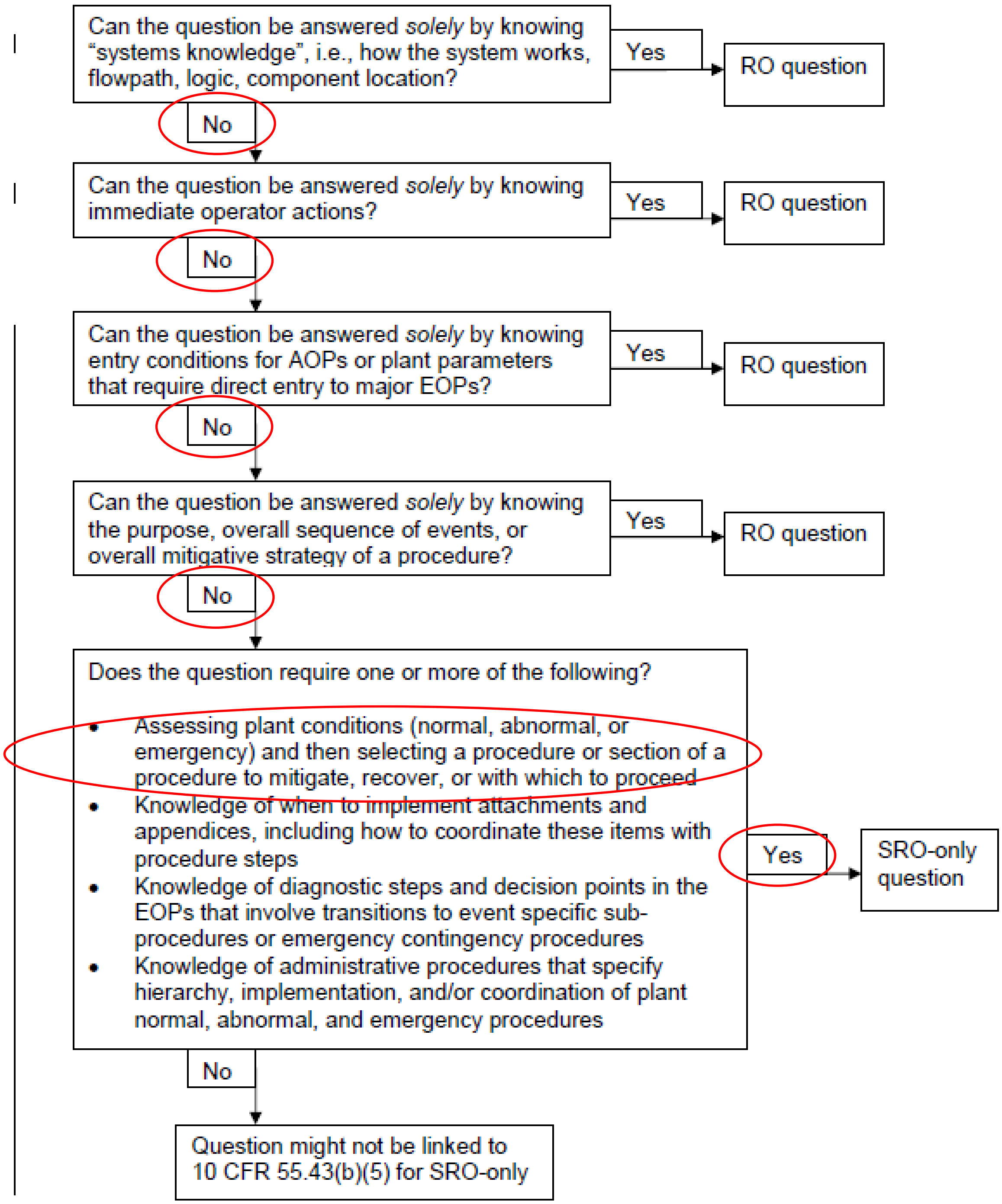
10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.12)

Comments:

Distractor Analysis:

A. Correct – AP-10.07 would have the highest priority in responding to the event, with AP-12.01 initiated from this procedure. 17.2 feet is derived from TS-3.14 basis for providing adequate flow through RSHXs during a Design Basis Event.
B. Incorrect – Reverse sequence of priority in Abnormal Procedure implementation. 23.5 feet derives from same paragraph of TS-3.14 Basis.
C. Incorrect – Sequence of procedure usage (correct). Incorrect Canal level value from TS-3.14 basis.
D. Incorrect – Reverse Sequence of procedure usage. Canal level value is correct.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



TS 3.14-4
08-30-01

including replacement of an Emergency Service Water pump without forcing dual unit outages, yet limits the amount of operating time without the specified number of pumps.

When one Unit is in Cold Shutdown and the heat load from the shutdown unit and spent fuel pool drops to less than 25 million BTU/HR, then one Emergency Service Water pump may be removed from service for the subsequent time that the unit remains in Cold Shutdown due to the reduced residual heat removal and hence component cooling requirements.

A minimum level of +17.2 feet in the High Level Intake canal is required to provide design flow of Service Water through the Recirculation Spray heat exchangers during a loss-of-coolant accident for the first 24 hours. If the water level falls below +23' 6", signals are generated to trip both unit's turbines and to close the nonessential Circulating and Service Water valves. A High Level Intake canal level of +23' 6" ensures actuation prior to canal level falling to elevation +23'. The Circulating Water and Service Water isolation valves which are required to close to conserve Intake Canal inventory are periodically verified to limit total leakage flow out of the Intake Canal. In addition, passive vacuum breakers are installed on the Circulating Water pump discharge lines to assure that a reverse siphon is not continued for canal levels less than +23 feet when Circulating Water pumps are de-energized. The remaining six feet of canal level is provided coincident with ESW pump operation as the required source of Service Water for heat loads following the Design Basis Accident.

NUMBER	PROCEDURE TITLE	REVISION
1-E-0	REACTOR TRIP OR SAFETY INJECTION	68
		PAGE 3 of 15

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[3] ___	CHECK BOTH AC EMERGENCY BUSES - ENERGIZED	Do the following: ☐ a) IF no AC Emergency Bus is energized, THEN GO TO 1-ECA-0.0, LOSS OF ALL AC POWER. ☐ b) Try to restore power to deenergized AC Emergency Bus. Initiate 1-AP-10.07, LOSS OF UNIT 1 POWER.

NUMBER	PROCEDURE TITLE	REVISION
1-AP-10.07	LOSS OF UNIT 1 POWER	67
		PAGE 8 of 37

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: The TD AFW pump can NOT be secured until SG narrow range level is greater than 17% on 2/3 SGs, AMSAC is reset, and the UV interlock is bypassed.

* *

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*16. ___	CHECK TD AFW PUMP - CONTINUED OPERATION DESIRED	Do the following: ☐ a) Check or increase all SG narrow range levels to greater than 21%. ☐ b) Put the TD AFW Pump U/V auto Function Bypass switches in BYPASS. c) Stop the TD AFW Pump by placing the control switches in OPEN-RESET and then in CLOSE. ☐ • 1-MS-SOV-102A ☐ • 1-MS-SOV-102B
17. ___	CHECK INTAKE CANAL LEVEL - STABLE OR INCREASING	☐ Initiate 0-AP-12.01, LOSS OF INTAKE CANAL LEVEL.

K/A Number: 078A2.01, Instrument Air /, Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:, (CFR: 41.5 / 43.5 / 45.3 / 45.13) Air dryer and filter malfunctions

Level: SRO Tier #: 2 Group #: 1 IR – RO: 2.4 IR-SRO: 2.9

Proposed Question:

Initial Conditions (Time 0200):

Unit 1 is at cold Shutdown, with RCS temperature at 195 °F.

- RCS is in solid plant pressure control.
- The "A" RHR pump and "A" RHR Heat Exchanger are in service.
- 1-RC-P-1C, is running.

Time 0205:

- Annunciator 1B-G5, INST AIR DRYER TRBL is received.
- Station Instrument Air pressure is 75 psig and lowering rapidly.

TIME 0210:

- Local report of both Instrument Air dryer towers blowing down.
- RCS temperature is 205 °F and rising.
- Operators are currently working on Instrument Air restoration.

Which ONE of the following identifies:

1) The impact of the loss of Instrument Air to the RHR system.
2) The EAL used to classify this event.

(REFERENCE PROVIDED)

A. 1) 1-RH-HCV-1758, RHR HXS Flow Fails Close.
 2) NOUE CU3.1.

B. 1) 1-RH-HCV-1758, RHR HXS Flow Fails Close.
 2) Alert CA3.1.

C. 1) 1-CC-TV-109A/B, RHR H/X Trip Valves Fail Close.
 2) NOUE CU3.1.

D. 1) 1-CC-TV-109A/B, RHR H/X Trip Valves Fail Close.
 2) Alert CA3.1.

Proposed Answer: C

Explanation: In this event a loss of decay heat removal would be caused by the loss of instrument air. The loss of Instrument Air will cause a loss of RHR cooling due to 1-CC-TV-109A/B failing close. CU3.1 classification is appropriate since a loss of decay heat removal has occurred. Upgrade to alert, CA 3.1, is required if the Unit temperature has exceeded 200 °F, and by Table C-3 – the duration exceeds 60 minutes. This allows time to respond to the event, and restore the capability to remove decay heat.

Technical Reference: 0-AP-40.00, Loss of Station Instrument Air.

Reference Provided to Applicant: No

Learning Objective: ND-95.1-LP-9, Loss of Instrument Air, Objective A, Describe the operator actions associated with the annunciator responses procedure for Service Air, Instrument Air, and Containment IA Alarms. And, Objective B, Given a copy of AP-40.00, Non-Recoverable Loss of Instrument Air, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13) 10 CFR 55.43(b)(5)

Comments:

Distractor Analysis

A. Incorrect – 1) 1-RH-HCV-1758 does NOT fail closed. This is plausible if Operator confuses fail position with 1-RH-FCV-1605, RHR HXS Bypass Flow which does fail close. 2) Per Surry EAL Table, Classification of CU 3.1 is correct. CA 3.1, would not apply unless the event duration would be expected to exceed 60 minutes.
B. Incorrect - 1)) 1-RH-HCV-1758 does NOT fail closed. This is plausible if Operator confuses fail position with 1-RH-FCV-1605, RHR HXS Bypass Flow which does fail close. 2) Per Surry EAL Table, Classification of CU 3.1 is correct. CA 3.1, would not apply unless the event duration would be expected to exceed 60 minutes.
C. Correct – 1) 1-CC-TV-109A/B does fail close. This will cause loss of RHR cooling and subsequent RCS temperature rise. 2) Per Surry EAL Table, Classification of CU 3.1 is correct
D. Incorrect – 1) 1-CC-TV-109A/B does fail close. This will cause loss of RHR cooling and subsequent RCS temperature rise. 2) CA 3.1, would not apply unless the event duration would be expected to exceed 60 minutes.

References from 0-AP-40.0, Attachment 1 that show fail positions for 1-CC-TV-109A/B, 1-RH-HCV-1758, and 1-RH-FCV-1605.

NUMBER 0-AP-40.00	ATTACHMENT TITLE LOSS OF AIR - SYSTEMS RESPONSE	ATTACHMENT 1
REVISION 30		PAGE 1 of 8

This attachment has been designed to give the operator the general overall status and capabilities of the various systems following a loss of air.

PRIMARY SYSTEMS

COMPONENT COOLING (( )-AP-15.00)

1. All Component Cooling flow from Containment secured:

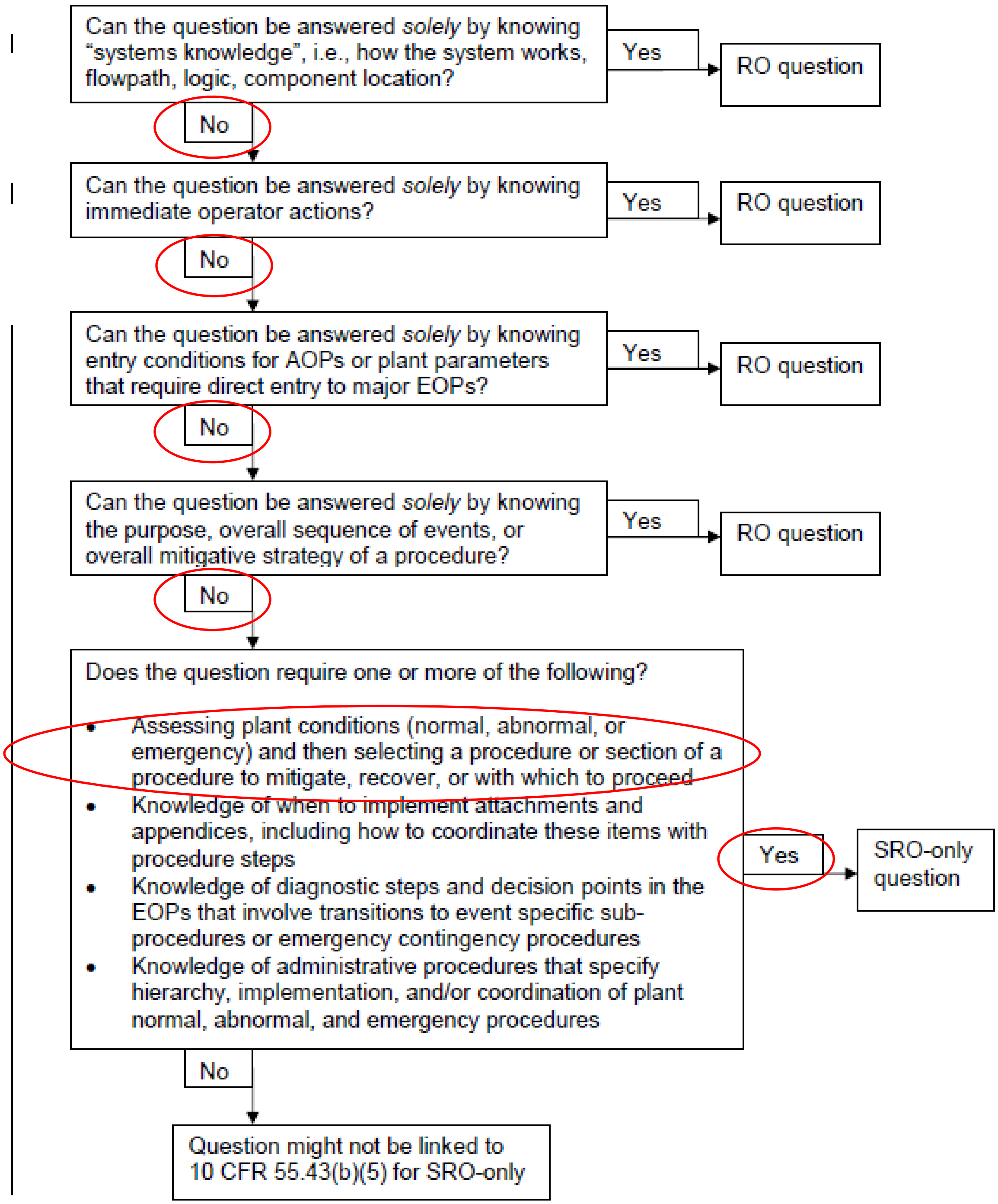
- ☐ a. RHR ( )-CC-TV-( )09A and B
- ☐ b. Excess Letdown ( )-CC-HCV-( )08
- ☐ c. RCP stator and lube oil cooling ( )-CC-TV-( )05A, B, C
- ☐ d. RCP thermal barrier cooling ( )-CC-TV-( )40A and ( )40B. (These valves have air lockup valves installed and should not fail closed.)
- ☐ e. Air recirc coolers and Neutron Shield tank coolers ( )-CC-TV-( )10A, B, C

NUMBER 0-AP-40.00	ATTACHMENT TITLE LOSS OF AIR - SYSTEMS RESPONSE	ATTACHMENT 1
REVISION 30		PAGE 8 of 8

RESIDUAL HEAT REMOVAL

1. RHR LETDOWN FLOW ( )-RH-HCV-( )142 fails CLOSED.
2. RHR HXS FLOW ( )-RH-HCV-( )758 fails OPEN.
3. RHR HXS BYP FLOW ( )-RH-FCV-( )605 fails CLOSED.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: G2.1.25, Conduct of operations, Ability to interpret reference materials such as graphs, monographs and tables which contain performance data., (CFR: 41.10 / 43.5 / 45.12)

Level: SRO Tier #: 3 Group #: IR – RO: 3.9 IR-SRO: 4.2

Proposed Question:

Initial Conditions:

- Unit 1 is at Cold Shutdown.
- 1-OPT-SI-014, Cold Shutdown Test of SI Check Valves to RCS Cold Legs, is currently in progress.
- Previous Leak Rate data has been provided by Engineering.
- Current Leak Rate data has just been determined.
- SI Check Valve leakage results are as follows:

	Mark Number	Previous Test	Current Results
A	1-SI-79	3.50 gpm	4.30 gpm
B	1-SI-82	0.5 gpm	0.9 gpm
C	1-SI-85	1.25 gpm	3.0 gpm
D	1-SI-241	4.8 gpm	5.2 gpm
E	1-SI-242	0.9 gpm	0.8 gpm
F	1-SI-243	0.5 gpm	2.8 gpm

Which one of the following lists **ALL** the Check Valves that are Inoperable in accordance with Technical Specifications section 3.1.C.5.a, Primary Coolant System Pressure Isolation Valves?

A. 1-SI-241.

B. 1-SI-79, 1-SI-243.

C. 1-SI-79, 1-SI-241, 1-SI-243.

D. 1-SI-79, 1-SI-85, 1-SI-241, 1-SI-243.

Proposed Answer: C. 1-SI-79, 1-SI-241, 1-SI-243

Explanation: Question requires the SRO Candidate to interpret the Table and determine which Pressure Isolation valves are inoperable per TS 3.1.C.5.a. The Table used is similar to the Table that is used in 1-OPT-SI-014. Engineering provides the previous leakage data, and the SRO evaluates Current leakage against Previous leakage to determine operability. The Tech Spec Criteria is as follows:

- Leakage must be < 5.0 gpm.
- Leakage < 1.0 gpm is acceptable.
- Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered

acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater. Based on this a formula can be devised for acceptable leakage: (5.0 – Current leakage) / (5.0 – Previous leakage) > 0.5.

Based on this criteria the SI Check Valve results are as follows:

	Mark Number	Previous Test	Current Results	**Operable/Inoperable**
A	1-SI-79	3.50 gpm	4.30 gpm	**Inop: 0.7/1.5 = .46**
B	1-SI-82	0.5 gpm	0.9 gpm	**Operable: 4.1/4.5 =.91**
C	1-SI-85	1.25 gpm	3.0 gpm	**Operable: 2.0/3.75=.53**
D	1-SI-241	4.8 gpm	5.2 gpm	**Inop: > 5.0 gpm**
E	1-SI-242	0.9 gpm	0.8 gpm	**Operable: 4.2/4.1=1.02**
F	1-SI-243	0.5 gpm	2.8 gpm	**Inop: 2.2/4.5 = .48**

Technical Reference: Tech Specs. 3.1.C.5a

Reference Provided to Applicant: No

Learning Objective:

Question Source: Modified TS00078

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12) 10 CFR 55.43 (b)(5)

Comments:

Distractor Analysis:

A. 1-SI-241. Incorrect because 1-SI-79, and 1-SI-243 are also inoperable because of margin. This is plausible If SRO candidate only applies criteria for greater than 5.0 gpm. 1-SI-241 is inoperable because it is greater than 5.0 gpm but it is *not the only* inoperable check valve.
B. 1-SI-79, 1-SI-243. Incorrect because 1-SI-241 is also inoperable. This is plausible if SRO candidate only applies criteria for margin to 5.0 gpm criteria.
C. 1-SI-79, 1-SI-241, 1-SI-243 (correct answer)
D. 1-SI-79, 1-SI-85, 1-SI-241, 1-SI-243. Incorrect because of these 1-SI-85, is operable. This is Plausible if SRO confuses acceptance criteria and believes < 1.0 gpm is only acceptance criteria as all these are > 1.0 gpm.

Reference from Tech Specs 3.1.C.5.a

5.a. Prior to going critical all primary coolant system pressure isolation valves listed below shall be functional as a pressure isolation device, except as specified in 3.1.C.5.b. Valve leakage shall not exceed the amounts indicated.

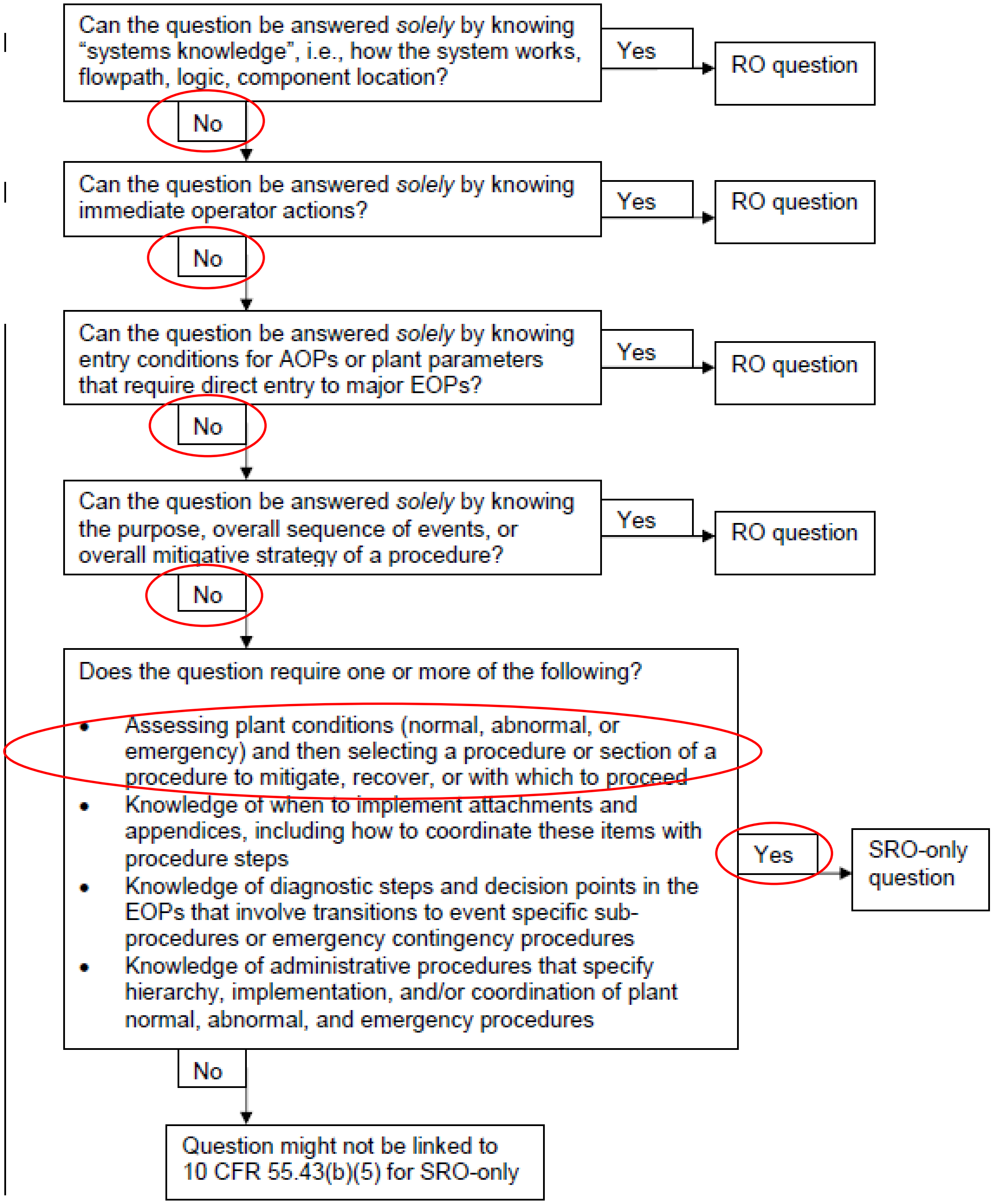
	Unit 1	Unit 2	Max. Allowable Leakage (see note (a) below)
Loop A, Cold Leg	1-SI-79, 1-SI-241	2-SI-79, 2-SI-241	≤ 5.0 gpm for each valve
Loop B, Cold Leg	1-SI-82, 1-SI-242	2-SI-82, 2-SI-242	
Loop C, Cold Leg	1-SI-85, 1-SI-243	2-SI-85, 2-SI-243	

b. If Specification 3.1.C.5.a cannot be met, an orderly shutdown shall be initiated and the reactor shall be in HOT SHUTDOWN within 6 hours and in COLD SHUTDOWN within the following 30 hours.

Notes

(a)
1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.
2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.
4. Leakage rates greater than 5.0 gpm are considered unacceptable.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: G2.1.32, Conduct of operations, Ability to explain and apply all system limits and precautions, (CFR: 41.10 / 43.2 / 45.12)

Level: SRO Tier #: 3 Group #: IR – RO: 3.8 IR-SRO: 4

Proposed Question:

Initial Conditions:

Unit 1 is operating at 100% power.

- Control Rod E-11, Shutdown Bank B, Group 1, has dropped and indicates zero (0) Steps.
- Indications and alarms received confirm that E-11 has dropped.
- The Team initiates 0-AP-1.00, Rod Control System Malfunction.
- The STA completes 0-AP-1.00 Attachment 6, Calculation of Excore Quadrant Power Tilt Ratios, and reports that the Upper channel tilt is 16%, and the Lower channel tilt is 13.5%.

Which ONE of the following states:

1) The reactor power level and time requirement in effect.
2) The most limiting basis for the power level requirement.

A. 1) Reduce Reactor Power to < 75% Rated Power in one (1) hour.
   2) To maintain the minimum DNBR greater than design limits.

B. 1) Reduce Reactor Power to < 75% Rated Power in thirty (30) minutes.
   2) To prevent core power distribution from exceeding design limits.

C. 1) Reduce Reactor Power to < 68% Rated Power in one (1) hour
   2) To maintain the minimum DNBR greater than design limits.

D. 1) Reduce Reactor Power to < 68% Rated Power in thirty (30) minutes.
   2) To prevent core power distribution from exceeding design limits.

Proposed Answer: D

Explanation: TS 3.12.B.6 describes the limitations on Quadrant Power Tilt to prevent exceeding core power distribution limits. With a calculated QPTR of >10%, Reactor Power must be reduced 2% for each percent of tilt within 30 minutes. With a dropped rod, the rod is mis-aligned from its bank > ±12 Steps, thus it is considered INOPERABLE. With an Inoperable rod, Reactor Power must be reduced to <75% within one hour. Maintenance of DNBR, per TS 3.12 Basis, relates to maintaining DNBR parameters within limits to ensure the Minimum DNBR > design limits. Candidate must assess the dropped rod condition and the QPTR calculation to correctly apply Tech Specs to this event. The location of the dropped rod causes a 20% reduction in N42 indication due to rod shadowing, resulting in the significant impact on calculated QPTR.

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Technical Reference: Tech Spec 3.12.B.6, a. through c. TS 3.12.C. TS 3.12 Basis.

Reference Provided to Applicant: NO

Learning Objective:

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.2 / 45.12)

Comments:

Distractor Analysis:

A. Incorrect –1) Per TS 3.12 The calculated QPTR has a greater effect on the amount of power reduction from rated power and a greater urgency to effect these changes than a dropped rod which is the basis for power reduction to <75%. Time allowed for the reduction for this case is 30 minutes, vice 1 hour. 2) The reason given in the distractor relates to control of DNBR parameters which the Candidate could confuse with core power distribution concerns.
B. Incorrect –1) The magnitude of the power reduction correlates to an inoperable rod, not to the calculated QPTR; the time for the reduction is correct. 2) The basis for the power reduction is correct.
C. Incorrect –1) the magnitude of the power reduction is correct, the allowed time is incorrect. 2) The reason given in the distractor relates to control of DNBR parameters which the Candidate could confuse with core power distribution concerns.
D. Correct–1) Magnitude of power reduction and time limit are correct. 2) The reason for the power reduction is also correct.

Choice 'D' (Correct Answer)

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06-25-09

5. The allowable QUADRANT POWER TILT is 2.0% and is only applicable while operating at THERMAL POWER > 50%.

6. If, except for operation at THERMAL POWER < 50% or for physics and control rod assembly surveillance testing, the QUADRANT POWER TILT exceeds 2%, then:

   a. Within 2 hours, either the hot channel factors shall be determined and the power level adjusted to meet the requirement of Specification 3.12.B.1, or

   b. The power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

   c. If the QUADRANT POWER TILT exceeds 10%, the power level shall be reduced from RATED POWER 2% for each percent of QUADRANT POWER TILT within the next 30 minutes. The high neutron flux trip setpoint shall be similarly reduced within the following 4 hours.

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10-19-10

A 2% QUADRANT POWER TILT allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod assembly and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rod assemblies producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The QPTR limit must be maintained during power operation with THERMAL POWER > 50% of RATED POWER to prevent core power distributions from exceeding the design limits.

Distractor Reference

3. Startup and POWER OPERATION may continue with one control rod assembly inoperable provided that within one hour either:

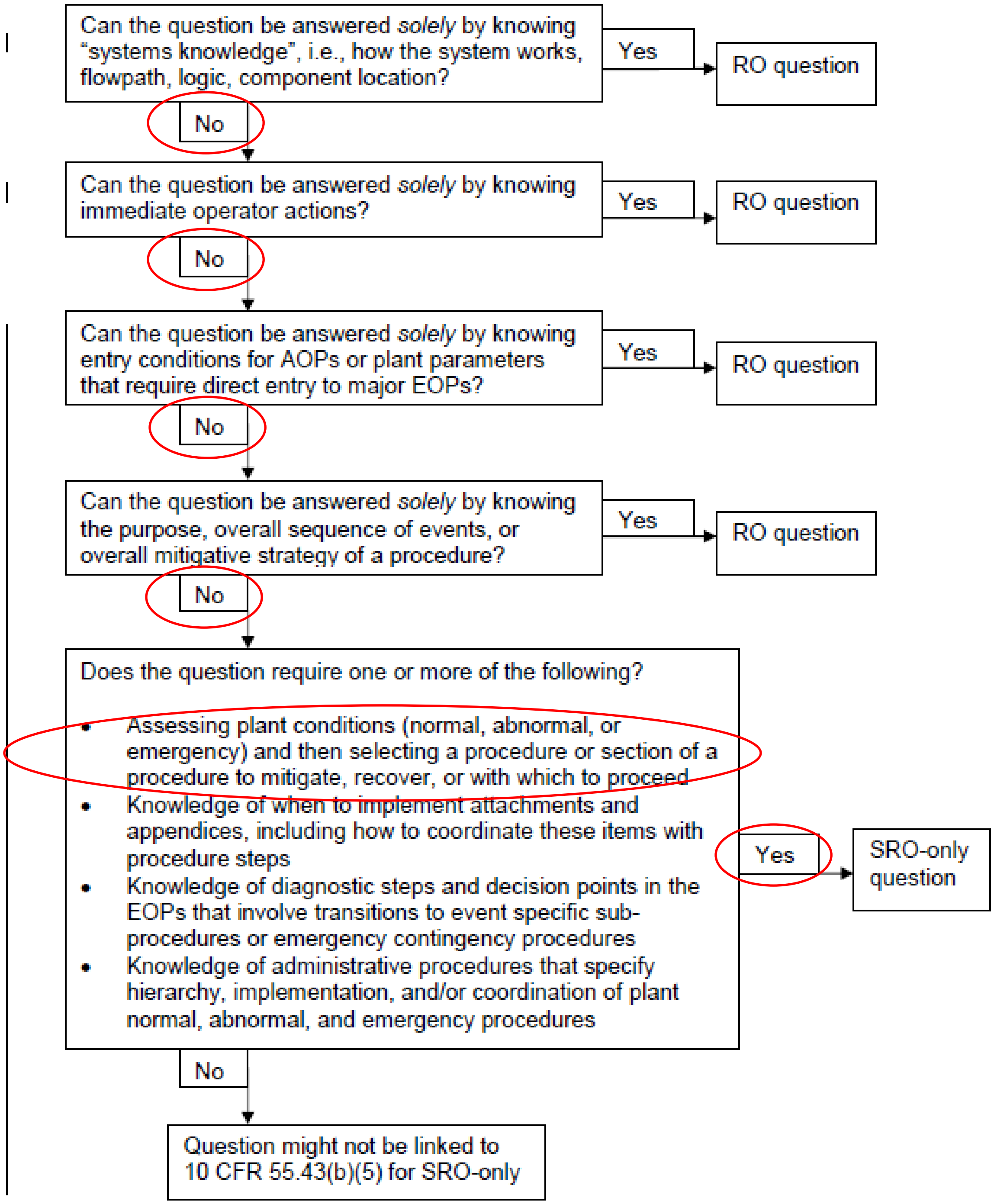
   a. The control rod assembly is restored to OPERABLE status, as defined in Specification 3.12.C.1 and 2, or

   b. the shutdown margin requirement of Specification 3.12.A.3.c is satisfied. POWER OPERATION may then continue provided that:

      1) either:

         (a) power shall be reduced to less than 75% of RATED POWER within one (1) hour, and the High Neutron Flux trip setpoint shall be reduced to less than or equal to 85% of RATED POWER within the next four (4) hours, or

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: G2.2.13, Equipment Control, Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)

Level: SRO Tier #: 3 Group #: IR – RO: 4.1 IR-SRO: 4.3

Proposed Question:

Which ONE of the following examples represents an INVALID waiver of Tag removal verification by Shift Supervision?

A. Tags for 1-SI-MOV-1890A, 1-SI-MOV-1890B (LHSI to Hot Legs) and 1-SI-MOV-1890C (LHSI to Cold Legs) during Team response to a LOCA outside of Containment in accordance with 1-ECA-1.2, LOCA outside Containment.

B. Tags for 1-SI-MOV-1869A and 1-SI-MOV-1869B (HHSI to Hot Legs) when cold leg SI flow cannot be established during a SBLOCA with a loss of subcooling on Unit 1.

C. Tags in an area with a Dose Rate of 1 R/hr.

D. Tags from a Transformer disconnect being worked by substation personnel during Switchyard restoration following a tornado because of a shortage of personnel.

Proposed Answer: D

Explanation: OP-AA-200, Equipment Clearance allows Shift Supervision (i.e., SRO assigned to the Shift or the Shift Manager) to waive verification of Tag removal for Plant Emergencies, High Radiation Areas, or Personnel Safety. Only Operations personnel are authorized to place or remove Tags.

Technical Reference: OP-AA-200, Equipment Clearance, Page 5, Step 3.1.12.

Reference Provided to Applicant: No

Learning Objective:

Question Source: Modified Bank

Question History: Last NRC Exam: NO

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: (CFR: 41.10 / 45.13)

Comments:

Distractor Analysis:

A) Incorrect – Shift Supervision may waive verification of Tag Removal during Plant Emergencies.
B) Incorrect - Shift Supervision may waive verification of Tag Removal during Plant Emergencies.
C) Incorrect - Shift Supervision may waive verification of Tag Removal due to high radiation areas.
D) Correct – The System Operator is required to approve the hanging or clearing of tags in the switchyard.

Link Question to 10 CFR 55.43 (b)(3). Tag removal and verification is a Plant Configuration control item; by guidance in reference procedure, only Shift Supervision may waive verification of Tag removal – thus question examines SRO level of knowledge of Equipment Clearance procedure.

DOMINION **OP-AA-200**
REVISION 19
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Operations/ Tagging Office

3.1.8 **ENSURE** a work order does <u>**NOT**</u> rely on two distinct different Tagouts for the isolation boundaries.

3.1.9 <u>**IF**</u> work orders associated with a DCP/plant modifications require testing, <u>**THEN**</u> a Test Engineer must **SIGN ON** the Tagout prior to beginning any testing and will **SIGN OFF** the Tagout when the equipment is ready to be released to Operations.

3.1.10 **REPLACE** tags in the field that are worn or illegible. The new tag shall have the same information and be attached and verified in accordance with the Tagging Record. **REMOVE** the old tag and **DISPOSE** of the old tag after the new tag is installed.

Individual Placing or Removing Tags

3.1.11 **POSSESS** a copy of the governing document when in the field. This document may be the Tagging Hang Sheet, Tag Clear Sheet, or their electronic equivalent.

Operations Supervision

3.1.12 In general, all Tagouts will be verified **HUNG** and **REMOVED** except when verification is waived by an SRO for one of the follow reasons:

- Plant Emergencies
- High Radiation Areas
- Personnel Safety

DOCUMENT waiver in Notes section along with reason.

K/A Number: G2.2.18, Equipment Control, Knowledge of the process for managing maintenance activities during shutdown operations. (CFR: 41.10 / 43.5 / 45.13)

Level: SRO Tier #: 3 Group #: IR – RO: 2.6 IR-SRO: 3.8

Proposed Question:

Unit 1 is in Cold Shutdown and in the process of starting up the plant following a Refueling outage with the following conditions:

- The RCS is solid
- Pressurizer pressure is being maintained 85-125 psig
- RCS temperature band is 90-98 ^{o}F

You have just been notified that Post Maintenance Tests (PMTs) 1-IPT-FT-RC-P-403/458, “Reactor Coolant System Pressure Functional Test”, have not been completed for Pressurizer PORVs; 1-RC-PCV-1455C, and 1-RC-PCV-1456. The PMTs should have been completed 16 hours earlier.

What is the Operability status of the Pressurizer PORVs?

A. PORV Operability is not required for the present plant conditions given.

B. Declare both PORVs inoperable from the time the PMT should have been completed.

C. Declare both PORVs inoperable from the time of discovery.

D. Delay declaring both PORVs inoperable for up to 24 hours to allow the PMTs to be completed.

Proposed Answer: C. Declare both PORVs inoperable from the time of discovery

Explanation: OP-AA-102, Operability Determination; 3.2.2 states that an immediate determination of operability is to be made without delay in a controlled manner. Note below that step states that there is no regulatory guidance defining a specific time frame to complete an immediate determination of operability. General industry practice is within several hours.

Technical Reference: OP-AA-102, Operability Determination. Surry Technical Specifications

Reference Provided to Applicant: NO

Learning Objective: SRO-UTP, For a Tier 2 procedure discuss the requirements for Operations personnel.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)

Comments:

A. PORV Operability is not required for the present plant conditions given. This is incorrect because Tech Specs 3.1.G requires that both PORVs shall be Operable when RCS temperature is < 350 ^{o}F and the vessel head is bolted.
B. Declare both PORVs inoperable from the time the PMT should have been completed. This is incorrect because OP-AA-102 states that the impact on Tech Spec equipment should be made from the time of discovery. This is plausible because part of returning a component to service is to perform PMT and return component to service prior to declaring component operable (Ref: WM-AA-100, Maintenance process, step 3.13.4). This should have been done 16 hours earlier before therefore this is a plausible distractor.
C. Declare both PORVs inoperable from the time of discovery. OP-AA-102, step 3.2.1 states that the Shift Manager should determine the impact on TS operability upon discovery.
D. Delay declaring both PORVs inoperable for up to 24 hours to allow the PMTs to be completed. This is incorrect because there is no requirement to allow a 24-hour delay on declaring a component inoperable following a missed PMT. There is an allowance to allow a 24-hour delay for a missed Surveillance test which is entirely different (Ref: OP-AA-102, 3.6.1). This makes this a plausible distractor.

Reference for correct answer, OP-AA-102, 3.2.1.

3.2 Operability Determination

NOTE: Conservative decision making shall apply at all times when making a determination of operability and establishing administrative control of station components (Reference 5.4.5)

NOTE: A degraded or non-conforming SSC that is <u>NOT</u> required to be operable in the current mode of operation does <u>NOT</u> require documentation of operability. However, the degraded or non-conforming condition of the SSC should be designated as a potential mode restraint, etc. and tracked through the applicable process to resolution.

Shift Manager

3.2.1 Upon discovery of a degraded or nonconforming condition, or discovery of an unanalyzed condition, **DETERMINE** the impact on TS SSC operability. *(IMC 4.5)*

a. Operability is separate from corrective action to restore full qualification. The OD process is focused on safe plant operation and should **<u>NOT</u>** be impacted by decisions or actions necessary to plan and implement corrective action (i.e., restore full qualification). *(IMC 6.3)*

Reference for Distractor A from TS3.1-23.

c. Whenever the RCS average temperature is less than or equal to 350°F and the reactor vessel head is bolted:

(1) A maximum of one charging pump shall be OPERABLE and capable of injecting into the RCS. Two charging pumps may be in operation momentarily during transfer of operation from one charging pump to another.

and

(2) The accumulators shall be isolated (accumulator discharge valves closed and their respective breakers locked, sealed or otherwise secured in the open position). Isolation is not required if the accumulator pressure is less than the pressurizer PORV setpoint specified in TS 3.1.G.1.c.(4).

and

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of $\leq$ 390 psig and verify each PORV block valve is open at least once per 72 hours,

Reference for distractor B from WM-AA-100 3.13.4

3.13 Post Maintenance Activities, Return to Service, and Closeout

FLS (or designee)

3.13.1 **AUTHORIZE** tag release by signing off tag-out in eSOMS.

3.13.2 **EVALUATE** activity in accordance with WM-AA-301, Operational Risk Assessment.

3.13.3 **UPDATE** Work Order status in WMS as required by performing the following:

a. **IF** Engineering Review is required **AND NOT** completed in WMS, **THEN PERFORM** the following:

1. **SIGN** Engineering Review in WMS.
2. **INITIATE** Operation Acceptance Checklist (OAC) or PMT in accordance with CM-AA-DDC-201, Design Changes.

Engineering

3. **PERFORM** Engineering Operation Acceptance activities as required per CM-AA-DDC-201.
4. **UPDATE** Work Order Status in WMS as required.
5. **FORWARD** package to Operations for post maintenance activities (as applicable) and return-to-service.

FLS (or designee)

b. **IF** Engineering Review is **NOT** required or waived in WMS, **THEN PERFORM** the following:

1. **UPDATE** Work Order status in WMS as required.
2. **FORWARD** package to Operations for post maintenance activities (as applicable) and return-to-service.

Operations

3.13.4 **PERFORM** post maintenance testing activities (as applicable), and **RETURN** component/system to service as required.

Maintenance and

3.13.5 **SUPPORT** post-maintenance testing activities as required.

Reference for Distractor D.

3.6 Missed Surveillance

Shift Manager

3.6.1 <u>IF</u> it is discovered that a surveillance was <u>NOT</u> performed within its specified frequency and a reasonable expectation of operability exists, <u>THEN</u> IMMEDIATELY **DECLARE** the TS SSC operable. Declaring the SSC inoperable (i.e., the LCO <u>NOT</u> met) may be delayed, from the time of discovery, up to 24 hours (MPS, SPS, NAPS, KPS) or the limit of the specified surveillance interval (MPS) / frequency (NAPS/SPS) whichever is greater.

Preparer

3.6.2 This delay period is permitted to allow performance of the surveillance. For MPS, NAPS, and SPS, **PERFORM** a risk evaluation, prior to exceeding the 24 hours, for any Surveillance delayed greater than 24 hours, and **MANAGE** the risk impact. The risk assessment shall be performed using the appropriate risk management tool (e.g., EOOS at MPS, NAPS, and SPS).

<u>IF</u> the SSC is <u>NOT</u> modeled in the risk management tool, <u>THEN</u> **CONSULT** the PRA staff for additional guidance. The risk assessment may be documented as part of the immediate determination of operability. Upon completion of a risk assessment, there is <u>NO</u> need to complete an OD evaluation.

Shift Manager

3.6.3 <u>IF</u> the surveillance is <u>NOT</u> performed within the delay period or the surveillance is <u>NOT</u> met when performed within the delay period, <u>THEN</u> IMMEDIATELY **DECLARE** the SSC inoperable (i.e., the LCO <u>NOT</u> met) and **ENTER** the applicable action / condition.

K/A Number: G2.3.11, Radiation Control, Ability to control radiation releases, (CFR: 41.11 / 43.4 / 45.10

Level: SRO Tier #: 3 Group #: IR – RO: 3.8 IR-SRO: 4.3

Proposed Question:

Initial Conditions:

- “A” WGDT is in service (lined up).
- “B” WGDT release is in progress in accordance with OP-23.2.4, “Release of Waste Gas Decay Tank 1B.”
- Annunciator 0-RMA-C6, “ Process Vent Part Alert /Hi” is received.
- The BOP has verified 1-GW-RM-130A, Rad Monitor Process Vent Particulate is valid and is reading greater than the “HI” alarm setpoint.

Which ONE of the following completes the statement:

1) __________ will automatically isolate to STOP the WGDT release flow path. AND

2) The quantity of radioactive material in each gas storage tank shall be limited to $\leq$ 24,600 curies of noble gases in order to provide assurance that in the event of an uncontrolled release of the tanks contents, the dose to an individual standing at the exclusion area boundary will not exceed _________________.

A. 1) 1-GW-FCV-101, Process Vnt WGDT Effluent Flow Controller.
   2) 0.5 rem TEDE over a 2 hour period.

B. 1) 1-GW-FCV-160, Ctmt Vac Pump Disch Hdr Isol.
   2) 25 rem to the thyroid from iodine exposure over a 2 hour period.

C. 1) 1-GW-FCV-160, Ctmt Vac Pump Disch Hdr Isol.
   2) 0.5 rem TEDE over a 2 hour period.

D. 1) 1-GW-FCV-101, Process Vnt WGDT Effluent Flow Controller.
   2) 25 rem to the thyroid from iodine exposure over a 2 hour period.

Proposed Answer: A

Explanation: 1) The automatic actions for HI alarm on 1-GW-RM-130A is to automatically isolate 1-GW-FCV-101 and 1-GW-FCV-160. Only 1-GW-FCV-101 will stop the release as 1-GW-FCV-160 is an input to the system (see attached figure). 2) TS 3.11, Radioactive Gas Storage; Specification B, Gas Storage Tanks; states the quantity of radioactivity contained in each storage tank… shall be limited to ≤ 24, 600 curies of noble gases (considered as Xe-133). The Basis for this requirement states that; the quantity of radioactivity is less than the quantity which provides assurance that in the event of an

uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

Technical Reference: Tech Specs 3.11.B Basis, Waste Gas Storage; ND-81.2-LP-5, Tech Spec and 10 CFR 100; ARP 0-RMA-C6, Process Vent Particulate ALERT/HI.

Reference Provided to Applicant: No

Learning Objective: ND-92.4-LP-1, GW and LW, Objective B, Describe the operation of the Gaseous Waste System.

ND-95.2-LP-1, Radioactive release, Objective B, Describe the Radioactive Gas Release Accidents as presented in Chapter 14 of the Updated Final Safety Analysis Report. Waste Gas Decay Tank.

ND-81.2-LP-5, 10CFR and Tech Spec, Objective C, State the purpose of 10CFR100.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Knowledge

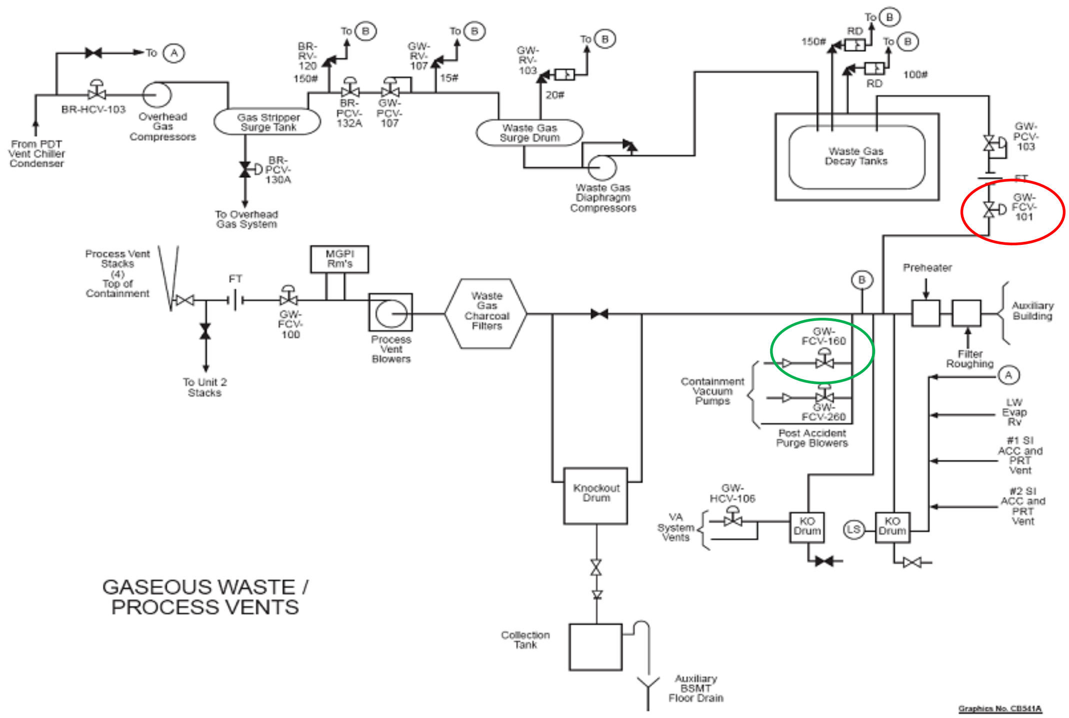
10 CFR Part 55 Content: (CFR: 41.11 / 43.4 / 45.10

Comments:

Distractor Analysis

A. Correct.
B. Incorrect – 1) Incorrect. 1-GW-FCV-160 will NOT isolate the release. Plausible because this is one of the automatic actions that will occur. 2) Incorrect. TS 3.11 Basis states that quantity will limit dose to .5 Rem over 2-hour period. Plausible because 25 Rem is whole body limit for the exclusion area design basis for postulated fission product release. Actual thyroid limit is 300 Rem but 25 Rem used to ensure there is no basis for claiming this is correct.(ND-81.2-LP-5).
C. Incorrect – 1) Incorrect. 1-GW-FCV-160 will NOT isolate the release. Plausible because this is one of the automatic actions that will occur. 2) Correct.
D. Incorrect – 1) Correct. 2) Incorrect. TS 3.11 Basis states that quantity will limit dose to .5 Rem over 2-hour period. Plausible because 25 Rem is whole body limit for the exclusion area design basis for postulated fission product release. Actual thyroid limit is 300 Rem but 25 Rem used to ensure there is no basis for claiming this is correct.(ND-81.2-LP-5).

References for part 1. Both 1-GW-FCV-101 and 1-GW-FCV-160 will isolate, but only GW-FCV-101 will stop the release.



NUMBER	PROCEDURE TITLE	REVISION
0-RMA-C6	PROCESS VENT PART ALERT / HI	3
		PAGE 3 of 7

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION: When CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-160 or 1-GW-FCV-260 is closed, the associated Vacuum Pumps must be placed in OFF.

* *

NOTE: If a high alarm has actuated, the automatic functions associated with that monitor shall be verified or performed.

4.___ CHECK AUTOMATIC ACTIONS:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a) Decay Tank Bleed Isolation valve 1-GW-FCV-101 - CLOSED	☐ a) Manually close valve(s).
b) Check the following	b) Do the following:
☐ • CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-160 - CLOSED	1) Check or place Ctmt Vacuum pumps in OFF
☐ • CTMT Vacuum Pump Discharge Isolation valve 1-GW-FCV-260 - CLOSED	☐ • 1-CV-P-1A
	☐ • 1-CV-P-1B
	☐ • 2-CV-P-1A

References for part 2.

Gas Storage Tanks

The tanks included in Specification 3.11.B are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

Reference for distractor (B) and (D). Wrong because wrong basis and thyroid value incorrect.

6. Each nuclear plant site has a defined radius for the exclusion and low population zones. The outer boundary of the exclusion area is "of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 Rem or a total radiation dose in excess of 300 Rem to the thyroid from iodine exposure." The exclusion area is bounded by a 1650-foot radius circle centered at the Unit 1 reactor containment building (*TS 5.1*). The circle size was determined by the shortest distance to the site boundary and is sufficient, in conjunction with the plant design, to ensure that the dose limitations of Part 100 are met. The exclusion area is owned by and is under control of Virginia Power.

K/A Number: G2.3.12, Radiation Control, Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (CFR: 41.12 / 45.9 / 45.10)

Level: SRO Tier #: 3 Group #: IR – RO: 3.2 IR-SRO: 3.7

Proposed Question:

Unit 2 is currently in a Refueling Shutdown with Reactor Head Lift in progress when the following occurs:

- Reactor Head dropped three (3) feet during movement to the head stand in the containment basement.
- A Supplemental employee is pinned beneath the Reactor Head lift rigging.
- Emergency Plan has been implemented based on HU6.1, Judgment, following consultation with the Operations Manager.
- The initial Emergency Medical Technicians (EMTs) have reported that the individual is alive, but contaminated and will require an ambulance.
- Additional EMTs have been dispatched in order to rescue the injured person.
- Dominion will be issuing a Press Release.
- The general dose rates in the area of the injured person is 1.5 R/hr.

Based on current conditions:

1) What is the maximum allowable dose (TEDE) that can be authorized by the SEM to rescue the individual?
2) What is the time requirement to report this event to the NRC.

(REFERENCE PROVIDED)

A. 1) 10 REM.
 2) 8 Hours.

B. 1) 10 REM.
 2) 24 Hours.

C. 1) 25 REM.
 2) 1 Hour.

D. 1) 25 REM.
 2) 4 Hours.

Proposed Answer: C

Explanation: 1) EPIP-4.04 authorizes a maximum of 25 REM for search and rescue, first aid, and removal of injured personnel where there is reasonable expectation that the individual is alive within the affected area. 2) VPAP-2802, Section 6.3 Note 1: Notifications for events that exceed an Emergency

Action Level, as specified in EPIP-1.01, Emergency Manager Controlling Procedure, are controlled by EPIP-2.01, Notification of State and Local Governments and EPIP-2.02, Notification of NRC. See also Steps 6.3.5 and 6.3.7. [10 CFR 50.72(a)(3), 10 CFR 50.72(c)(1), 10 CFR 50.72(c)(2)]. With the Emergency Plan in effect, the NRC notifications will be driven by EPIP-2.02, Notification of NRC.

Technical Reference: VPAP-2802, Notifications and Reports. EPIP-4.04, Emergency Personnel Radiation Exposure, Attachment 1, Emergency Exposure Limits.

Reference Provided to Applicant: YES, VPAP -2802 pages 78 – 89. This pertains to 1-hour, 4-hour, 8-hour, and 24-hour reporting requirements.

Learning Objective: ND-81.2-LP-3, External Exposure. Objective C: "Explain the federal exposure limits for Total Effective Dose Equivalent, extremity, skin, and Lens of the Eye doses including necessary requirements and limitations for extensions." SROU-02, Admin Procedures. Tier 2. VPAP-2802 Notifications and Reports (SRO.) (Ensure memorization of 1 hour or less notifications.)

Question Source: NEW

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.12 / 45.9 / 45.10) 10 CFR 55.43(b)(4)

Comments:

Distractor Analysis

A. 1) 10 REM. Incorrect, EPIP 4.04 allows 25 REM (not 10) for lifesaving activities. This is plausible if SRO candidate confuses limit for lifesaving with limit for protecting equipment.
2) 8 Hours. Incorrect - VPAP-2802 transport of a contaminated injured personnel is an 8-hour report; however the report of the event and subsequent notification of transport of a contaminated individual will be controlled by EPIP-2.02, Notification of NRC.

B. 1) 10 REM. Incorrect, EPIP 4.04 allows 25 REM (not 10) for lifesaving activities. This is plausible if SRO candidate confuses limit for lifesaving with limit for protecting equipment.
2) 24 Hours. Incorrect, per VPAP-2802 transport of a contaminated injured personnel is an 8-hour report. Plausible because a 24-hour notification is required for an event involving licensed material that causes or threatens to cause an individual to receive a TEDE exceeding 5 REM.

C. 1) 25 REM. Correct per EPIP 4.04, Attachment 1.
2) 1 Hours. Correct per VPAP-2802, Note 1 of Section 6.3.

D. 1) 25 REM. Correct per EPIP 4.04.
2) 4 Hours. Incorrect, per VPAP-2802 transport of a contaminated injured personnel is an 8-hour report. Plausible because a 4 hour report is required IAW VPAP-2802 "Media

Significant" or "Newsworthy" as stated in Note prior to item 5. Stated in Stem that Dominion will make a Press Release.

References for part 1 correct answer and distractor.

NUMBER EPIP-4.04	ATTACHMENT TITLE	ATTACHMENT 1
REVISION 9	EMERGENCY EXPOSURE LIMITS	PAGE 1 of 1

TABLE 1: EPA-400 EMERGENCY EXPOSURE LIMITS			
ACTIVITY	TEDE (Rem)	LDE (Rem)	SDE, THY, CDE, OR OTHER ORGAN (Rem)
GENERAL EMERGENCY EXPOSURE ACTIVITIES	5	15	50
PROTECTING VALUABLE PROPERTY (1)	10	30	100
LIFESAVING OR PROTECTION OF LARGE POPULATIONS (2)	25	75	250
LIFESAVING OR PROTECTION OF LARGE POPULATIONS (3)	> 25 Only on a voluntary basis to persons fully aware of the risks involved.	> 75	> 250

(1) Protecting Valuable Property:

- To save valuable equipment.
- To limit off-site releases.

(2) Lifesaving Activity:

- For search and rescue, first aid, and removal of injured personnel where there is reasonable expectation that the individual(s) is alive within the affected area.
- For entry to correct conditions which, if left uncorrected, could result in on-site or off-site injury.

(3) No limit given in extreme case because loss of thyroid may be acceptable to save a life. This may not be necessary if respirators and/or blocking agents are available for rescue personnel.

References for part 2 correct answer. (VPAP 2802 page 84)

DOMINION

VPAP-2802
REVISION 37
PAGE 67 OF 198

6.3 Immediate to 72-Hour Notifications

This subsection consolidates requirements for situations or events addressed by Subsections 6.5 through 6.29, for which notifications or reports are required within 72 hours.

6.3.1 **General Requirements**

a. When this subsection (Subsection 6.3) designates someone other than the Shift Manager or a member of Station management to notify a government agency, that person shall ensure the Shift Manager or a member of Station management is advised before making the notification. See also Step 6.3.4.a.5.

NOTE: Notifications for events that exceed an Emergency Action Level, as specified in EPIP-1.01, Emergency Manager Controlling Procedure, are controlled by EPIP-2.01, Notification of State and Local Governments and EPIP-2.02, Notification of NRC. See also Steps 6.3.5 and 6.3.7. **[10 CFR 50.72(a)(3), 10 CFR 50.72(c)(1), 10 CFR 50.72(c)(2)]**

References for part 2 distractors.

6.3.4 **Four-hour Notifications**

NOTE: Some conditions, indicated by "See EPIP-1.01," may exceed an Emergency Action Level (EAL) as specified in EPIP-1.01, Emergency Manager Controlling Procedure. If a condition exceeds an EAL, EPIPs control State and Federal agency notifications. If an event or condition does not exceed an EAL, it may still be reportable in accordance with this procedure.

NOTE: "Notification to other government agencies has been or will be made" is not necessarily an automatic notification to the NRC. Refer to NUREG – 1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, for discussions and examples (e.g., newsworthy events, environmental events, spurious, emergency siren actuations) or contact Station Licensing if clarification is needed. **[NUREG-1022, Section 3.2.12]**

5. Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned, or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials. **[Commitment 3.2.12]** **[10 CFR 50.72(b)(2)(xi)]**

6.3.5 **Eight-hour Notifications**

a. As soon as practical, but within eight hours, the Shift Manager shall notify the NRC Operations Center via the ENS of:

6. Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment. See also Step 6.27.2.
[10 CFR 50.72 (b)(3)(xii) and 10 CFR 72.75 (c)(3)]

6.3.6 **Twenty-four Hour Notifications**

a. As soon as practical, but within 24 hours, the Shift Manager shall notify the NRC Operations Center with the ENS of [10 CFR 20.2202(b)]:

NOTE: The requirements of Step 6.3.6.a.1. do not apply to doses that result from planned special exposures, that are within the limits for planned special exposures, and that are reported in accordance with Step 6.10.11.c. [10 CFR 20.2202(e)]

1. An event that involves licensed material possessed by Dominion that may have caused or threatens to cause:
   - An individual to receive, in a period of 24 hours:
     - A total effective dose equivalent exceeding 5 rems
     - An eye dose equivalent exceeding 15 rems
     - A shallow-dose equivalent to the skin or extremities exceeding 50 rems
   - Release of radioactive material inside or outside a restricted area, so that, if an individual had been present for 24 hours, they could have received an intake in excess of one occupational annual limit on intake.

   If an event involves radiological overexposure, DEM must be notified as specified in Step 6.27.2. See also Step 6.6.3.c.

K/A Number: G2.4.28, Emergency Procedures/Plans, Knowledge of procedures relating to emergency response to sabotage., (CFR: 41.10 / 43.5 / 45.13)

Level: SRO Tier #: 3 Group #: IR – RO: 3.2 IR-SRO: 4.1

Proposed Question:

Unit 1 and 2 are operating at 100% with all systems and cross-ties OPERABLE.

- A Hostile Force has landed at the Low Level Intake Structure.
- The Outside Watch has been taken hostage, and the Operations vehicle has been used to breach the protected area fence from the construction side into the plant.
- Security Supervisor reports that they are currently engaging the hostile force in the Polishing Building. All personnel should remain clear of the Polishing Building. No other areas are affected at this time.
- Explosions were heard from the vicinity of the Polishing Building.

Which ONE of the following states the classification for this event?

(REFERENCE PROVIDED)

A. HG4.1, General Emergency.

B. HA2.1, Alert.

C. HA4.1, Alert.

D. HS4.1, Site Area Emergency.

Proposed Answer: D

Explanation:

Technical Reference: Surry EAL Matrix, Surry Power Station Emergency Action Level Technical Basis Document.

Reference Provided to Applicant: Yes, EAL Matrix.

Learning Objective:

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13) 10CFR 55.43(b)(5)

Comments: Question is SRO-only due to EAL Classification.

Distractor Analysis

A. Incorrect – Hostile Action within the Protected Area is occurring however there is no report of loss of the ability to operate equipment needed to maintain safety functions.
B. Incorrect - EAL does apply, but is not the highest classification.
C. Incorrect – EAL does apply, but is not the highest classification.
D. Correct – EAL does apply, and is the highest classification. There is a hostile force that is occurring within the protected area.

Reference for correct answer D and distractor A.

4 **Security**	HG4 Hostile action resulting in loss of physical control of the facility 1 2 3 4 5 6 DEF **HG4.1** A hostile action has occurred such that plant personnel are unable to operate equipment required to maintain safety functions **OR** A hostile action has caused failure of Spent Fuel Cooling Systems and imminent fuel damage is likely for a freshly off-loaded reactor core in pool (i.e. freshly off-loaded reactor core means any spent fuel in the Spent Fuel Pit)	HS4 Hostile action within the Protected Area 1 2 3 4 5 6 DEF **HS4.1** A hostile action is occurring or has occurred within the Protected Area as reported by the Security Shift Supervisor
		HS5 Control Room evacuation has been initiated and plant control cannot

Reference for Distractor B.

HA2 Fire or explosion affecting the operability of plant safety-related structures, systems or components required to establish or maintain safe shutdown

1	2	3	4	5	6	DEF

HA2.1

Fire or explosion in any Table H-1 area

AND EITHER:

Plant personnel report visible damage to any safety-related structure, system, or component within the area

OR

Affected system parameter indications show degraded performance

Table H-1 Safe Shutdown Areas
- Cable Vaults & Tunnels - Emergency Switchgear & Relay Rooms - Unit Switchgear Room - Reactor Containment - Safeguards Complex (incl. Cont. Spray Pump Area & Main Steam Valve House) - Main Control Room - Emergency Diesel Generators Rooms 1, 2 and 3 - Auxiliary / Fuel / Decontamination Buildings - Underground Fuel Oil Pump House Rooms - Intake Structure - Emergency Service Water Pump House - Turbine Building - Mechanical Equipment Rooms 3, 4 & 5 - Cable Tray Room

None

Reference for Distractor C

HA4 Hostile action within the Owner Controlled Area or airborne attack threat

1	2	3	4	5	6	DEF

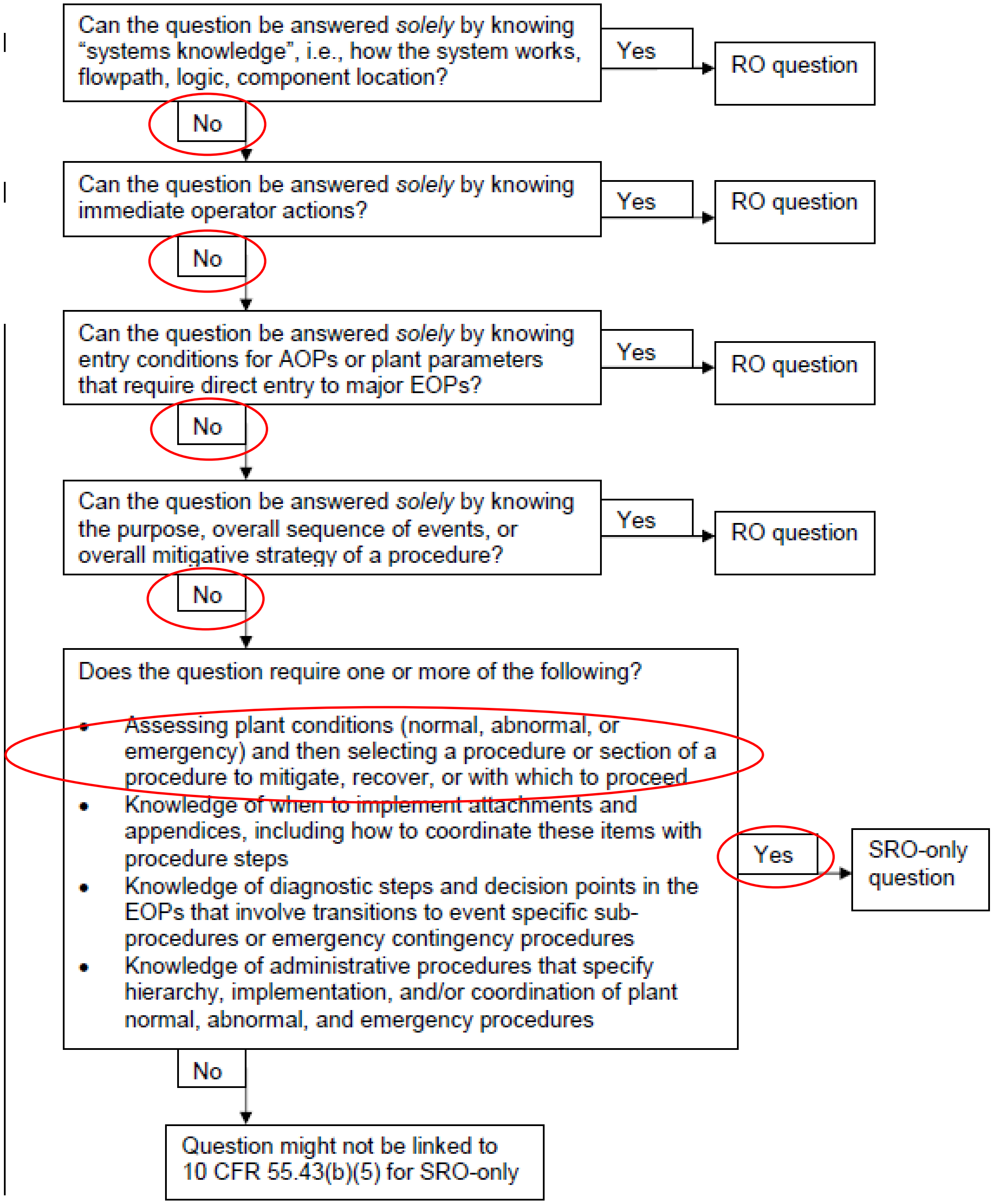
HA4.1

A hostile action is occurring or has occurred within the Owner Controlled Area as reported by the Security Shift Supervisor

OR

A validated notification from NRC of an airliner attack threat < 30 min. away

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: WE11EG2.1.32, Loss of Emergency Coolant Recirc. /4, Ability to explain and apply all system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)

Level: SRO Tier #: 1 Group #: 1 IR – RO: 3.8 IR-SRO: 4

Proposed Question:

Unit 1 is operating at 100% power.

Unit 2 has experienced a Loss of Offsite power and a LBLOCA:

- EDG 3 failed to start.
- Containment Pressure is 44 psia and slowly rising.
- Annunciator, 2B-B1, CS PP2A LOCKOUT OR OL TRIP is lit.
- All RCPs have secured.
- RCS pressure is 50 psig and steady.
- Core Exit Thermocouple temperature: 610 ^{o}F and slowly rising.
- RVLIS 44% and steady.
- The Team has just established Cold Leg Recirculation in accordance with 2-ES-1.3, Transfer to Cold Leg Recirculation, (step 5 completed).
- Oscillating Pump Amps and Discharge Pressure indication on the following: LHSI pump, 2-SI-P-1A; Inside Recirc Spray pump, 2-RS-P-1A; Outside Recirc Spray Pump, 2-RS-P-2A.

Which ONE of the following identifies the procedure that should be performed for the NEXT step?

A. Continue in ES-1.3, Transfer to Cold Leg Recirculation.

B. Go to FR-C.2, Response to Degraded Cooling.

C. Go to ECA-1.1, Loss of Emergency Coolant Recirculation.

D. Go to FR-Z.1, Response to Containment High Pressure.

Proposed Answer: C

Explanation: With indications given, a loss of Sump suction has occurred (oscillating pump amps and discharge pressure indication). With only one train of equipment available (due to loss of J bus) there is a complete loss of sump suction. Note before step 1, and Attachment 1 step 1.e in ES-1.3 states that if there is a complete loss of sump suction capability then GO TO ECA-1.1.

Technical Reference: 1-ECA-1.1, 1(2)-OP-CS-004.

Reference Provided to Applicant: No

Learning Objective:.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.2 / 45.12)

Comments:

Distractor Analysis:

A. Incorrect – If there were two trains of LHSI available then this would be correct as the team would first secure one LHSI pump to attempt to relieve the sump blockage. Plausible because efforts to free sump blockage are first performed in ES-1.3.
B. Incorrect – With indications of Sump Blockage, priority is to relieve sump blockage Conditions. Plausible because entry conditions for FR-C.2 (Low Subcooling, RCPs secured, RVLIS < 46%) are met. SRO must prioritize most important action.
C. Correct – With a Total Loss of Sump suction capability, ECA-1.1 is required.
D. Incorrect - With indications of Sump Blockage, priority is to relieve sump blockage Conditions. Plausible because entry conditions for FR-Z.1 are met (Containment pressure > 23 psia, no Containment Spray pumps running). SRO must prioritize most important action.

References for correct answer:

NUMBER	PROCEDURE TITLE	REVISION
1-ES-1.3	TRANSFER TO COLD LEG RECIRCULATION	20
		PAGE 2 of 10

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* *

CAUTION:
- SI recirculation flow to RCS must be maintained at all times.
- Transfer to recirculation may cause high radiation in the Auxiliary Building.

* *

NOTE:
- Steps 1 through 5 should be performed without delay. FRs should NOT be implemented before the completion of these steps.
- If sump blockage or a complete loss of sump suction capability occurs, FRs should NOT be implemented until directed in Attachment 1, or in 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

NUMBER 1-ES-1.3	ATTACHMENT TITLE CONTAINMENT SUMP SCREEN BLOCKAGE - CONTINGENCY ACTIONS	ATTACHMENT 1
REVISION 20		PAGE 1 of 1

* *

CAUTION: If the suction source is lost to any SI or spray pump, the pump should be stopped.

* *

NOTE: This attachment remains in effect during sump recirculation.

1. IF any LHSI Pump indicates sump blockage, THEN do the following:

   ___ a. Check or place all CHG Pumps in PTL.

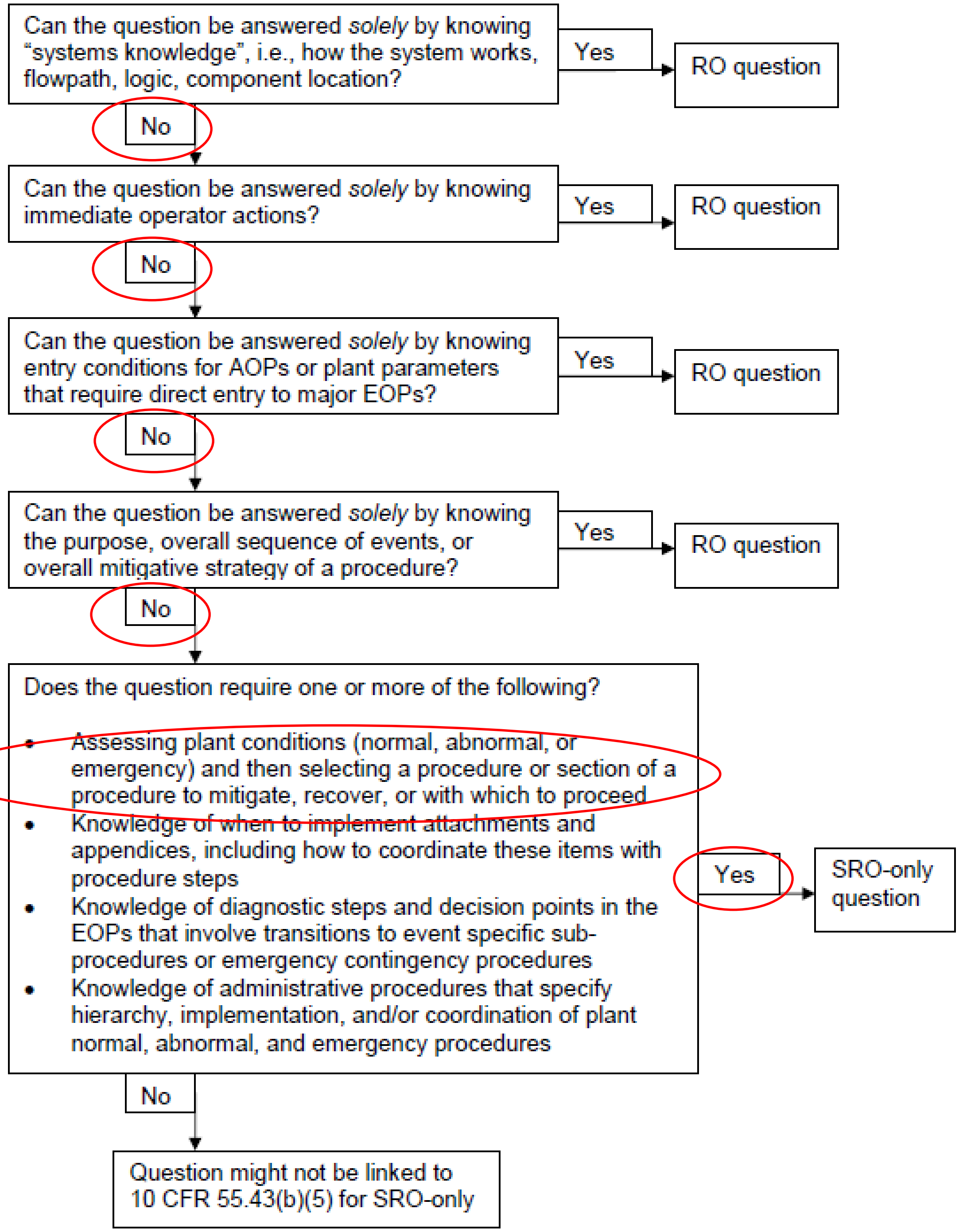
   ___ b. Stop one affected LHSI Pump and place in PTL.

   ___ c. Align CHG pump suction crosstie IAW Attachment 2.

   ___ d. Monitor running LHSI Pump.

   ___ e. IF sump blockage is indicated on running LHSI Pump, THEN stop running LHSI pump AND GO TO 1-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: WE12EG2.4.4, Steam Line Rupture - Excessive Heat Transfer /4, Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures., (CFR: 41.10 / 43.2 / 45.6)

Level: SRO Tier #: 1 Group #: 1 IR – RO: 4.5 IR-SRO: 4.7
(linked to WE E12, Uncontrolled Depressurization of all SGs).

Proposed Question:

Unit 2 is stable at approximately 2% reactor Power following a Refueling Shutdown.

- Main Steam Lines have been warmed, and the MSTVs are open.
- Steam Dumps are in AUTO with 4% demand indicated.
- The BOP is performing 2-OPT-FW-003, Turbine Driven Auxiliary Pump 2-FW-P-2, full flow test.

The following indications are noted:

- 2-FW-P-2 Feed Pressure light goes out.
- Annunciator 2H-C8, AFW PP3B Lockout or OL Trip is lit.
- AFW Flow: 140 gpm to each Steam Generator (420 gpm total).
- S/G A/B/C Pressure: 400 psig and decreasing rapidly.
- S/G Level: S/G “A” – 58% (WR), S/G “B” – 55% (WR), S/G “C” – 50% (WR), (All are lowering).
- Containment Pressure: 10.5 psia and steady.
- Team has just completed step 6 of E-0, Reactor Trip or SI.

Which ONE of the following identifies the required sequence of procedure transitions implemented in response to this event?

A. E-2 Faulted SG Isolation, E-1 Loss of Reactor or Secondary Coolant, ES-1.1 SI Termination.

B. FR-H.1 Response to Loss of Secondary Heat Sink, E-2 Faulted SG Isolation, ECA-2.1 Uncontrolled Depressurization of all SGs.

C. E-2 Faulted SG Isolation, FR-H.1 Response to Loss of Secondary Heat Sink.

D. E-2 Faulted SG Isolation, ECA-2.1 Uncontrolled Depressurization of all SGs.

Proposed Answer: D. E-2 (Faulted SG Isolation), ECA-2.1 (Uncontrolled Depressurization of all SGs).

Explanation: For indications and initial condition, all 3 SGs will be faulted with a leak path upstream of the MSTVs but outside containment. Containment is not adverse therefore AFW flow > 350 gpm meets requirement, therefore FR-H.1 is not applicable.

Technical Reference: ECA-2.1, Uncontrolled depressurization of All Steam Generators.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-22, ECA-2.1, Objective B, Given a copy of ECA-2.1, Uncontrolled Depressurization of All Steam Generators, apply the basis of each procedural step to be able to determine the appropriate response for a given plant condition.

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 41.10 / 43.2 / 45.6) 10 CFR 55.43(b)(5)

Comments:

Distractor analysis:

A. Incorrect – procedural progression would occur as stated IF the candidate assumes steam isolation in E-2. E-2 step 2 requires at least one intact SG.
B. Incorrect – Per FR-H.1 Entry would only occur IF candidate assumes all three SGs AFW flow is isolated in E-2 or Adverse Containment.
C. Incorrect – FR-H.1 Entry would only occur IF candidate assumes all three SGs AFW flow is isolated in E-2.
D. Correct – Based on the event in progress, ECA-2.1 would be entered. Team would not throttle <350 gpm until step 2 of ECA-2.1.

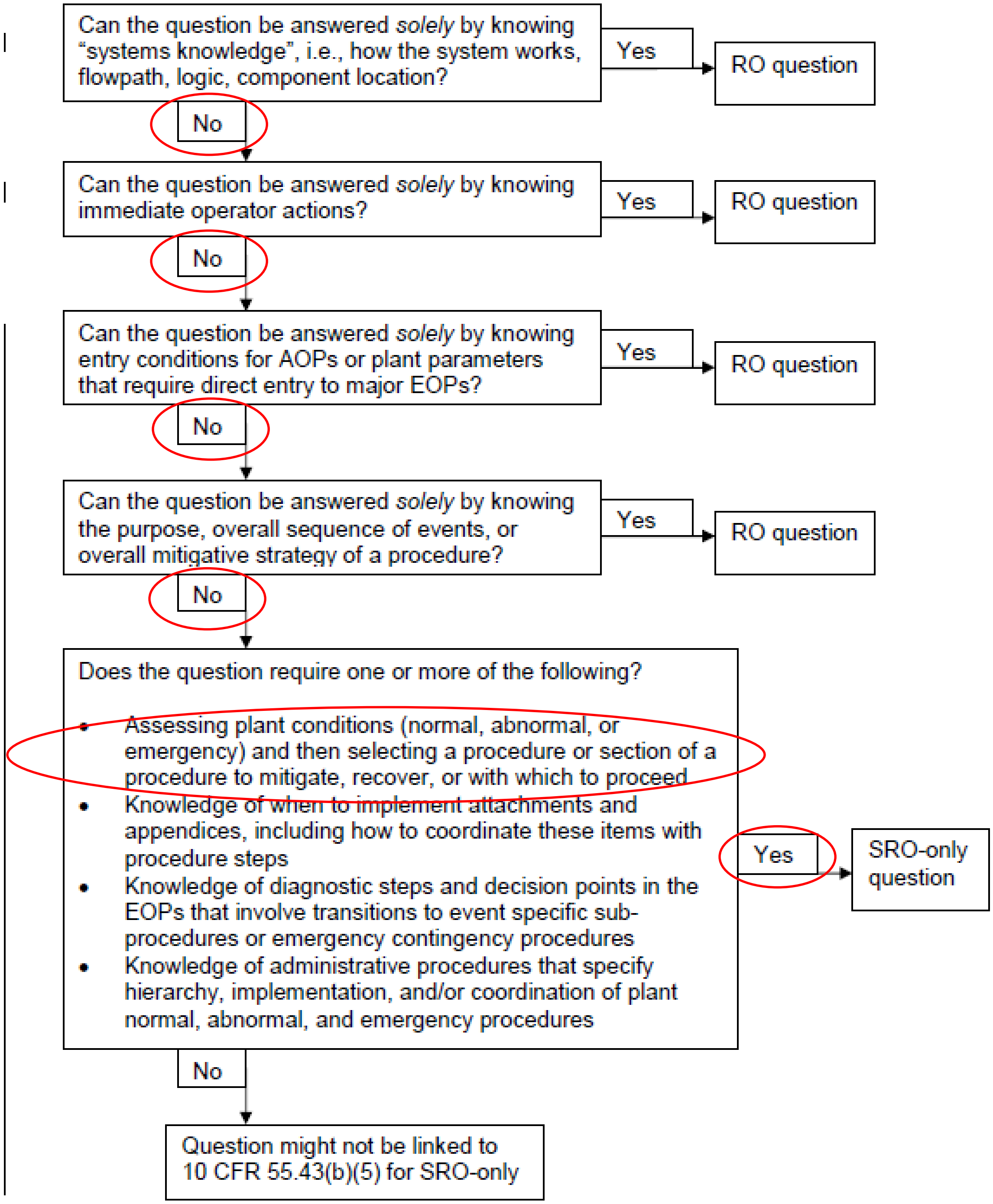
NUMBER	PROCEDURE TITLE	REVISION
2-E-0	REACTOR TRIP OR SAFETY INJECTION	74
		PAGE 6 of 15

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.___	CHECK IF SGs ARE NOT FAULTED: • Check pressures in all SGs: ☐ • STABLE OR INCREASING AND ☐ • GREATER THAN 100 PSIG	☐ IF any SG pressure decreasing in an uncontrolled manner OR is completely depressurized, THEN GO TO 2-E-2, FAULTED STEAM GENERATOR ISOLATION.

NUMBER	PROCEDURE TITLE	REVISION
2-E-2	FAULTED STEAM GENERATOR ISOLATION	22
		PAGE 2 of 6

STEP	ACTION/ EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		MS NRV.
2.___	CHECK IF ANY SG SECONDARY SIDE IS INTACT: ☐ • Check pressures in all SGs - ANY STABLE OR INCREASING	☐ • IF all SG pressures decreasing in an uncontrolled manner, THEN GO TO 2-ECA-2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.
3.	IDENTIFY FAULTED SG(s):	

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)



K/A Number: WE16EA2.2, High Containment Radiation / 9, Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments, (CFR: 43.5 / 45.13)

Level: SRO Tier #: 1 Group #: 2 IR – RO: 3 IR-SRO: 3.3

Proposed Question:

A LBLOCA occurred on Unit 1, followed by fuel failure. A Category 4 Hurricane has caused power lines to drop on numerous roadways, and tornadoes have been seen throughout Surry and James City County.

- At 0400: A General Emergency has just been declared on FG1.1.
- At 0410: State and Local authorities are notified in accordance with EPIP-2.01, Notification of State and Local Governments.
- HP reports the following dose assessment data based on the release in progress:
- Projected TEDE dose is 0.8 Rem at 2 miles.
- Projected Thyroid CDE dose is 7.0 Rem at 2 miles.
- Projected Thyroid CDE dose is 5.2 Rem at the Site Boundary.
- The primary downwind sector is "C" with "B" and "D" as buffer sectors.

Which one of the following correctly answers the following questions:

1) What is the correct Protective Action Recommendation?
2) What is the latest time that the PAR must be made to the State?

(REFERENCE PROVIDED)

A. 1) Shelter-in-place: 2 Mile radius and 5 Miles downwind. 2) 0415.

B. 1) Expanded PAR: Derive from EPIP-4.07. 2) 0415.

C. 1) Shelter-in-place: 2 Mile radius and 5 Miles downwind. 2) 0425.

D. 1) Expanded PAR: Derive from EPIP-4.07. 2) 0425.

Proposed Answer: A 1) PAR A Shelter-in-place: 2 Mile radius and 5 Miles downwind. 2) 0415

Explanation: 1) EPIP-1.06, Attachment 2 first checks for any "known impediments". Known impediments are defined as any severe weather such as hurricanes, tornados, flooding or blizzards; and any traffic issues such as dropped power lines. Given information provides "Hurricane, numerous tornadoes, and dropped power lines". This meets the first the criteria for known impediments that make evacuation dangerous. Therefore in accordance with EPIP-1.06 Attachment 2 PAR A; Shelter in place 2 mile radius and 5 miles downwind is the appropriate PAR. 2) EPIP-2.01, Notification of State and

Local requires PARs be made within 15 minutes of General Emergency Declaration therefore PARs must be communicated to the State no later than 0415.

Technical Reference: EPIP-1.06, / Recommendations. EPIP-2.01, Notification to State and Local Governments.

Reference Provided to Applicant: Yes (EPIP-1.06)

Learning Objective: ND-95.5-LP-2, SEM, Objective D, Describe the corrective actions required by a classified plant emergency.

Question Source: BANK (LORP, CEP 275, LEPIP060)

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

Distractor Analysis:

A. Correct: 1) PAR A Shelter-in-place: 2) 0415 is the correct time.
B. Incorrect. 1) PAR C Expanded PAR is incorrect because of impediments that make evacuation dangerous. This is plausible because dose at 2 miles is greater than allowed making PAR C the correct PAR if there would be no known impediments. 2) 0415 is the correct time.
C. Incorrect. 1) PAR A Shelter-in-place is the correct PAR. 2) 0425 is incorrect because the requirement is to make PARs within 15 minutes of GE declaration. Plausible because a common misconception is to make PARS within 15 minutes of Notification, and normal practice is to make PARs after the Notification.
D. Incorrect. 1) PAR C Expanded PAR is incorrect because of impediments that make evacuation dangerous. This is plausible because dose at 2 miles is greater than allowed making PAR C the correct PAR if there would be no known impediments. 2) 0425 is incorrect because the requirement is to make PARs within 15 minutes of GE declaration. Plausible because a common misconception is to make PARS within 15 minutes of Notification, and normal practice is to make PARs after the Notification.

References for Part 1 from EPIP-1.06.

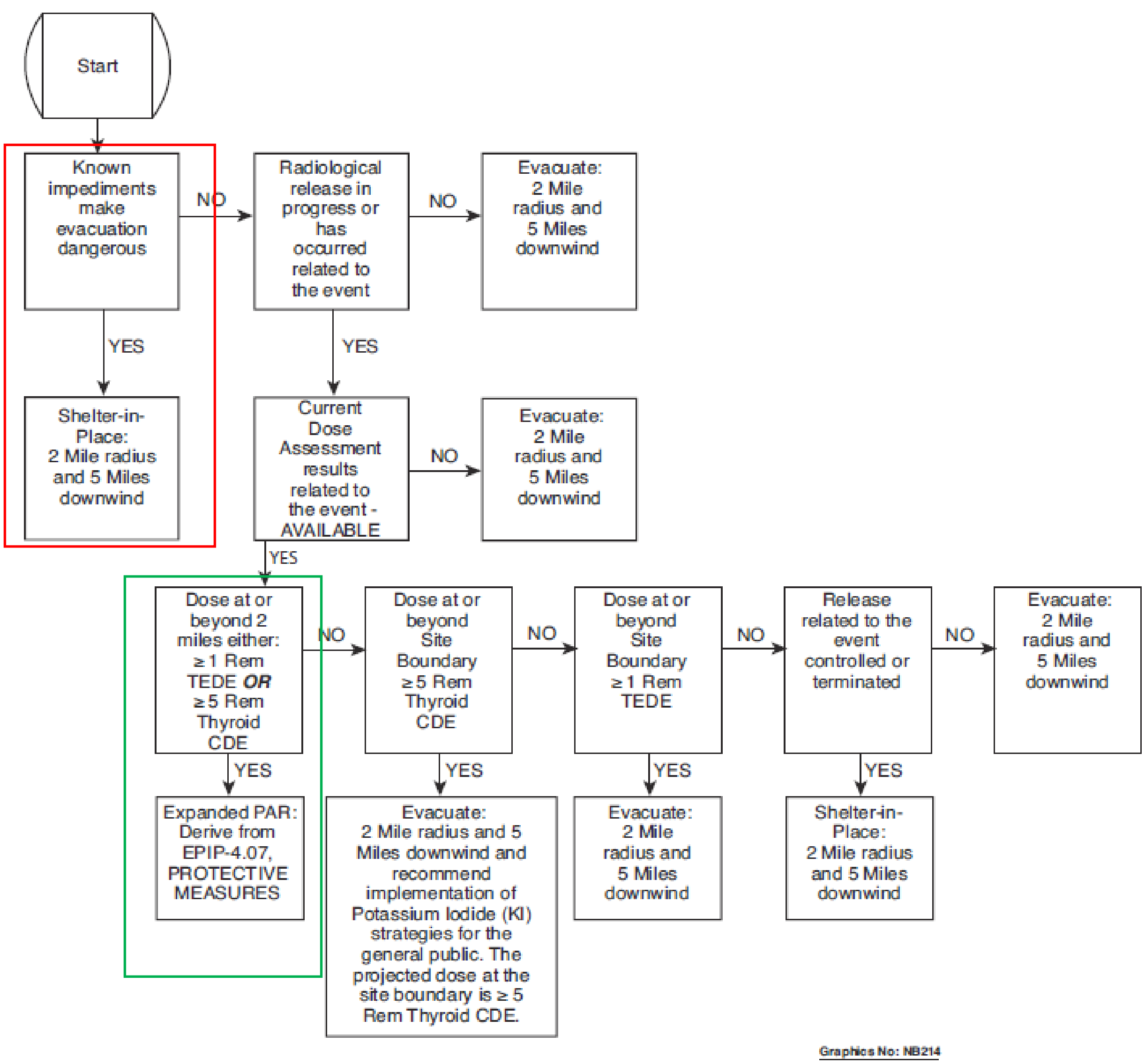
NUMBER EPIP-1.06	ATTACHMENT TITLE REFERENCE INFORMATION (SPS)	ATTACHMENT 4
REVISION 10		PAGE 1 of 1

KNOWN EVACUATION IMPEDIMENTS:	Refers to already known conditions, for example: • Severe weather such as hurricanes, tornados, flooding, or blizzards. • Traffic issues such as inadequate roads, major accident(s). • An attack on an off-site infrastructure such as water, power lines, transportation, communication systems, and public institutions including schools, post offices and prisons. It is not expected that Protective Action Recommendation (PAR) development be delayed by attempting to obtain information from outside resources.

References for part 1 from EPIP-1.06 shows correct answer (red) and distractor (green).

NUMBER EPIP-1.06	ATTACHMENT TITLE PROTECTIVE ACTION RECOMMENDATION FLOWCHART SPS	ATTACHMENT 1
REVISION 10		PAGE 1 of 1

NOTE: The radiological release related statements below do NOT apply if a radiological release is occurring due to routine, normal plant operations and NOT related to the event.

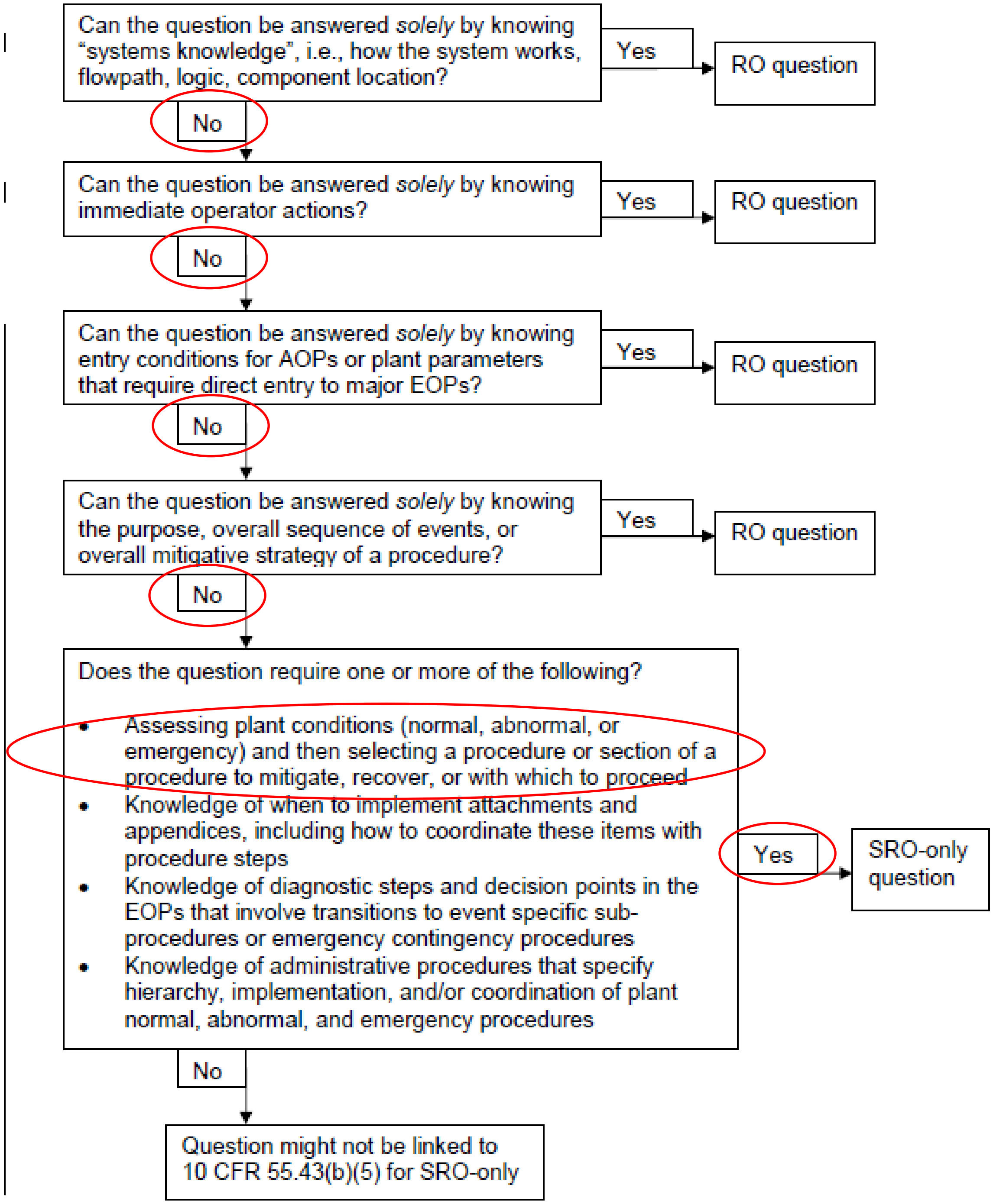


References for Part 2 from EPIP-2.01.

2. PROTECTIVE ACTION RECOMMENDATION (PAR) TRANSMITTAL CRITERIA

   IF message contains an initial PAR (or a PAR change), THEN transmit PAR directly to the Virginia EOC within 15 minutes of the General Emergency declaration (or PAR change development).

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



K/A Number: WE01EA2.1, Rediagnosis / 3, Facility conditions and selection of appropriate procedures during abnormal and emergency operations. (CFR: 43.5 / 45.13)

Level: SRO Tier #: 1 Group #: 2 IR – RO: 3.2 IR-SRO: 4

Proposed Question:

Initial Conditions:

- Unit 1 reactor has been tripped from 100% due to a LOCA.
- The Team has just transitioned to ES-1.2, POST LOCA Cooldown and Depressurization.

Current Conditions:

- Subcooling: 25^{o}F and lowering.
- AFW flow: 140 gpm to each S/G.
- SG level: S/G "A/B" 70% WR and steady. S/G "C" 10% NR and rising.
- Annunciator 1-RM-G8, "CNDSR AIR EJCTR ALERT/FAILURE has just alarmed.
- BOP confirms that 1-SV-RM-111 is at the Alert setpoint and rising.
- The Team became concerned about conflicting indications and transitioned to ES-0.0, Rediagnosis.

Which ONE of the following describes the correct procedural flowpath from ES-0.0?

A. Go directly to E-3, Steam Generator Tube Rupture; then to ECA-3.1, SGTR with Loss of Reactor Coolant Subcooled Recovery.

B. Return to ES-1.2, POST LOCA Cooldown and Depressurization until SI signals have been blocked; then go to E-3, SGTR; then ECA-3.1, SGTR with Subcooled Recovery.

C. Return to ES-1.2, POST LOCA Cooldown and Depressurization until SI signals have been blocked; then go to E-3, SGTR; then ES-3.1, Post SGTR Cooldown using Backfill.

D. Go directly to E-3, SGTR; then go to ES-3.1, Post SGTR Cooldown using Backfill.

Proposed Answer: A

Explanation: Indications given provide the SRO with a LOCA outside containment followe3d by a SGTR. There is no transition from ECA-1.2 to E-3, therefore ES-0.0 will provide a procedural mechanism to deal with the rupture. After entering ES-0.0 step 3 asks if there is a SGTR. If so ES-0.0

states than an E-3 or ECA-3 procedure is in effect. The SRO will then enter E-3, and due to a low subcooling the SRO will then enter ECA-3.1 when directed in E-3.

Technical Reference: ES-0.0, Rediagnosis.

Reference Provided to Applicant: No

Learning Objective: ND-95.3-LP-3, E-3; Objective A, "Given the major action categories associated with E-3, Steam Generator Tube Rupture, explain the purpose of E-3, the transition criteria for entering and exiting E-3 and the types of operator actions that will occur within each category."

Question Source: New

Question History: Last NRC Exam: NO

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: (CFR: 43.5 / 45.13)

Comments:

Distractor Analysis:

A. Correct, ES-0.0 directs entry into E-3 first. Once in E-3, then entry into ECA-3.1 will occur due to low subcooling.
B. Incorrect, because once in ES-0.0, a SGTR will have a higher priority then an E-1 or ES 1.2 procedure. Also SRO needs to understand entry and exit transitions; ES-1.2 is only entered from E-1 or ES-1.1 for small break LOCAs. Additionally ES-1.2 has an exit transition requirement for SGTR. Plausible because leak may not be isolated and SRO may confuse priority of ES-0.0.
C. Incorrect, because once in ES-0.0, a SGTR will have a higher priority then an E-1 or ES-1.2 procedure. Plausible if SRO takes wrong path in ES-0.0 step 3 and goes to ES-1.2 first, instead of E-3. Also if SRO went to E-1 first then Continuous Action would direct entry into E-3.
D. Incorrect. E-3 is correct procedure to go to first. ES-3.1 is not correct because of low subcooling. Plausible because ES-3.1 is desired procedure due to less release to general public.(Ref: ND-95.3-LP-14)

References for ES-0.0; E-3 correct transition, E-1 is not.

NUMBER	PROCEDURE TITLE	REVISION
1-ES-0.0	REDIAGNOSIS	4
		PAGE 3 of 3

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3. ___	CHECK IF SG TUBES ARE RUPTURED: ☐ • ANY SG LEVEL INCREASING IN AN UNCONTROLLED MANNER OR ☐ • ANY SG WITH HIGH RADIATION	☐ An E-1 or ECA-1 series procedure should be in effect.
4. ___	AN E-3 OR ECA-3 SERIES PROCEDURE SHOULD BE IN EFFECT	

References for E-3; ECA-3.1 correct transition, ES-3.1 is NOT.

NUMBER	PROCEDURE TITLE	REVISION
1-E-3	STEAM GENERATOR TUBE RUPTURE	48
		PAGE 14 of 40

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
17. ___	CHECK RUPTURED SG(s) PRESSURE - STABLE OR INCREASING	IF ruptured SG pressure is 250 psi greater than pressure of highest intact SG used for cooldown, THEN do the following: ☐ a) Check ruptured SG isolation. ☐ b) Initiate cooldown at less than 100°F/hr using intact SG(s). ☐ c) Maintain ruptured SG pressure greater than 250 psi above pressure of highest intact SG used for cooldown. ☐ IF 250 psi pressure differential does NOT exist OR can NOT be maintained by cooldown, THEN GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.
18. ___	CHECK RCS SUBCOOLING BASED ON CETCs - GREATER THAN 50°F [105°F]	☐ GO TO 1-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY.

49. ___ GO TO APPROPRIATE POST-SGTR COOLDOWN METHOD AS DIRECTED BY TSC OR PLANT STAFF:

- ☐ • GO TO 1-ES-3.1, POST-SGTR COOLDOWN USING BACKFILL

<u>OR</u>

- ☐ • GO TO 1-ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN

<u>OR</u>

- ☐ • GO TO 1-ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP

Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)

Can the question be answered *solely* by knowing "systems knowledge", i.e., how the system works, flowpath, logic, component location?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry to major EOPs?

Yes → RO question

No ↓

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No ↓

Does the question require one or more of the following?

- Assessing plant conditions (normal, abnormal, or emergency) and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed
- Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- Knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific sub-procedures or emergency contingency procedures
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No ↓

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only