QUESTION 76

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2011
Exam Type:	SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	008	AA2.30
Importance Rating		4.7

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Inadequate core cooling

Proposed Question: SRO Question # 76

The following conditions exist:

- Unit 3 tripped from 100% power due to a Loss of All Feedwater.
- All HHSI Pumps are unavailable.
- A Pressurizer Safety valve is leaking by and will not reseat.

After several minutes in this event:

- The crew has restored Auxiliary Feedwater.
- Unit 3 S/G narrow range levels are all at 35%.
- Seven of the highest Core Exit Thermocouples (CETs) are rising and temperatures are as follows: 1231°F, 1227°F, 1210°F, 1207°F, 1201°F, 1170°F, and 1151°F.

Which ONE of the following describes the required procedure to immediately transition to and the mitigation strategy to establish core cooling?

A. 3-EOP-FR-C.2, Response to Degraded Core Cooling;
Stop a RCP, and if temperatures continue to rise, then perform a < 100°F/hr cooldown to inject Accumulators.

B. 3-EOP-FR-C.1, Response to Inadequate Core Cooling;
Perform a MAXIMUM RATE cooldown to inject the Accumulators and if temperatures continue to rise, then start a RCP

C. 3-EOP-FR-C.2, Response to Degraded Core Cooling;
Perform a MAXIMUM RATE cooldown to inject the Accumulators and if temperatures continue to rise, then start a RCP

D. 3-EOP-FR-C.1, Response to Inadequate Core Cooling;
Stop a RCP, and if temperatures continue to rise, then perform a < 100°F/hr cooldown to inject Accumulators.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. 5 CETs are greater than 1200°F; therefore the correct priority is 3-EOP-FR-C.1. Plausibility – Student must count all CET instruments which leads to 3-EOP-FR.C.1. Also, if the student uses a CET Average temperature it equals less than 1200°F. 3-EOP-FR.C.1 starts a RCP after max. rate cooldown, but 3-EOP-FR-C.2 stops a RCP prior to cooldown.

B. Correct.

C. Incorrect. 5 CETs are greater than 1200°F; therefore the correct priority is 3-EOP-FR-C.1. Plausibility – Student must count all CET instruments which leads to 3-EOP-FR.C.1. Also, plausible because 2nd part is correct.

D. Incorrect, The wrong procedure actions are selected. Plausibility – RCPs are operated in 3-EOP-FR-C.1. The RCP is only started after RCS temperatures continue to rise at Step 19.

Technical Reference(s): 3-EOP-F-0 (Attach if not previously provided)
3-EOP-FR.C.1
3-EOP-FR.C.2

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (3SPK)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

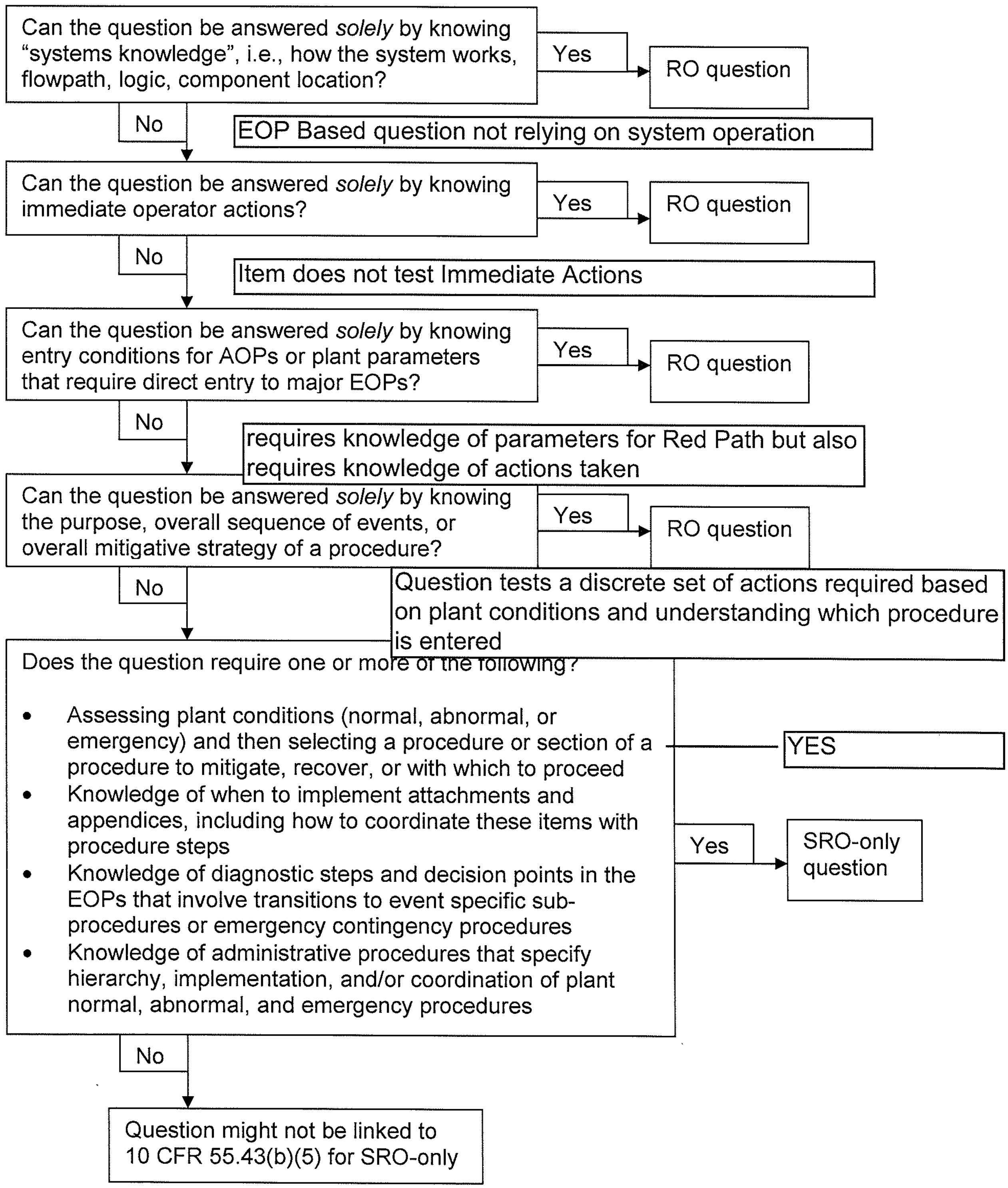
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Matches KA because the applicant must determine and interpret if inadequate core cooling condition exists and prioritize the best mitigation strategy during a vapor space break.

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

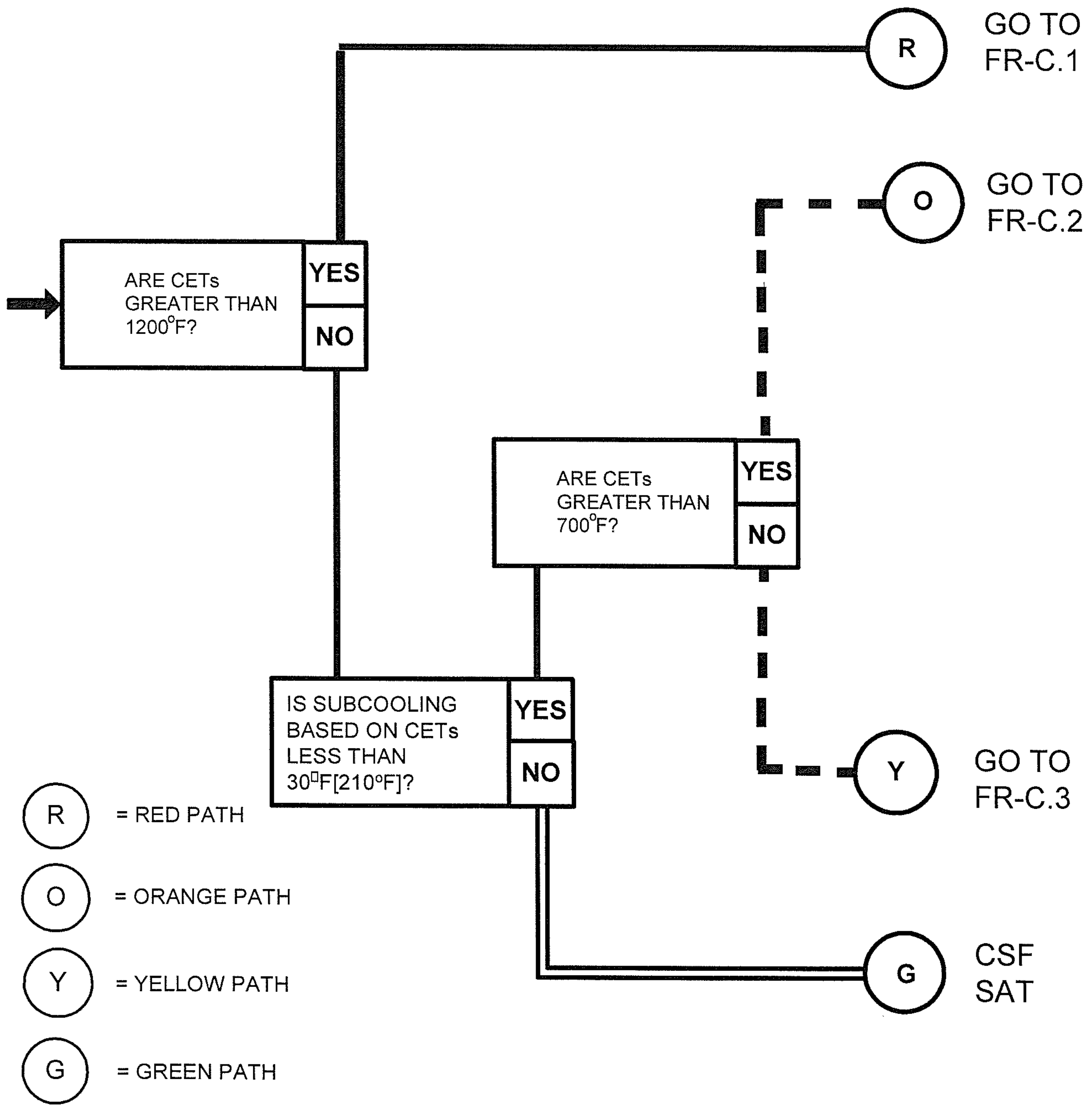


STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ENCLOSURE 2
(Page 1 of 1)
CSF F-0.2 CORE COOLING

NOTE

Obtain core exit temperature using at least five of the hottest core exit thermocouples.



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STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTES

Partial uncovering of S/G tubes is acceptable in the following steps.

Excessive steam flow can lead to main steamline isolation and loss of condenser steam dump capability.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Depressurize All Intact S/Gs To 80 Psig**	
	a. Dump steam to condenser at maximum rate	a. Dump steam at maximum rate from intact S/G(s) using steam dump to atmosphere valve(s).
	b. Check S/G pressures - LESS THAN 80 PSIG	b. Perform one of the following: * **IF** S/G pressure decreasing, **THEN** observe CAUTION prior to Step 10 **AND** return to Step 10. * **IF** S/G pressure **NOT** decreasing, **THEN** observe NOTE prior to Step 19 **AND** go to Step 19.
	c. Check RCS hot leg temperatures - AT LEAST TWO LESS THAN 340°F	c. Perform one of the following: * **IF** RCS hot leg temperatures decreasing, **THEN** observe CAUTION prior to Step 10 **AND** return to Step 10. * **IF** RCS hot leg temperatures **NOT** decreasing, **THEN** observe NOTE prior to Step 19 **AND** go to Step 19.
	d. Stop S/G depressurization	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Normal conditions are desired, but NOT required for starting RCPs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	**Check If RCPs Should Be Started**	
	a. Core Exit TCs - GREATER THAN 1200°F	a. Go to Step 20
	b. Check if an idle RCS cooling loop is available • Narrow range S/G level - GREATER THAN 6% [32%] • RCP in associated loop - AVAILABLE **AND NOT** OPERATING	b. Perform the following: 1) Open all PRZ PORVs and block valves. 2) IF core exit TCs remain greater than 1200°F AND all PRZ PORVs and block valves open, THEN perform the following: a) Verify fuses installed for all RCS vent valves: • 3101 for SV-3-6318A • 3102 for SV-3-6318B • 3103 for SV-3-6319A • 3104 for SV-3-6319B • 3105 for SV-3-6612 • 3106 for SV-3-6611 b) Open all other RCS vent paths to containment: • SV-3-6318A • SV-3-6318B • SV-3-6319A • SV-3-6319B • SV-3-6611 • SV-3-6612 c) Go to Step 20.
	c. Start RCP in one idle RCS cooling loop	
	d. Return to Step 19a	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9	**Check If One RCP Should Be Stopped**	
	a. All RCPs - RUNNING	a. Go to Step 11.
	b. Stop RCP in loop B	
	c. Go to Step 11	
10	**Check Core Cooling**	
	a. Core exit TCs - LESS THAN 700°F	a. Perform one of the following: * **<u>IF</u>** temperature is decreasing, **<u>THEN</u>** observe NOTE prior to Step 1 **<u>AND</u>** return to Step 1. * **<u>IF</u>** temperature is **<u>NOT</u>** decreasing, **<u>THEN</u>** go to Step 11.
	b. Return to procedure **<u>AND</u>** step in effect	
11	**Check SI Accumulator Isolation Valve Status**	
	a. Power to Accumulator Discharge MOVs - AVAILABLE	a. Locally unlock and close the following breakers: • 30532 for MOV-3-865A • 30631 for MOV-3-865B • 30733 for MOV-3-865C
	b. Accumulator Discharge MOVs - OPEN • MOV-3-865A • MOV-3-865B • MOV-3-865C	b. **<u>IF</u>** accumulators are **<u>NOT</u>** discharged, **<u>THEN</u>** open Accumulator Discharge MOVs.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- ***If CST level decreases to less than 10%, makeup water sources for CST will be necessary to maintain secondary heat sink.***
- ***A faulted or ruptured S/G should NOT be used in subsequent steps unless no intact S/G is available.***

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12	**Maintain Intact S/G Levels**	
	a. Narrow range level – GREATER THAN 6%[32%]	a. Increase total feed flow to restore narrow range level greater than 6%[32%] in at least one S/G.
	b. Control feed flow to maintain narrow range level between 15%[32%] and 50%	
	c. Narrow range level - LESS THAN 50%	c. Stop feed flow to any S/G with narrow range level greater than 50%.

CAUTION

The following step will cause accumulator injection which may cause a red path condition on the F-0.4, INTEGRITY Status Tree. This procedure should be completed before transitioning to 3-EOP-FR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	**Depressurize All Intact S/Gs To 80 Psig**	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR	
	b. Dump steam to condenser	b. Manually dump steam from intact S/G(s) using steam dump to atmosphere valve(s).
	c. Check S/G pressures - LESS THAN 80 PSIG	c. Observe CAUTIONS prior to step 12 **AND** return to Step 12.
	d. Stop S/G depressurization	

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	011	2.4.9
Importance Rating		4.2

(Large Break LOCA) Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: SRO Question # 77

Unit 3 is in drain down for Reduced Inventory Operations with the following:

- The RCS is depressurized at 100°F.
- RCS Heatup Rate is 9°F/ hr.
- S/G 3A, 3B and 3C Wide Range Levels are at 20%.
- The following alarms are received in the Control Room:
  - H 6/2, RHR HX HI/LO FLOW
  - I 8/6, RHR SUMP PUMP ROOM A HI LEVEL.
- PZR Cold Cal Level, LI-3-462 is off-scale low.
- PZR Drain Down Levels, LI-3-6421 and LI-3-6423, are 10% and lowering quickly.
- RHR Pump 3A was tripped in 3-ONOP-050.

After the 3A RHR Pump was stopped, which ONE of the following describes (1) the required procedure for this event and (2) the mitigation strategy required?

REFERENCE PROVIDED

A. (1) Transition to 3-ONOP-041.8, Shutdown LOCA [Mode 5 or 6];
(2) Initiate RCS Feed and Bleed cooling using High Head SI and PZR PORVs.

B. (1) Transition to 3-ONOP-041.8, Shutdown LOCA [Mode 5 or 6];
(2) Vent the RHR system and start one RHR Pump to re-establish RHR cooling.

C. (1) Continue in 3-ONOP-050, Loss of RHR;
(2) Initiate RCS Feed and Bleed cooling using High Head SI and PZR PORVs.

D. (1) Continue in 3-ONOP-050, Loss of RHR;
(2) Vent the RHR system and start one RHR Pump to re-establish RHR cooling.

Proposed Answer: A

Explanation (Optional):

A. CORRECT. After RHR Pump 3A was tripped in 3-ONOP-050, the crew transitions to 3-ONOP-041.8 due to low RCS level. They perform Attachment 2 to initiate Feed and Bleed RCS cooling.

B. Incorrect because this is the wrong action. Plausible because correct procedure is referenced, because venting the RHR system is also performed in this procedure when adequate RCS level is present. Also, this action is plausible because there is indication of RHR pump cavitation.

C. Incorrect since this procedure is an improper mitigation strategy for a Shutdown LOCA when in MODE 5 or 6. Plausible because the actions listed are correct for this set of conditions.

D. Incorrect since this procedure is an improper mitigation strategy for a Shutdown LOCA when in MODE 5 or 6. Plausible because venting RHR is required after a cavitation condition. However, with the reduced inventory status of RCS level, 3-ONOP-041.8 cools the RCS by use of Feed and Bleed.

Technical Reference(s): 3-ONOP-041.8, Shutdown LOCA [Mode 5 or 6] (Attach if not previously provided)
3-ONOP-050, Loss of RHR

Proposed References to be provided to applicants during examination: None

Learning Objective: None (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests knowledge of the parameters (RCS inventory loss) used to assess the status of safety functions, such as core cooling and heat removal, during a Loss of RHR.

SRO ONLY Justification:

From SRO Only guidance:

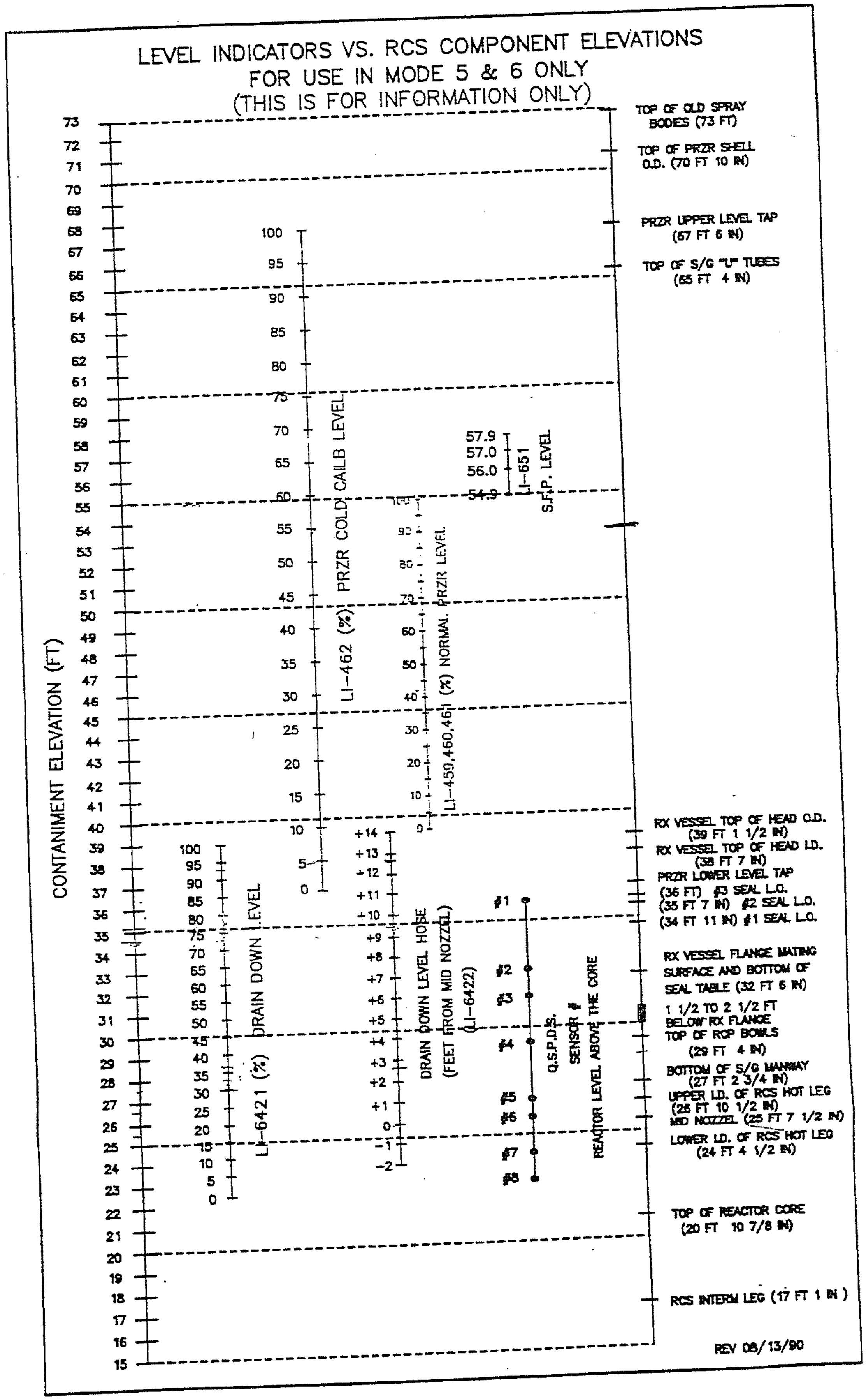
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

- 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 emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Section V
Figure 9



QUESTION 77

Question # 77

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

AOP Based question not relying on knowledge of system operation

No ↓

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

Yes → RO question

Item does not test Immediate Actions. This is a selection between 2 plausible procedures and understanding of mitigation strategy based on plant conditions

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

Requires knowledge of AOP entry conditions, but based upon 2 plausible choices, and requires additional knowledge beyond entry conditions

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

Question tests a discrete set of actions required based on plant conditions and understanding which procedure is entered

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

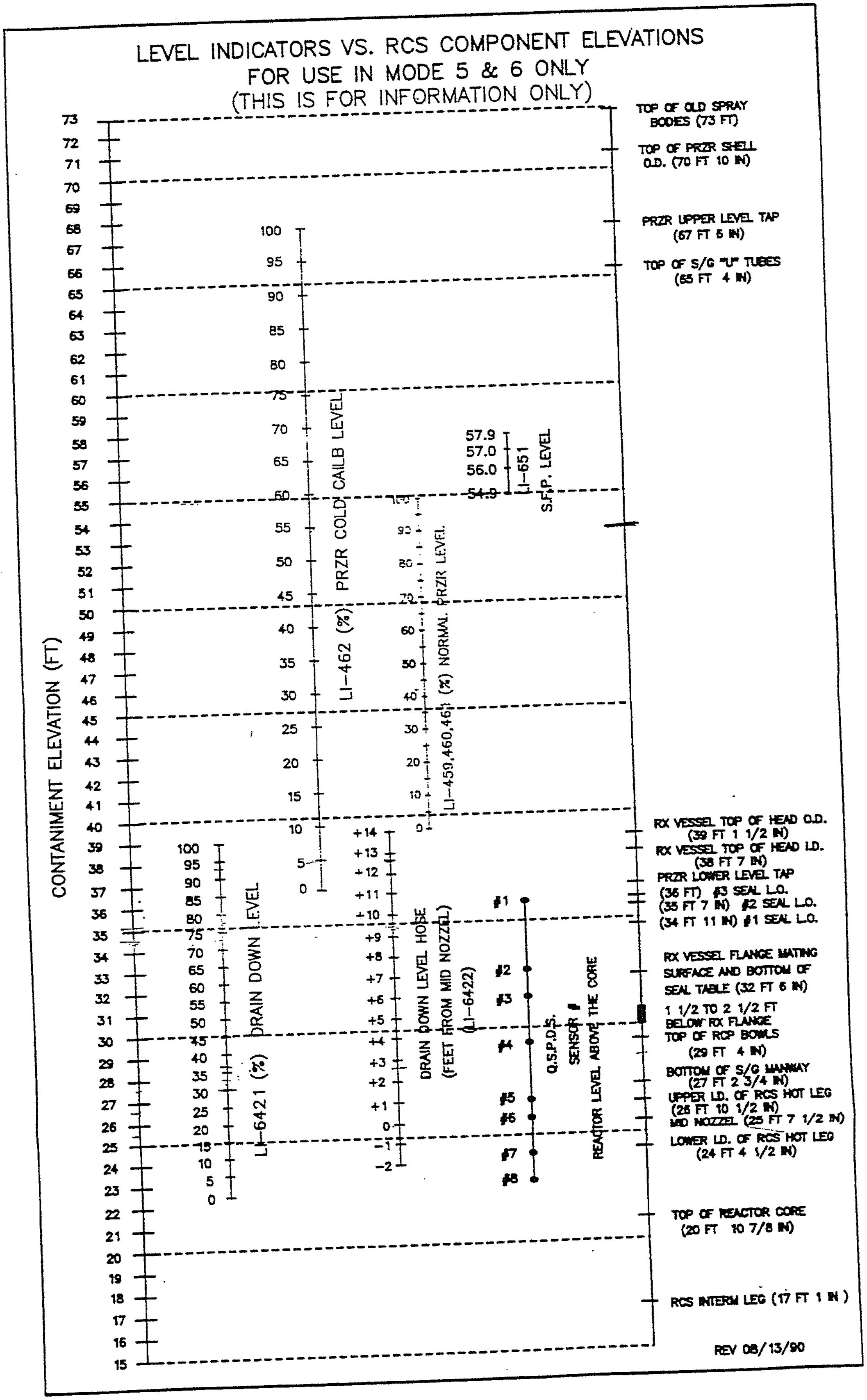
YES

Yes → SRO-only question

No ↓

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

Section V
Figure 9



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

If leakage from the RHR system is discovered, the leak should be isolated using 3-ONOP-041.3, EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE.

NOTES

- *Oscillations in flow or motor amps may be indicative of RHR pump cavitation.*
- *If loss of RHR is due to a loss of off-site power capability, power and RHR flow should be restored utilizing 3-ONOP-004, LOSS OF OFFSITE POWER or 3-EOP-ECA-0.0, LOSS OF ALL AC. During a loss of power, this procedure should be used to establish containment closure and alternate cooling if RHR flow remains unavailable.*
- *The foldout page shall be monitored during the performance of this procedure.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. RCS level - GREATER THAN 10% PRESSURIZER COLD CAL	a. **IF** RCS Draindown Level Instrumentation is not available or RCS draindown level is LESS than 23%, **THEN** stop the running RHR pump **AND** go to 3-ONOP-041.8, Shutdown LOCA (Mode 5 or 6).
	b. RHR pumps - ANY RUNNING	b. Go to Step 2.
	c. RHR pumps - NOT CAVITATING • Amps Stable at normal value • Flow Stable at normal value	c. Stop RHR pumps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- *Changes in RCS pressure may result in inaccuracies in RCS level readings.*
- *If the refueling Cavity is flooded, then go to 3-ONOP-033.2, REFUELING CAVITY SEAL FAILURE.*
- *If entering this procedure from 3-ONOP-050, LOSS OF RHR, then go to Step 21.*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check If RHR Pumps Should Be Stopped**	
	a. RHR pumps - ANY RUNNING	a. Go to Step 2.
	b. RCS LEVEL - ADEQUATE FOR PLANT CONDITIONS • Drain Down Level 1) LI-3-6421 - GREATER THAN 23% 2) LI-3-6423 - GREATER THAN 23% **OR** • Pressurizer Level, LI-3-462 - GREATER THAN 10%	b. Perform the following: 1) Stop both RHR pumps **AND** place them in standby. 2) Go to Step 2.
	c. Check RCS Level - STABLE **OR** INCREASING	c. Perform the following: 1) Maintain RCS inventory using the following methods while continuing with this procedure: a) Charging flow (Step 13). b) RWST Gravity Feed (Step 14). c) VCT Overpressure Feed (Step 15).
	d. RHR flow - LESS THAN 3000 GPM	d. Reduce RHR flow to 3000 gpm.
	e. RHR pumps - CAVITATING	e. Perform the following: 1) **IF** level stable or increasing, **THEN** go to appropriate plant procedure as determined by the Shift Manager. 2) **IF** level decreasing, **THEN** go to Step 1f.
	f. Stop both RHR pumps and place them in standby.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Establish Conditions To Start RHR Pump**	
	a. Verify RCS Level * LI-3-6421/6423 - GREATER THAN OR EQUAL TO 23% **OR** * Pressurizer Level, LI-3-462, GREATER THAN **OR** EQUAL TO 10%	a. Perform the following: 1) Monitor RCS heatup rate. 2) Feed and Bleed cooling using ATTACHMENT 2. 3) Return to Step 6.
	b. RHR pump - AVAILABLE	b. Perform the following: 1) Monitor RCS heatup rate 2) Feed and Bleed cooling using ATTACHMENT 2. 3) Return to Step 6.
	c. Verify RHR System Valves In Proper Alignment As Follows • MOV-3-750 - OPEN • MOV-3-751 - OPEN • MOV-3-744A - OPEN • MOV-3-744B - OPEN • MOV-3-749A - OPEN • MOV-3-749B - OPEN • MOV-3-862A - CLOSED • MOV-3-862B - CLOSED • MOV-3-863A - CLOSED • MOV-3-863B - CLOSED	c. Manually align valves to establish at least one train of RHR flow.
	d. Verify CCW cooling to RHR System - IN SERVICE	d. Restore CCW cooling to the RHR heat exchangers using Step 8a of this procedure.
	e. Check core exit TCs - LESS THAN 200°F	e. Initiate makeup to the RCS using one High-head SI pump.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ATTACHMENT 2
(Page 1 of 3)

FEED AND BLEED COOLING

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Check Secondary Heat Sink NOT AVAILABLE**	Return to step in effect.
2	**Verify RCS FEED Path**	Manually Start pumps and align valves to establish one train of safety injection equipment for an RCS Feed Path. **IF** a RCS feed path can **NOT** be established, **THEN** return to step in effect.
	a. One train of SI valves aligned for injection.	
	b. High-head SI pumps – AT LEAST ONE RUNNING	
3	**Verify Containment Isolation Phase A White Lights On VPB – ALL BRIGHT**	**IF** any containment isolation phase A valve is **NOT** closed, **THEN** manually close valve. **IF** valve(s) can **NOT** be manually closed, **THEN** manually **OR** locally isolate affected penetration.
4	**Establish RCS Bleed Path**	
	a. Verify power to PZR PORV Block valves – AVAILABLE	a. Restore power to block valves.
	b. Open both PZR block valves	
	c. Open both PZR PORVs	

QUESTION 78

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	025	2.4.21
Importance Rating		4.6

(Loss of RHR) Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: SRO Question # 78

A Unit 3 RCS cooldown is in progress with the following:

- Unit 3 was shutdown 5 days ago.
- RCS temperature is 105°F.
- RCS Pressure is150 psig.
- RHR Pump 3A is in service.
- RHR Pump 3B is in Standby.
- CCW Pump 3C is out of service.
- S/G levels are 35% Narrow Range on all three S/Gs.

Then,

- RHR Pump 3A trips.
- RHR Pump 3B is started.

In accordance with ADM-051, Outage Risk Assessment and Control, which ONE of the following determines (1) the required Enclosure for Contingency Actions to mitigate a Loss of Decay Heat Removal Function and (2) the classification of the Safe Shutdown Function Color Code?

A. (1) **Enclosure 1**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees with RCS Loops Filled
(2) Orange

B. (1) **Enclosure 2**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees without RCS Loops Available
(2) Red

C. (1) **Enclosure 1**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees with RCS Loops Filled
(2) Red

D. (1) **Enclosure 2**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees without RCS Loops Available
(2) Orange

Proposed Answer: A

Explanation (Optional):

A. CORRECT. RCS Loops are filled and available since RCS Pressure is at 150 psig. Also, one RHR Pump is out of service which is a classification of the Safe Shutdown Function Color Code of Orange.

B. Incorrect because **Enclosure 2**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees without RCS Loops Available is the wrong enclosure. Plausible since RCS pressure is low and a loss of a RHR Pump is top priority for restoration which is assumed to be a Red Safe Shutdown Function Color Code.

C. Incorrect because a loss of a RHR Pump is an Orange Safe Shutdown Function Color Code. Plausible since RCS pressure is low and a loss of a RHR Pump is top priority for restoration which is assumed to be a Red Safe Shutdown Function Color Code.

D. Incorrect because **Enclosure 2**, Minimum Required Equipment, Phase I, Large Decay Heat Load and RCS Temp Less Than 200 Degrees without RCS Loops Available is the wrong enclosure. Plausible since a loss of a RHR Pump is top priority for restoration which is an Orange Safe Shutdown Function Color Code.

Technical Reference(s): ADM-051 Enclosure 1 & 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: ADM-051 Enclosure 1 & 2 (1st page of each)

Learning Objective: None (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPK)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests knowledge of the parameters (RCS inventory loss) used to assess the status of safety functions, such as core cooling and heat removal, during a Loss of RHR.

SRO ONLY Justification:

From SRO Only guidance:

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

- 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 emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.
- Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

Question # 78

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 ↓

Question is not testing system knowledge. Shutdown Risk is topic

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

Yes → RO question

No ↓

Item does not test Immediate Actions. This item requires understanding of conditions that result in a degradation of defense in depth

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 ↓

Item does not test entry to AOPs

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 ↓

Item tests understanding of level of degradation of the plant during shutdown conditions and strategy for restoring Ctmt integrity during heatup

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

YES. SD Risk procedure enclosures provide guidance for strategy on maintaining SD safety functions during the abnormal event

No ↓

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

ENCLOSURE 1
(Page 1 of 9)

MINIMUM REQUIRED EQUIPMENT, PHASE I, LARGE DECAY HEAT LOAD AND RCS TEMP LESS THAN 200 DEGREES WITH RCS LOOPS FILLED

Function	Required Equipment	Contingency Action	Color Code
Decay Heat Removal	Two RHR Pumps	1. Take immediate action to repair the failed pump. 2. Go to 3/4-ONOP-050, LOSS OF RHR, for loss of RHR. 3. Maximize RCS inventory as much as is achievable. 4. Maintain RCS pressurized to 100 psig to meet loops filled criteria. 5. Maintain RCS temperature as low as possible. 6. Ensure Feed and Bleed Decay Heat Removal capability, using the HHSI pumps, is available. 7. Maintain Decay Heat Removal capability with at least two Steam Generators, including a source of feedwater. 8. Suspend any activities that would risk the remaining RHR, ICW, or CCW Pumps; EDG; and 4KV Bus. 9. Upon loss of both pumps, initiate actions to establish Containment Closure except for the equipment hatch. 10. Verify that the equipment hatch can be closed within the time frame to heat up to Mode 4 following a loss of shutdown cooling as predicted using the heatup rates of Figure 1. Station the necessary personnel at the hatch with direct communications to the RO.	1 – ORANGE 0 - RED
	Two RHR Heat Exchangers	1. Take action as described for RHR Pumps.	1 – ORANGE 0 - RED
	Two CCW Pumps	1. Go to 3/4-ONOP-030, COMPONENT COOLING WATER MALFUNCTION, for loss of CCW. 2. Carry out other actions per RHR pumps above.	1 – ORANGE 0 - RED

ENCLOSURE 2
(Page 1 of 9)

MINIMUM REQUIRED EQUIPMENT, PHASE I,
LARGE DECAY HEAT LOAD AND RCS TEMP LESS
THAN 200 DEGREES WITHOUT RCS LOOPS AVAILABLE

Function	Required Equipment	Contingency Action	Color Code
Decay Heat Removal	Two RHR Pumps	1. Take immediate action to repair the failed pump. 2. Go to 3/4-ONOP-050, LOSS OF RHR, for loss of RHR. 3. Establish or maintain Reactor Vessel level higher than three feet below the vessel flange. Maximize RCS inventory as much as is achievable. 4. Maintain RCS temperature as low as possible. 5. Ensure Feed and Bleed decay heat removal capability, using the HHSI pumps, is available. 6. Suspend any activities that would risk the remaining RHR, ICW, or CCW Pumps; EDG; **AND** 4KV Bus. 7. Upon loss of both pumps, initiate actions to establish Containment Closure except for the equipment hatch. 8. Verify that the equipment hatch can be closed within the time frame to heat up to Mode 4 following a loss of shutdown cooling as time to saturation predicted using the Figure 2. Station the necessary personnel at the hatch with direct communications to the RO. 9. Investigate the possibility of flooding the Reactor Cavity to a height of 23 feet above the Reactor Vessel Flange. 10. If RCS inventory is at Mid-Loop, then refer to Figure 5, Time to Core Uncovery from Mid-Loop.	1 – ORANGE 0 - RED
	Two RHR Heat Exchangers	1. Take action as described for RHR Pumps.	1 Hx – ORANGE 0 Hx – RED
	Two CCW Pumps*	1. Go to 3/4-ONOP-030, COMPONENT COOLING WATER MALFUNCTION, for loss of CCW. 2. Carry out other actions as described per RHR pumps above.	1 – ORANGE 0 - RED

* Powered from independent power sources if RCS level lower than 3 feet below the vessel flange. [Commitment Step – 2.3.1]

QUESTION 79

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	056	AA2.53
Importance Rating		3.2

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Status of emergency bus under voltage relays

Proposed Question: SRO Question # 79

Which ONE of the following completes the statements below?

A failure of ONE 3A 4KV Bus Loss of Voltage Relay (ESFAS) is verified by ___(1)___.

In accordance with TS Table 4.3-2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements, the Loss of Voltage Relays are tested ___(2)___ to verify OPERABILITY for a Loss of Offsite Power.

NOTE: This test is satisfied with 3-OSP-203.1, Train A Safeguards Test.

A. (1) checking the amber light (PL-11) at Sequencer Panel 3C23A is OFF.
(2) at least once per 18 months (R)

B. (1) checking Annunciator X2/1, 4KV BUS 3A LO VOLTAGE, is LIT.
(2) at least once per 18 months (R)

C. (1) checking the amber light (PL-11) at Sequencer Panel 3C23A is OFF.
(2) at least once per 92 days (Q)

D. (1) checking Annunciator X2/1, 4KV BUS 3A LO VOLTAGE, is LIT.
(2) at least once per 92 days (Q)

Proposed Answer: A

Explanation (Optional):

A. CORRECT. RO Daily Logs, 3-OSP-201.1, check the amber light (PL-11) at Sequencer Panel 3C23A is ON to satisfy TS 3.3.2 FU 7.a. Also, 0-ADM-218, Technical Specification Matrix, lists 3-OSP-203.1 as a refueling surveillance (18 month) along with TS Table 4.3-2.

B. Incorrect because Annunciator X2/1, 4KV BUS 3A LO VOLTAGE, setpoint at 3325 VAC is not use indicative of a bus undervoltage (2975 VAC). Plausible since the LO Voltage relay gives an alarm associated with the bus. Also, the surveillance test is conducted on an "R" frequency (18 months) per TS Table 4.3-2.

C. Incorrect because the surveillance test is conducted on an "R" frequency (18 months) per TS Table 4.3-2. Plausible because the first part is correct.

D. Incorrect because Annunciator X2/1, 4KV BUS 3A LO VOLTAGE, setpoint at 3325 VAC is not use indicative of a bus undervoltage (2975 VAC). Incorrect because the surveillance test is conducted on an "R" frequency (18 months) per TS Table 4.3-2. Plausible since the LO Voltage relay gives an alarm associated with the bus. Also, the surveillance test conducted on an "Q" frequency (3 months) per TS Table 4.3-2 is an acceptable frequency for other testing.

Technical Reference(s): TS Table 4.3-2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements (Attach if not previously provided)
0-ADM-218
3-OSP-201.1, pg. 31
3-ARP-097.CR.X, X2/1
5613-E-3 Sheet 11

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902157, Obj. 11 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)
Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

K/A Match Justification:

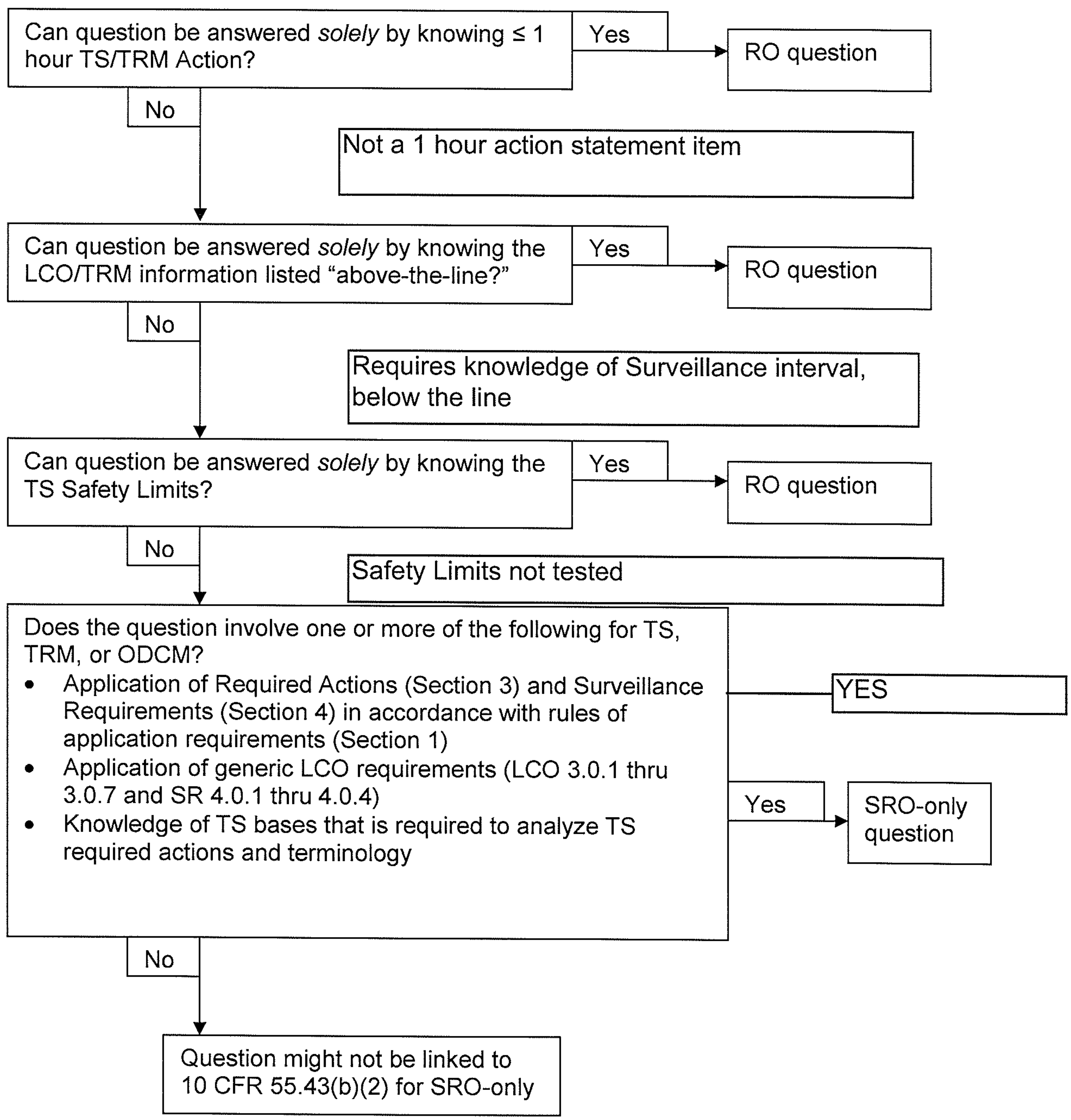
This question matches the K/A in that it tests how to check the status of the Loss of Voltage relays that are used to sense a Loss of Offsite Power, including the surveillance requirements that check the operability of those relays.

SRO Only Justification:

This question is SRO Only because it satisfies the guidance in the "SRO Only document," Page 3, as highlighted below:

B. Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]
Some examples of SRO exam items for this topic include:
• Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).

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



ATTACHMENT 1
(Page 15 of 16)

MINIMUM INSTRUMENTATION AND EQUIPMENT LIST - MODE 1

UNIT 3 MODE 1 REACTOR PROTECTION STATUS LIGHT CHECK

ESF INSTRUMENTATION BISTABLES (Cont'd)									
INSTRUMENT/BISTABLE			**INSTR NO.**		**MIDS**	**DAYS**	**PEAKS**	**TECH SPECS**	**NOTES**
SG PRESS	LO	LOOP A LO STM PRESS	PC474A	OFF				3.3.2 Func. Unit 1.f	
			PC485A	OFF				3.3.2 Func. Unit 1.f	
			PC496A	OFF				3.3.2 Func. Unit 1.f	
CONTAINMENT PRESS	HI	HI CONMT PRESS (RED)		OFF				3.3.2 Func. Unit 1.c, 2.b, 3.b.3), 4.c	
		HI CONMT PRESS (WHITE)		OFF				3.3.2 Func. Unit 1.c, 2.b, 3.b.3), 4.c	
		HI CONMT PRESS (BLUE)		OFF				3.3.2 Func. Unit 1.c, 2.b, 3.b.3), 4.c	
	HI-HI	HI-HI CONMT PRESS (RED)		OFF				3.3.2 Func. Unit 2.b, 3.b.3), 4.c	
		HI-HI CONMT PRESS (WHITE)		OFF				3.3.2 Func. Unit 2.b, 3.b.3), 4.c	
		HI-HI CONMT PRESS (BLUE)		OFF				3.3.2 Func. Unit 2.b, 3.b.3), 4.c	
SG HI LEVEL	A	SG A HI LEVEL	LC474-1	OFF				3.3.2 Func. Unit 5.c	Steam Generator overfill protection is not part of the Engineered Safety Features Actuation System (ESFAS) and is added to the Technical Specifications only in accordance with NRC Generic Letter 89-19.
			LC475-1	OFF				3.3.2 Func. Unit 5.c	
			LC476-1	OFF				3.3.2 Func. Unit 5.c	
	B	SG B HI LEVEL	LC484-1	OFF				3.3.2 Func. Unit 5.c	
			LC485-1	OFF				3.3.2 Func. Unit 5.c	
			LC486-1	OFF				3.3.2 Func. Unit 5.c	
	C	SG C HI LEVEL	LC494-1	OFF				3.3.2 Func. Unit 5.c	
			LC495-1	OFF				3.3.2 Func. Unit 5.c	
			LC496-1	OFF				3.3.2 Func. Unit 5.c	
SEQUENCER UNDERVOLTAGE RELAY	3A 4KV BUS		127-3A1	See Note				3.3.2 Func. Unit 7.a	PL-11 amber light ON at 3C23A
			127-3A2	See Note				3.3.2 Func. Unit 7.a	PL-12 amber light ON at 3C23A
	3B 4KV BUS		127-3B1	See Note				3.3.2 Func. Unit 7.a	PL-11 amber light ON at 3C23B
			127-3B2	See Note				3.3.2 Func. Unit 7.a	PL-12 amber light ON at 3C23B

TABLE 3.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater### (Continued)					
b. Stm. Gen. Water Level--Low-Low	3/steam generator	2/steam generator in any steam generator	2/steam generator	1, 2, 3	15
c. Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. Bus Stripping	1/bus	1/bus	1/bus	1, 2, 3	23
e. Trip of all Main Feed-water Pumps Breakers	1/breaker	(1/breaker) /operating pump	(1/breaker) /operating pump	1, 2	23
7. Loss of Power					
a. 4.16 kV Busses A and B (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4	18
b. 480 V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	2 per load center	2 on any load center	2 per load center	1, 2, 3, 4	18
Coincident with: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)						
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
d. Bus Stripping	N.A.	R	N.A.	R	N.A.	1, 2, 3
e. Trip of All Main Feedwater Pump Breakers.	N.A.	N.A.	N.A.	R	N.A.	1. 2
7. Loss of Power						
a. 4.16 kV busses A and B (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	1, 2, 3, 4
b. 480V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Undervoltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. 480V Load Centers 3A, 3B, 3C, 3D and 4A, 4B, 4C, 4D Degraded Voltage	S	R	N.A.	M(1)	N.A.	1, 2, 3, 4

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

ENCLOSURE 2

TECHNICAL SPECIFICATION MATRIX

Tech Spec	Table	Item	Fre-quency	Mode	Implementing Document	Resp Dept	Scheduling Document	Remarks
4.3.1.1	4.3-1	17.c	R	1	4-PMI-059.12	IC	0-ADM-215	
4.3.1.1	4.3-1	17.c	R	1	4-PMI-059.13	IC	0-ADM-215	
4.3.1.1	4.3-1	17.c	R	1	4-PMI-059.14	IC	0-ADM-215	
4.3.1.1	4.3-1	17.c	R	1	4-PMI-059.15	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	3-OSP-059.4	OPS	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	3-PMI-059.12	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	3-PMI-059.13	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	3-PMI-059.14	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	3-PMI-059.15	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	4-OSP-059.4	OPS	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	4-PMI-059.12	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	4-PMI-059.13	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	4-PMI-059.14	IC	0-ADM-215	
4.3.1.1	4.3-1	17.d	R	12	4-PMI-059.15	IC	0-ADM-215	
4.3.1.1	4.3-1	18	R	1	QTI-5-PS/PTN-2.01	SPS	0-ADM-215	
4.3.1.1	4.3-1	18	R	1	3-OSP-203.1	OPS	0-ADM-215	
4.3.1.1	4.3-1	18	R	1	3-OSP-203.2	OPS	0-ADM-215	
4.3.1.1	4.3-1	18	R	1	4-OSP-203.1	OPS	0-ADM-215	
4.3.1.1	4.3-1	18	R	1	4-OSP-203.2	OPS	0-ADM-215	
4.3.1.1	4.3-1	19	62D*	12-345**	3-OSP-049.1	OPS	0-ADM-215	* On a staggered basis. **With Rx trip bkrs closed & capable of withdrawal
4.3.1.1	4.3-1	19	62D*	12-345**	4-OSP-049.1	OPS	0-ADM-215	* On a staggered basis. **With Rx trip bkrs closed & capable of withdrawal
4.3.1.1	4.3-1	20	62D*	12-345**	3-OSP-049.1	OPS	0-ADM-215	* On a staggered basis. **With Rx trip bkrs closed & capable of withdrawal
4.3.1.1	4.3-1	20	62D*	12-345**	4-OSP-049.1	OPS	0-ADM-215	* On a staggered basis. **With Rx trip bkrs closed & capable of withdrawal

CAUSES:
1. Voltage regulator failure
2. Peak system power demand with low system voltage

X2/1

4KV BUS 3A
LO VOLTAGE

DEVICE: UV Relay 127/3A

SETPOINT: 3325 VAC

LOCATION: Panel 3C11T

ALARM CONFIRMATION

1. **CHECK** the following:
   - 3A 4KV Bus Indicator on console
   - 240KV Line Voltage on VPA
   - System frequency
   - Megavar load on Unit 3 Generator

OPERATOR ACTIONS

1. **NOTIFY** system of need to raise bus voltage.
2. **ADJUST** Megavar load within tolerances of Plant Curve Book and system generation requirements.
3. IF 4KV Bus voltage can **NOT** be stabilized, THEN **SHUTDOWN** unit using 3-GOP-103, Power Operation to Hot Standby, or 3-ONOP-100, Fast Load Reduction.

REFERENCES:
1. FPL EWD 5613-E-28, Sh 11A
2. FPL Plant Curve Book

QUESTION 80

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	038	2.2.44
	Importance Rating		4.4

Equipment Control: 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.

Proposed Question: SRO Question # 80

Given the following:

- Unit 4 experienced a Steam Generator Tube Rupture (SGTR) from 100% power.
- Containment temperature on TE-4-6700, TE-4-6701, and TE-4-6702 is 135°F and rising.
- The operating crew is implementing 4-EOP-E-3, Steam Generator Tube Rupture.
- The crew stopped the RCS cooldown and verified the ruptured S/G pressure is increasing slowly.
- The current RCS Subcooling is 70°F.

Which ONE of the choices below completes the following statements?

The required RCS Subcooling to stay in 4-EOP-E-3, Steam Generator Tube Rupture is __(1)__ .

If below the required RCS Subcooling for 4-EOP-E-3, then transition to __(2)__ for RCS cooldown and depressurization.

A. (1) MET
(2) 4-EOP-ECA-3.1, SGTR with Loss of Reactor Coolant, Subcooled Recovery Desired

B. (1) NOT MET
(2) 4-EOP-ECA-3.1, SGTR with Loss of Reactor Coolant, Subcooled Recovery Desired

C. (1) MET
(2) 4-EOP-ECA-3.2, SGTR with Loss of Reactor Coolant; Saturated Recovery Desired

D. (1) NOT MET
(2) 4-EOP- ECA-3.2, SGTR with Loss of Reactor Coolant; Saturated Recovery Desired

Proposed Answer: A

Explanation (Optional):

A. CORRECT.
4-EOP-E-3, Step 21
Check RCS Subcooling Based On Core Exit TCs - GREATER THAN 50°F[230°F]
Go to 4-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, Step 1.

B. Incorrect since the minimum subcooling under these conditions is 50°F, not 230°F. Plausible because the 2nd part is correct. Also plausible because the 1st part would be correct if adverse containment conditions existed.

C. Incorrect since the proper transition is to 4-EOP-ECA-3.1, not 4-EOP-ECA-3.2. Plausible because the 1st part is correct. Also plausible because with 4-EOP-ECA-3.2, SGTR with Loss of Reactor Coolant Saturated Recovery Desired, the applicant using Adverse Cotainment values would deduce since RCS subcooling is inadequate, then a transition to a procedure with "saturated recovery" in the title is desired

D. Incorrect since the minimum subcooling under these conditions is 50°F, not 230°F. Also incorrect since the proper transition is to 4-EOP-ECA-3.1, not 4-EOP-ECA-3.2. Also plausible because with 4-EOP-ECA-3.2, SGTR with Loss of Reactor Coolant Saturated Recovery Desired, the applicant could deduce since RCS subcooling is inadequate, then the transition to a procedure with "saturated recovery" in the title is desired. Also plausible because the 1st part would be correct if adverse containment conditions existed.

Technical Reference(s): LP 6902339, Steam Generator Tube Rupture

4-EOP-E-3, Steam Generator Tube Rupture

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902339, Obj. 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests ability to monitor CR indications (subcooling) to verify the status of a system (RCS) and understanding of how actions (go to 4-EOP-ECA-3.1) affect the plant.

SRO ONLY Justification:

From SRO Only guidance:

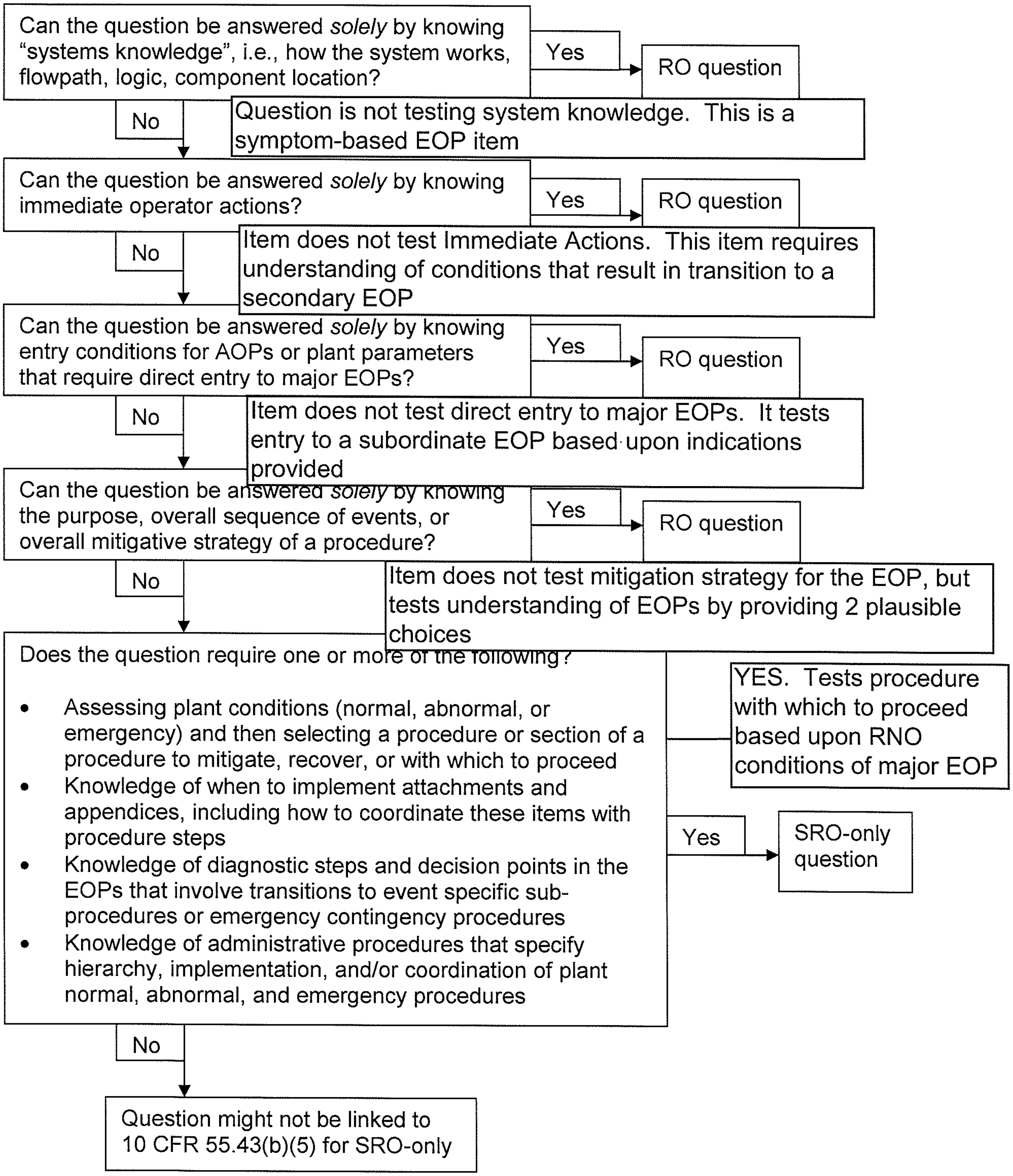
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. [10 CFR 55.43(b)(5)]

This 10 CFR 55.43 topic involves both 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

The applicant's knowledge can be evaluated at the level of 10 CFR 55.43(b)(5) by ensuring that the additional knowledge of the procedure's content is required to correctly answer the written test item, for example:

- 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 emergency operating procedures (EOP) 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.

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



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
21	**Check RCS Subcooling Based On Core Exit TCs - GREATER THAN 50°F[230°F]**	Go to 4-EOP-ECA-3.1, SGTR WITH LOSS OF REACTOR COOLANT-SUBCOOLED RECOVERY DESIRED, Step 1.
22	**Depressurize RCS To Minimize Break Flow And Refill PRZ**	
	a. Normal PRZ spray – AVAILABLE	a. Observe CAUTIONS and NOTE prior to Step 23 **AND** go to Step 23.
	b. Spray PRZ with maximum available spray until any of the following conditions satisfied – Use ATTACHMENT 6 as reference * Both of the following 1) RCS pressure - LESS THAN RUPTURED S/G(s) PRESSURE 2) PRZ level - GREATER THAN 17%[50%] **OR** * PRZ level - GREATER THAN 71%[50%] **OR** * RCS subcooling based on core exit TCs - LESS THAN 30°F[210°F]	
	c. Stop depressurization by closing spray valve(s): * Close normal spray valves **OR** * Close Auxiliary Spray Valve, CV-4-311	* Stop RCP(s) as necessary to stop spray flow. * Perform the following: a) Reduce charging pump speed to minimum. b) Close Charging Flow To Regen Heat Exchanger, HCV-4-121. c) Adjust charging pump speed to maintain seal injection flow.
	d. Observe CAUTION prior to Step 25 **AND** go to Step 25	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
50	**Go To Appropriate Post–SGTR Cooldown Method** * Go to 4-EOP-ES-3.1, POST-SGTR COOLDOWN USING BACKFILL, Step 1 **<u>OR</u>** * Go to 4-EOP-ES-3.2, POST-SGTR COOLDOWN USING BLOWDOWN, Step 1 **<u>OR</u>** * Go to 4-EOP-ES-3.3, POST-SGTR COOLDOWN USING STEAM DUMP, Step 1	

END OF TEXT

QUESTION 81

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	E04	EA2.1
Importance Rating		4.3

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: SRO Question # 81

The following conditions exist:

3-EOP-ECA-1.2, LOCA Outside Containment is in progress. MOV-3-744A/B, RHR Discharge to Cold Leg Isolation Valves, have been closed.

- RCS pressure indicates 1540 psig and rising
- PZR level is 20% and rising

Which ONE of the following provides the required mitigation actions for the above plant conditions?

A. 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, to minimize HHSI flow

B. 3-EOP-E-1, Loss of Reactor or Secondary Coolant, to align Charging to terminate Safety Injection

C. 3-EOP-ECA-1.1, Loss of Emergency Coolant Recirculation, to makeup to the RWST

D. 3-EOP-E-1, Loss of Reactor or Secondary Coolant, for post LOCA cooldown and depressurization

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Leak is isolated. Transition should be to 3-EOP-E-1 to align Charging to

terminate Safety Injection

B. CORRECT. The SRO should transition to 3-EOP-E-1 based on RCS pressure rising.

C. Incorrect. The strategy is to isolate the leak, verify isolation, and return to 3-EOP-E-1 for these conditions. Plausible because 3-EOP-ECA-1.1 is the correct procedure to transition to if RCS pressure is stable or lowering

D. Incorrect. The conditions are met for transition to 3-EOP-E-1. However, a post LOCA cooldown and depressurization is not necessary due isolation. Plausible because 3-EOP-EOP-E-1 is the correct procedure.

Technical Reference(s): 3-EOP-ECA-1.2
3-EOP-E-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6900333 Obj 6 (As available)

Question Source: Bank #

Modified Bank # WTSI 66716 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPK)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41

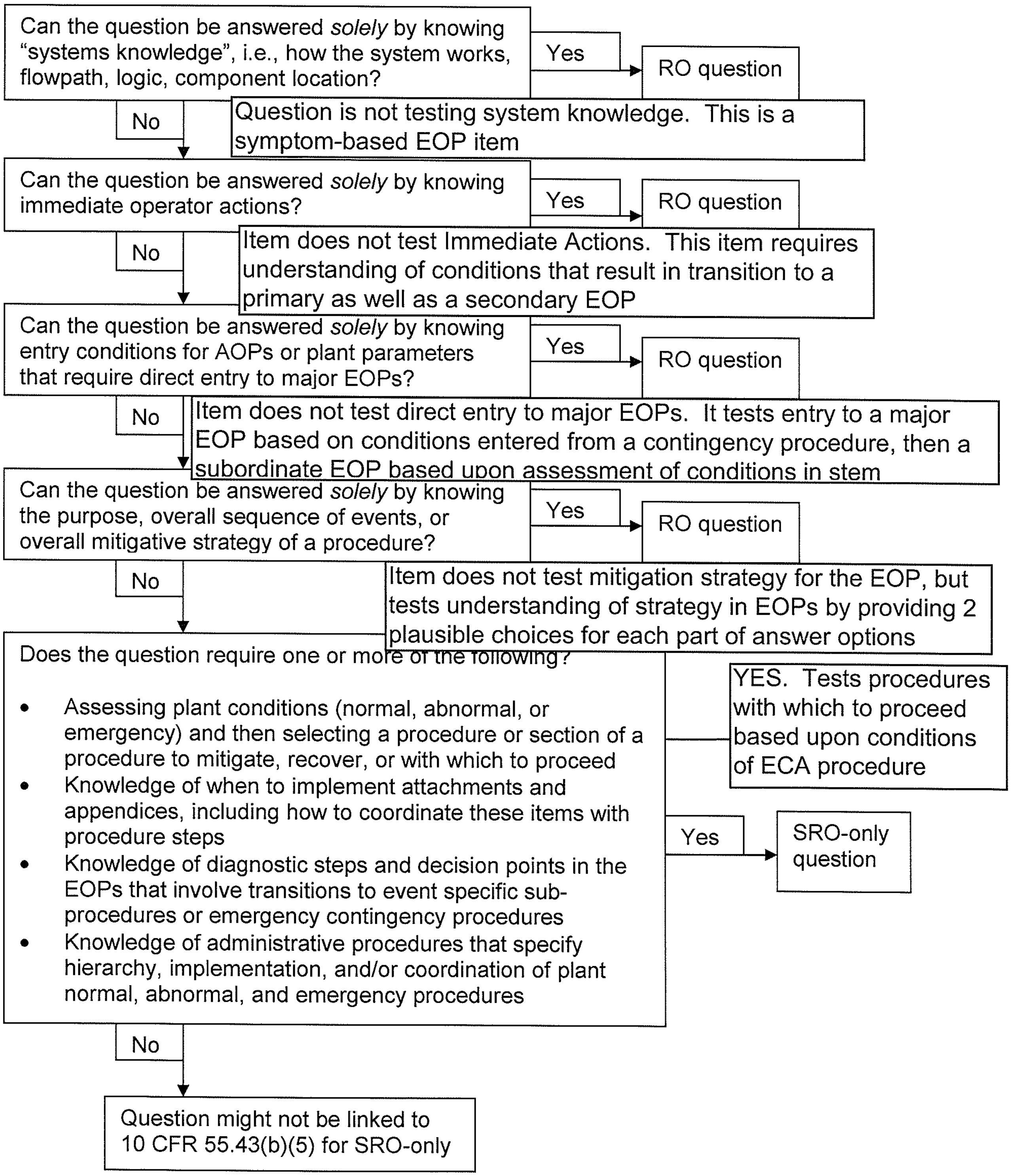
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Modified from 2009 Turkey Point Exam. Changed conditions in stem and changed correct answer. Modified other distracters for plausibility of conditions

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



STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	**Try To Identify <u>AND</u> Isolate Break**	
	a. Contact H.P. Department for survey of auxiliary building to determine source of high radiation	
	b. Close RHR Discharge To Cold Leg Isolation valves • MOV-3-744A • MOV-3-744B	
	c. RCS pressure - STABLE OR DECREASING	c. Go to Step 3.
	d. Open RHR Discharge To Cold Leg Isolation valves • MOV-3-744A • MOV-3-744B	
	e. Close SI To Cold Leg Isol Valves • MOV-3-843A • MOV-3-843B	e. Locally close valve(s).
	f. RCS pressure - STABLE OR DECREASING	f. Go to Step 3
	g. Open SI To Cold Leg Isol Valves • MOV-3-843A • MOV-3-843B	
3	**Check If Break Is Isolated**	
	a. RCS pressure - INCREASING	a. Go to 3-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to 3-EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1	

END OF TEXT

FINAL PAGE

WCO/daj/nw

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2009
Exam Type:	S

Original Item for Question 81

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #	1	1
	K/A #	E04	EA2.1
	Importance Rating	3.4	4.3

Ability to determine and interpret the following as they apply to the (LOCA Outside Containment) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question:

Operators are performing 3-EOP-ECA-1.2, "LOCA Outside Containment".

- After closing all valves required by 3-EOP-ECA-1.2, RCS pressure is still decreasing.

Which ONE of the following identifies the procedure and mitigating strategy the SRO will use next?

Transition from 3-EOP-ECA-1.2 to:

A: 3-EOP-ECA-1.1, "Loss of Emergency Coolant Recirculation".
Minimize HHSI flow and maintain RCS cooldown rate at less than 100°F/hr

B: 3-EOP-E-1, "Loss of Reactor or Secondary Coolant".
Minimize HHSI flow and maintain RCS cooldown rate at less than 100°F/hr

C: 3-EOP-ECA-1.1, "Loss of Emergency Coolant Recirculation".
Maintain two HHSI pumps running and cool down the RCS at the maximum achievable rate

D: 3-EOP-E-1, "Loss of Reactor or Secondary Coolant".
Maintain two HHSI pumps running and cool down the RCS at the maximum achievable rate

Proposed Answer: A

Explanation (Optional):

A: Correct per the references and discussion above. Transition should be to EOP-ECA-1.1 and the strategy is to cool down rapidly and minimize HHSI flow

B: Incorrect because the SRO should transition to EOP-ECA-1.1, not EOP-E-1. Plausible because EOP-E-1 is the correct procedure for most LOCAs and would be correct if RCS pressure was increasing

C: Incorrect because the strategy is to cool down rapidly (but <100°F/hr) and minimize HHSI flow. Plausible because ECA-1.1 is the correct procedure to transition to

D: Incorrect because the SRO should transition to EOP-ECA-1.1, not EOP-E-1 and should minimize HHSI flow. Plausible because EOP-E-1 is the correct procedure for most LOCAs and would be correct if RCS pressure was increasing. Also plausible because it is desired to cool down to Cold Shutdown as soon as possible but it must be done within the 100°F/hr limit

Technical Reference(s): 3-EOP-ECA-1.2 Step 3
3-EOP-ECA-1.1 Steps 5 and 18 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank #
Modified Bank # X (Note changes or attach parent)
New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Level 2 because the SRO must evaluate the given conditions and then select the EOP that will provide direction to mitigate this LOCA outside containment. Then the SRO must recall the mitigating strategy of ECA-1.1 which is to minimize HHSI flow and cool down rapidly but within the limit of 100°F/hr.

SRO level because the SRO is assessing plant conditions and then prescribing a procedure to mitigate the event (EOP-ECA-1.1)and then is recalling the mitigating actions of the appropriate procedure . (Ref Guidance for SRO-only Questions Rev 0 - page 7 of 19)

QUESTION 82

Facility: Turkey Point

Vendor: WEC

Exam Date: 2007

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	001	AA2.05
	Importance Rating		4.6

Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indications

Proposed Question: SRO Question # 82

Unit 3 is raising power from 50% to 100% power at 10%/hr by boron dilution and rod withdrawal with the following conditions:

- Reactor Power is 85%.
- Control Bank D Rods are at 200 steps.
- 30 minutes ago, a 200 gallon dilution to the RCS was performed over 2 minutes.
- When the Rod Motion In/Out Switch is released, Tavg and Reactor Power steadily rise.
- VCT level remains stable at 35% for the last 15 minutes.

Which ONE of the following identifies (1) the accident including the required procedure and (2) the bases for those actions in accordance with 0-ADM-536, Technical Specifications Bases Control Program?

A. (1) Uncontrolled Rod Withdrawal Accident
(2) To minimize the impact on power distribution limits

B. (1) Uncontrolled Rod Withdrawal Accident
(2) To prevent Axial Flux Difference from exceeding Technical Specification limits

C. (1) Dilution Accident
(2) To minimize the impact on power distribution limits

D. (1) Dilution Accident
(2) To prevent Axial Flux Difference from exceeding Technical Specification limits

Proposed Answer: A

Explanation (Optional):

A. Correct. Correct event. Bases is IAW T.S.

B. Incorrect. Plausibility - Correct event. Plausible because AFD will be affected. Incorrect since AFD will go positive for outward rod movement, not negative.

C. Incorrect. Plausibility - Incorrect event. The candidate should realize VCT level will increase for a constant rate dilution. For a over dilution (batch) event, they will understand diluting 200 gallons (enough for 1% increase) which is about ten times the water used for temperature control at 100% power. Also, plausible because the correct reason is listed to minimize the impact on power distribution limits

D. Incorrect. Plausibility - Incorrect event. The candidate should realize VCT level will increase for a constant rate dilution. For a over dilution (batch) event, they will understand diluting 200 gallons (enough for 1% increase) which is about ten times the water used for temperature control at 100% power. Also, plausible because AFD will be affected.

Technical Reference(s): 3-ONOP-028 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: None Found (No LP for this procedure?) (As available)

Question Source: Bank # WTSI 68772

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 VC Summer

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

TECHNICAL SPECIFICATION BASES

3/4.1.2 (Cont'd)

The boron concentration of the RWST in conjunction with manual addition of borax ensures that the solution recirculated within containment after a LOCA will be basic. The basic solution minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The temperature requirements for the RWST are based on the containment integrity and large break LOCA analysis assumptions.

The OPERABILITY of one Boron Injection flowpath during REFUELING ensures that this system is available for reactivity control while in Mode 6. Components within the flowpath, e.g., boric acid transfer pumps or charging pumps, must be capable of being powered by an OPERABLE emergency power source, even if the equipment is not required to operate.

The OPERABILITY requirement of 55°F and corresponding surveillance intervals associated with the Boric Acid Tank System ensures that the solubility of the boron solution will be maintained. The temperature limit of 55°F includes a 5°F margin over the 50°F solubility limit of 3.5 wt.% boric acid. Portable instrumentation may be used to measure the temperature of the rooms containing boric acid sources and flow paths.

3/4.1.3 Movable Control Assemblies

The specifications of this section ensure that: (1) Acceptable power distribution limits are maintained, (2) The minimum SHUTDOWN MARGIN is maintained, and (3) The potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within the Allowed Rod Misalignment of Specification 3.1.3.1 of the demand counter position. For the Shutdown Banks and Control Banks A and B, the Position Indication requirement is defined as the group demand counter indicated position between 0 and 30 steps withdrawn inclusive, and between 200 steps withdrawn and All Rods Out (ARO) inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position. For Control Banks C and D, the Position Indication requirement is defined as the group demand counter indicated position between 0 steps withdrawn and All Rods Out (ARO) inclusive.

The increase in the Allowable Rod Misalignment below 90% or Rated Thermal Power is as a result of the increase in the peaking factor limits as reactor power is reduced.

Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits.

TECHNICAL SPECIFICATION BASES

3/4.2 Power Distribution Limits

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) Maintaining the minimum DNBR in the core greater than or equal to the applicable design limit during normal operation and in short-term transients, and (2) Limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{XY}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 Axial Flux Difference

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ limit defined in the CORE OPERATING LIMITS REPORT times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

ATTACHMENT 1
(Page 27 of 114)

TECHNICAL SPECIFICATION BASES

3/4.2.1 (Cont'd)

At power level below PT, the limits on AFD are specified in the CORE OPERATING LIMITS REPORT (COLR) for RAOC operation. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event that such a deviation occurs, a 15 minute period of time allowed outside of the AFD limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the power level.

With PT greater than 100%, two modes are permissible: 1) RAOC with fixed AFD limits as a function of reactor power level, and 2) Base Load operation which is defined as the maintenance of the AFD within a band about a target value. Both the fixed AFD limits for RAOC operation and the target band for Base Load operation are defined in the COLR and the Peaking Factor Limit Report, respectively. However, it is possible during extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with FQ(Z) less than its limiting value. Therefore, PT is calculated to be less than 100%. To allow operation at the maximum permissible value above PT Base Load operation restricts the indicated AFD to a relative small target band and power swings. For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed (15 minutes) will not result in significant xenon redistribution such that the envelope of peaking factors will change sufficiently to prohibit continued operation in the power region defined above. To assure that there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period within a defined range of PT and AFD allowed by RAOC is necessary. During this period, load changes and rod motion are restricted to that allowed by the Base Load requirement. After the waiting period, extended Base Load operation is permissible.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitoring Alarm. The computer monitors the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are: 1) Outside the acceptable AFD (for RAOC operation), or 2) Outside the acceptable AFD target band (for Base Load operation). These alarms are active when power is greater than: 1) 50% of RATED THERMAL POWER (for RAOC operation), or 2) PT (Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short time period during which operation outside of the target band is allowed.

TECHNICAL SPECIFICATION BASES

3/4.2.2 &
3/4.2.3 Heat Flux Hot Channel Factor and Nuclear Enthalpy Rise Hot Channel Factor

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) The design limits on peak local power density and minimum DNBR are not exceeded, and (2) In the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. The LOCA peak fuel clad temperature limit may be sensitive to the number of steam generator tubes plugged.

$F_Q(Z)$, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

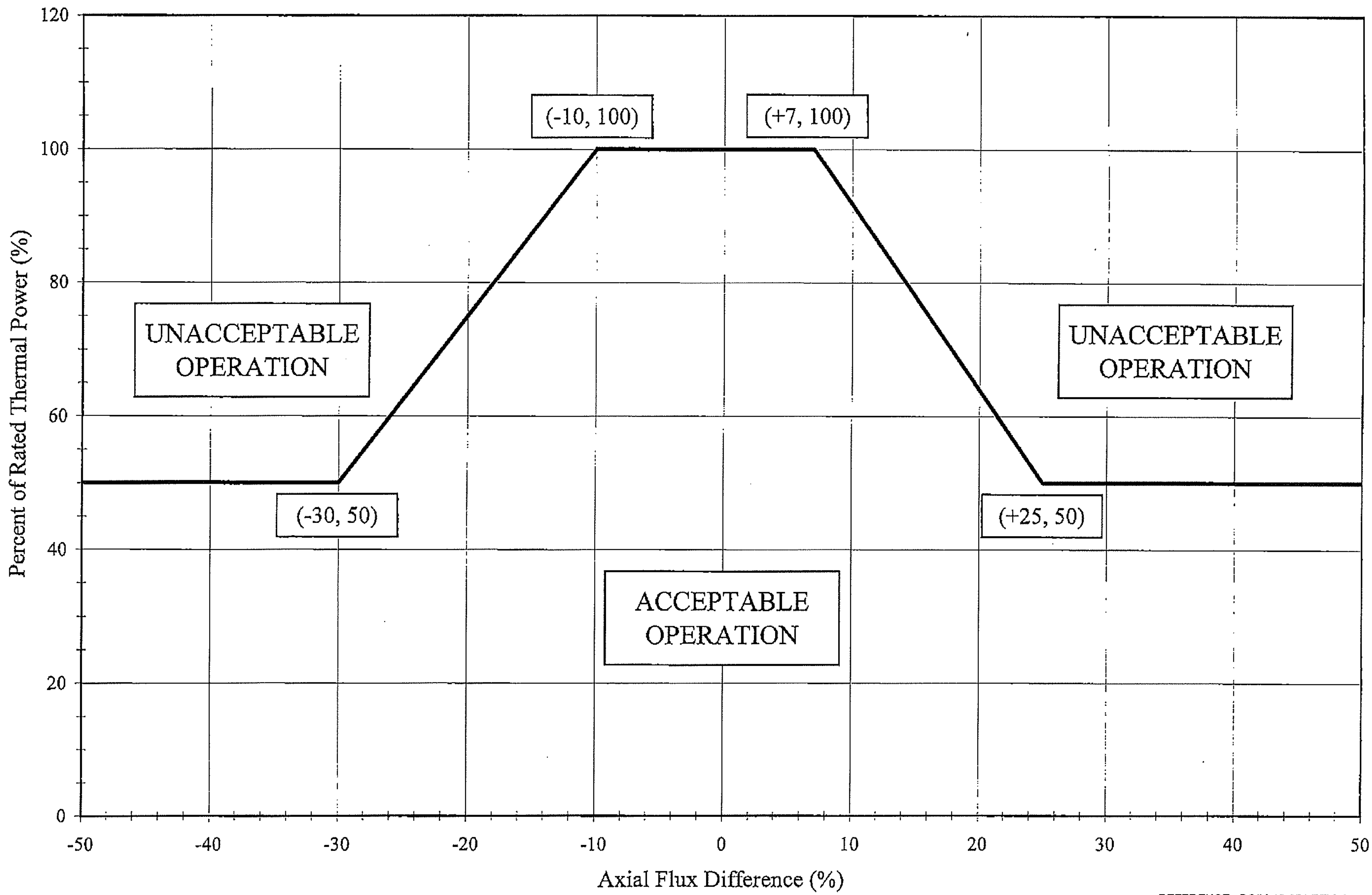
c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

Plant Curve Book

UNIT 3 CYCLE 25

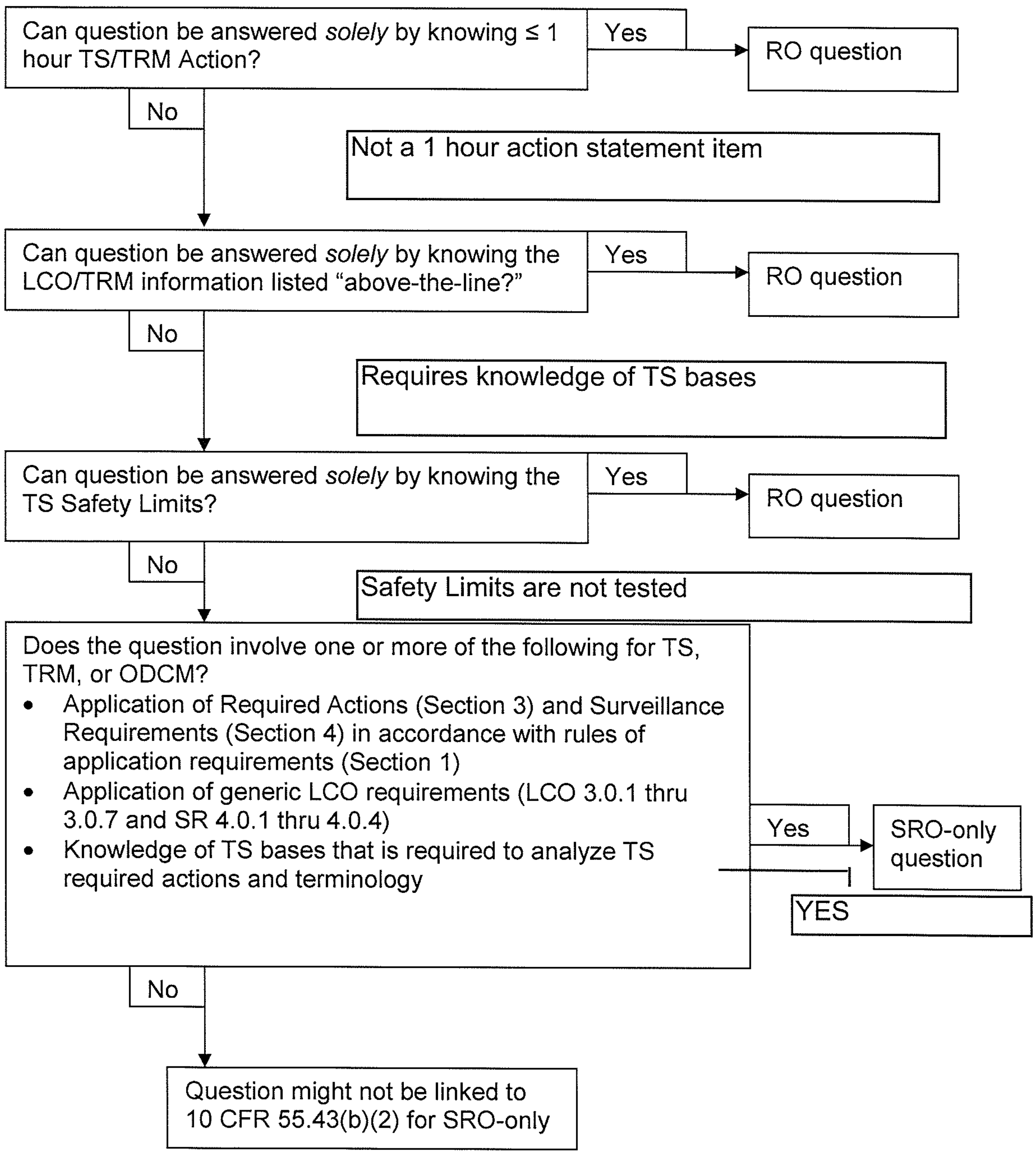
Axial Flux Difference as a Function of Rated Thermal Power



REFERENCE: PC/M 10-051 REV. 0

Question # 82

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



2.4 Continuous Withdrawal of an RCC Control Bank

2.4.1 RCCs stepping out as indicated on the RPIs or group demand step counters, and not manually initiated by the operator.

2.4.2 Tavg increases more than 1.5 degrees F above Tref

2.4.3 Annunciators

1. B 4/4, TAVG/TAVG - TREF DEVIATION
2. B 4/5, RCS HI/LO TAVG
3. B 6/3, POWER RANGE OVERPOWER ROD STOP

2.5 Control Bank D Demanded Past ARO Position

2.5.1 Control Bank D group demand step counters indicate greater than the ARO position (228, 229, or 230 steps as defined in Plant Curve Book, Section 7, COLR).

1.0 **PURPOSE**

1.1 This procedure provides instructions to control RCS boron concentration and maintain VCT level when there is a malfunction of the Boron Concentration Control System.

2.0 **SYMPTOMS OR ENTRY CONDITIONS**

2.1 VCT level continuing to decrease below 17 percent.

2.2 VCT level and pressure are not both increasing or both decreasing.

2.3 Unexpected flow rates of boric acid or primary water during an auto makeup.

2.4 Unexplained deviation of VCT level indication between LI-3-115 in the Control Room and LI-3-112 in the Charging Pump Room.

2.5 **ANNUNCIATORS**

2.5.1 A 2/5 BORIC ACID MAKEUP FLOW DEVIATION

2.5.2 A 2/6 PRI WATER MAKEUP FLOW DEVIATION

2.5.3 A 4/5 VCT HI TEMP/HI/LO PRESS

2.5.4 A 4/6 VCT HI/LO LEVEL

2.5.5 B 3/2 PRI WATER TO BLENDER LO PRESS

2.5.6 I 7/4 PWST LO LEVEL

2.5.7 I 7/5 PRI WATER HEADER LO PRESS

2.5.8 I 5/6 PRI WATER STANDBY PUMP RUNNING

QUESTION 83

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	069	2.4.41
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications.

Proposed Question: SRO Question # 83

Given the following:

Unit 3 has experienced a Large Break Loss of Coolant Accident (LOCA) from 100% power.

The operating crew notes the following parameters:

- Containment pressure is 38 psig.
- Containment temperature is 220°F
- Only 4A HHSI Pump and 3A Containment Spray Pump is able to be started.
- All 3 Emergency Containment Coolers have failed.
- RCS is superheated.
- CHRRMs are reading 1.3E5 R/hr.
- RAD-6304, Plant Vent SPING, reads 5.0E-1 μC/cc for 20 minutes.
- Dose assessments indicate dose at the Site Area Boundary is 150 mRem TEDE.

Which ONE of the following describes 1) The Emergency Action Level (EAL) required to be declared and 2) the Protective Action Recommendations (PARs), if any, that should be issued for this EAL?

REFERENCES PROVIDED

A. (1) General Emergency
   (2) Evacuate ALL people 0-2 miles AND 2-5 miles (downwind sectors) radius from the plant.

B. (1) General Emergency
   (2) Shelter ALL people 0-2 miles AND 2-5 miles (downwind sectors) radius from the plant.

C. (1) Site Area Emergency
   (2) Shelter ALL people 0-2 miles AND 2-5 miles (downwind sectors) radius from the plant.

D. (1) Site Area Emergency
   (2) None. Low offsite dose projections require NO minimum PARs.

Proposed Answer: A

Explanation (Optional):

A. Correct. A GE should be declared and based on indications given, action is correct.

B. Incorrect. Plausible because the first part is correct, and also because the second part is true for minimum PAR when a GE is declared

C. Incorrect since a GE, vs a SAE, should be declared. Plausible because the values given for the RAD-6304 and dose assessments exceed the Threshold Values for RS1, 1 and 2, on Attachment I of 0-EPIP-20101 (Page 6). Also plausible because, if there are extremely stable radiological conditions, and offsite does projections are high, there may be some PARs issued. Per 0-EPIP-20134, Section 1.7.1: "Protective actions for the general public are ordinarily NOT required prior to declaration of a General Emergency. It is possible however, that due to unusually stable and constant meteorological conditions, protective actions could be recommended at a Site Area Emergency based on projected doses.

D. Incorrect since a GE, vs SAE, should be declared. Plausible because the 2nd part is correct for a SAE. Also plausible because the values given for the RAD-6304 and dose assessments exceed the Threshold Values for RS1, 1 and 2, on Attachment I of 0-EPIP-20101 (Page 6).

Technical Reference(s): 0-EPIP-20101, Duties of Emergency Coordinator
0-EPIP-20134, Offsite Notifications and Protective Action Recommendations (Attach if not previously provided)

Proposed References to be provided to applicants during examination: F444, F668, and F669

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3PEO)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO Only Justification:

This question is SRO Only because it tests ability to declare the proper EAL.

K/A Match Justification:

This question matches the K/A in that it tests Loss of Containment Integrity (Potential Loss of the Primary Containment Barrier) and its impact on an EAL Threshold (GE - based on loss of 2/3 barriers and potential loss of the 3rd).

HOT CONDITIONS TABLE (RCS > 200°F)

HOT CONDITIONS TABLE INDEX

QUESTION 84

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	076	2.4.31
Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: SRO Question # 84

Given the following plant conditions:

- Unit 3 at 100% power.
- Annunciator H1/4, PRMS HI RADIATION, is lit.
- R-3-20, Reactor Coolant Letdown Monitor is in High Alarm.
- Chemistry reported Dose Equivalent I-131 has exceeded 100 μCi/gm.
- A unit shutdown is in progress.

Which ONE of the choices below completes the following statements?

After the unit shutdown, Tavg is 525°F. In accordance with 3-ONOP-041.4, Excessive Reactor Coolant System Activity, the require average Reactor Coolant temperature is ___(1)___ .

During the shutdown, the Reactor Coolant activity rises to 320 μCi/gm. The emergency classification is ___(2)___.

REFERENCE PROVIDED

A. (1) MET
(2) Alert

B. (1) NOT MET
(2) Alert

C. (1) MET
(2) Notification Of Unusual Event

D. (1) NOT MET
(2) Notification Of Unusual Event

Proposed Answer: B

Explanation (Optional):

A. Incorrect since the required temperature is < 500°F, not 525°F. Plausible because the 2nd part is correct.

B. CORRECT.

Per 3-ONOP-041.4, Section 5.4 (Page 5):

Be in at least Hot Standby with average reactor coolant temperature less than 500°F within 6 hours. (T.S. 3.4.8) When specific activity exceeds 1 μCi/gm Dose Equivalent I-131 for more than 48 hours during one continuous time interval. After the Reactor Coolant activity rises to 300 uCi/gm. The emergency classification is Alert.

C. Incorrect since after the Reactor Coolant activity rises above 300 uCi/gm. The emergency classification is Alert. Plausible because being in at least Hot Standby with an average reactor coolant temperature at 525°F sounds reasonable due to being less than no load temperature.

D. Incorrect since after the Reactor Coolant activity rises above 300 uCi/gm. The emergency classification is Alert. Plausible 1st part is correct. Also plausible because being in at least Hot Standby with average reactor coolant temperature at 525°F sounds reasonable, but is not less than 500°F.

Technical Reference(s):

0-EPIP-20101, Duties of Emergency Coordinator

3-ONOP-041.4, Excessive Reactor Coolant System Activity

F668 and F669

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: F668 and F669

Learning Objective: LP 6902228, Obj. 6 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

K/A Match Justification:

This question matches the K/A in that it tests knowledge of a response procedure (3-ONOP-041.4) for a High RCS Activity.

Question Selection Methodology:

Deleted original question selected by the Exam Generator because it had nothing to do with RCS activity. It was a Service Water System question. Wrote a new question to replace it.

SRO Only Justification:

This question is SRO Only because it tests knowledge of bases for a TS that is not a Safety Limit. Additionally, it tests knowledge of a TS action that is greater than 1 hour.

HOT CONDITIONS TABLE (RCS > 200°F)

HOT CONDITIONS TABLE INDEX

QUESTION 85

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	E03	EA2.2
Importance Rating		4.1

Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: SRO Question # 85

Following a LOCA, Unit 3 entered 3-EOP-E-1, Loss of Reactor or Secondary Coolant.

Currently, the following plant conditions exist:

- Tcold is 330°F
- RCS pressure is 660 psig and stable
- S/G pressures are 580 psig and lowering slowly due AFW Pump operation
- Containment temperature is 195°F
- RWST level = 275,000 gallons
- HHSI Flow is 750 gpm.
- RHR Flow is 0 gpm

~~Based on the above conditions,~~ which ONE of the following procedures provides the required actions to mitigate these plant conditions?

A. 3-EOP-ES-1.1, SI Termination

B. 3-EOP-ES-1.2, Post LOCA Cooldown and Depressurization

C. 3-EOP-ES-1.3, Transfer To Cold Leg Recirculation

D. 3-EOP-FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

Proposed Answer: B

Explanation (Optional):

A. Incorrect procedure transition based on above conditions.

B. Correct. IAW 3-EOP-E-1, Step 19, if RCS pressure is > 250 [650] psig, go to 3-EOP-ES-1.2.

C. Incorrect procedure transition based on above conditions.

D. Incorrect procedure transition based on above conditions. Plausible why?

Technical Reference(s): 3-EOP-E-1, Step 19 (Attach if not previously provided)

3-EOP-ES-1.2, Step 1

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902327, Obj 6 (As available)

Question Source: Bank #

Modified Bank # WTSI 62347 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

From Turkey Point 2005 NRC Exam. Stem Modification and Major changes to Distracters A, C, and D

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

Question is not testing system knowledge. Testing procedure selection and strategy based on a specific set of conditions

No ↓

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

Yes → RO question

Item does not test Immediate Actions. It tests understanding of actions and procedure required to mitigate a specific set of conditions

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

Plant conditions presented require knowledge of near term mitigation strategy and the procedure that provides it

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

This item requires interpretation of conditions to arrive at a conclusion for a specific strategy, not overall strategy

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. Selection of procedure and determine strategy

Yes → SRO-only question

No ↓

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19	**Check If RCS Cooldown And Depressurization Is Required**	
	a. RCS pressure - GREATER THAN 250 PSIG[650 PSIG]	a. Perform the following: 1) **IF** RHR pump flow greater than 1000 gpm, **THEN** go to Step 20. 2) **IF** RHR pump flow less than or equal to 1000 gpm, **THEN** go to 3–EOP–ES–1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.
	b. Go to 3-EOP-ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1	

FOLDOUT FOR PROCEDURE E-1

1. **ADVERSE CONTAINMENT CONDITIONS**
IF either of the conditions listed below occurs, **THEN** use adverse containment setpoints:
Containment atmosphere temperature ≥ 180°F
OR
Containment radiation levels ≥ 1.3×10^5 R/hr
WHEN containment parameters drop below the above values, **THEN** normal setpoints can again be used **IF** containment integrated dose rate has not exceeded 10^6 Rads.

2. **RCP TRIP CRITERIA**
a. **IF** all conditions listed below occur, **THEN** trip all RCPs:
   1) High-head SI pumps - AT LEAST ONE RUNNING **AND** SI FLOWPATH VERIFIED
   2) RCS subcooling - LESS THAN 25°F[65°F]
   3) Controlled RCS cooldown is NOT in progress
b. **IF** phase B actuated, **THEN** trip all RCPs

3. **SI TERMINATION CRITERIA**
IF all conditions listed below occur, **THEN** go to 3-EOP-ES-1.1, SI TERMINATION, Step 1:
a. RCS subcooling based on core exit TCs - GREATER THAN 30°F[See below Table]

SI TERMINATION ADVERSE SUBCOOLING VALUE	
RCS PRESSURE (PSIG)	**ADVERSE SUBCOOLING VALUE**
< 2485 AND ≥ 2000	≥ 55 °F
< 2000 AND ≥ 1000	≥ 85 °F
< 1000	≥ 210°F

b. Total feed flow to intact SGs - GREATER THAN 345 GPM **OR** narrow range level in at least one intact SG - GREATER THAN 6%[32%]
c. RCS pressure - GREATER THAN 1600 PSIG[2000 psig] **AND** STABLE OR INCREASING
d. PRZ level - GREATER THAN 17%[50%]

4. **SECONDARY INTEGRITY CRITERIA**
IF any S/G pressure is decreasing in an uncontrolled manner **OR** has completely depressurized **AND** that S/G has NOT been isolated, **THEN** go to 3-EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.

5. **E-3 TRANSITION CRITERIA**
IF any S/G level increases in an uncontrolled manner **OR** any S/G has abnormal radiation, **THEN** manually start SI pumps as necessary and go to 3-EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.

6. **COLD LEG RECIRCULATION SWITCHOVER CRITERIA**
IF RWST level decreases to less than 155,000 gallons, **THEN** go to 3-EOP-ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

7. **RECIRCULATION SUMP BLOCKAGE**
IF RHR pump flow **AND** amps become erratic **OR** abnormally low after recirculation has been established, **THEN** transition to 3-EOP-ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.

8. **CST MAKEUP WATER CRITERIA**
IF CST level decreases to less than 10%, **THEN** add makeup to CST using 3-NOP-018.01, Condensate Storage Tank (CST).

9. **LOSS OF OFFSITE POWER OR SI ON OTHER UNIT**
IF SI has been reset **AND** either offsite power is lost **OR** SI actuates on the other unit, **THEN** restore safeguards equipment to required configuration. Refer to ATTACHMENT 3 for essential loads.

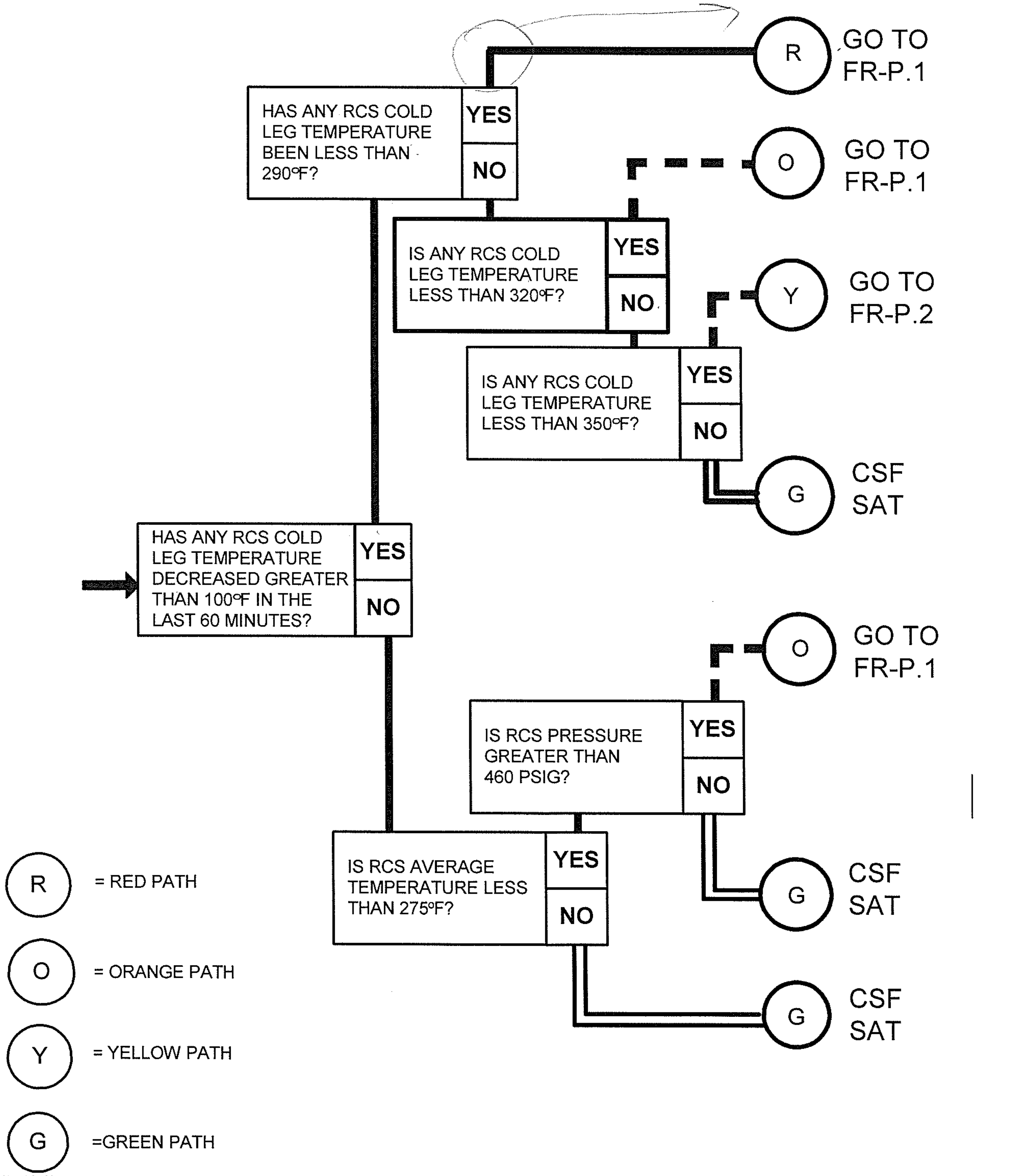
10. **RHR SYSTEM OPERATION CRITERIA**
IF RHR flow is less than 1000 gpm, **THEN** the RHR pumps shall be shut down within 44 minutes of the initial start signal.

W97/DH/ln/cls/mr

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

ENCLOSURE 4
(Page 1 of 1)

CSF F-0.4 INTEGRITY



W97/daj/nw

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2005
Exam Type:	S

Original Item for question 85

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	1
	Group #	2	2
	K/A #	E03	EA2.2
	Importance Rating	3.5	4.1

Ability to determine and interpret the following as they apply to the (LOCA Cooldown and Depressurization) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question:

Following a LOCA with a concurrent Loss Of Offsite Power (LOOP), Unit 3 entered E-1, LOSS OF REACTOR OR SECONDARY COOLANT. Currently, the following plant conditions exist:

- Tavg is 345°F
- RCS pressure is 350 psig
- Containment temperature is 178°F
- RWST level = 255,000 gallons
- A mechanical failure of one train of SI has just occurred

Which ONE of the following describes the required operator actions in accordance with E-l?

A: Transition to ES-1.1, SI TERMINATION. Stop the running HHSI and RHR pumps and place in standby.

B: Transition to ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. Stop the running RHR pump and place in standby.

C: Transition to ES-1.3, TRANSITION TO COLD LEG RECIRCULATION. Align RHR suction to containment recirc sump

D: Transition to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION. Establish makeup to the Unit 3 RWST.

Proposed Answer: B

Explanation (Optional):

A: Incorrect. The transition is not correct. The stated action would be correct if the transition were made to ES-1 .

B: Correct. IAW E-1, Step 19, if RCS pressure is > 250 psig, go to ES-1.2, step 1 which directs stopping RHR pumps.

C: Incorrect. While this is a valid transition from E-1, it occurs when RWST drops below 155,000 gallons so this transition would be incorrect.

D: Incorrect. While this is a valid transition from E-I, it occurs if neither RHR pump is available to support cold leg recirculation so this transition would be incorrect.

Technical Reference(s):

1. E-1 , LOSS OF REACTOR OR SECONDARY COOLANT, pages 18,21, rev 04/03/02
2. BD-E-1, LOSS OF REACTOR OR SECONDARY COOLANT, page 35, rev 04/03/02
3. TECHNICAL SPECIFICATION, 3.5.3, page 314 5-9, Amendment Nos. 138 & 133
4. 0-ADM-536, TECH SPEC BASES CONTROL PROGRAM, page 72, rev 05/01/03
5. ES-1.2, POST LOCA COOLDOWN & DEPRESSURIZATION, pgs 3,6,12, rev 04/03/02
6. BD ES-1.2, POST LOCA COOLDOWN AND DEPRESS, page 8, rev 04/03/02
7. TECHNICAL SPECIFICATION, 3.5.4, page 314 5-10, Amendment Nos. 138 & 133

(Attach if not previously provided)

Proposed References to be provided to applicants during examination:		No
Learning Objective:		(As available)
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New X	
Question History:		
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis X	
10 CFR Part 55 Content:	55.41	
	55.43	

Comments:

QUESTION 86

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	003	A2.03
Importance Rating		3.1

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

Proposed Question: SRO Question # 86

The following conditions exist:

- Unit 4 is operating at 100% power.
- H-9/6, RCP A/B/C PUMP/MOTOR HI TEMP, is lit.
- H-9/1, RCP A MOTOR BEARING HI TEMP, is lit.
- 4A RCP Motor Bearing temperature is 187°F and rising at 5°F/minute.
- 4A RCP Motor Stator temperature is 223°F and rising at 5°F/minute.

In accordance with 4-ONOP-041.1, Reactor Coolant Pump Off-Normal, which ONE of the following describes (1) what action is required for 4A RCP and (2) the EARLIEST required time to notify the Nuclear Regulatory Commission Operations Center (NRCOC) in accordance with 0-ADM-115, Notification of Plant Events?

A. (1) Trip the Unit 4 Reactor and 4A RCP based on RCP Motor Bearing temperature
(2) 1 hour

B. (1) Trip the Unit 4 Reactor and 4A RCP based on Motor Stator temperature
(2) 1 hour

C. (1) Trip the Unit 4 Reactor and 4A RCP based on Motor Stator temperature
(2) 4 hours

D. (1) Trip the Unit 4 Reactor and 4A RCP based on RCP Motor Bearing temperature
(2) 4 hours

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because it is the correct action with correct parameter. However, the time is incorrect for notification. 1 hour is plausible since it is the time associated with emergency notification per 0-ADM-115.

B. Incorrect. Plausible because it is the correct action with an incorrect parameter. The calculated time is close, but RCP Motor Bearing temperature will be exceeded first. However, the time is incorrect for notification. 1 hour is plausible since it is the time associated with emergency notification per 0-ADM-115.

C. Incorrect. Plausible because it is the correct action with an incorrect parameter. The calculated time is close, but RCP Motor Bearing temperature will be exceeded first. However, the time is correct for notification. 4 hours is the required NRC notification time for an unplanned trip due to faulty equipment operation per 0-ADM-115.

D. Correct.
4A RCP Motor Bearing temperature is 187°F. (195-187)/5 = 1.6 minutes
4A RCP Motor Stator temperature is 223°F. (248-223)/5 = 5 minutes

Trip the Unit 4 Reactor and 4A RCP based on RCP Motor Bearing temperature.

4 hours is the required NRC notification time for an unplanned trip due to faulty equipment operation per 0-ADM-115

Technical Reference(s): 4-ONOP-041.1 (Attach if not previously provided)
H-9/1 & 6 (4-ARP-097.CR.H)
0-ADM-115

Proposed References to be provided to applicants during examination: None

Learning Objective: 6902205 Obj 7 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal,

abnormal, and emergency situations.

Comments:

SRO because procedure selection must be made based upon prioritization of multiple abnormal indications

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

Question is not testing system knowledge. Testing procedure action based upon conditions that requires RCP S/D, and NRC notification

No ↓

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

Yes → RO question

Item does not test Immediate Actions. It tests understanding of actions and procedure required to mitigate a specific set of conditions

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

Plant conditions presented require knowledge of near term mitigation strategy inside an AOP and the required notification to NRC

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

This item requires interpretation of conditions to arrive at a conclusion for a specific strategy (1 of several potential strategies, not overall strategy

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. Selection of strategy

Yes → SRO-only question

Yes. NRC notification requirements

No ↓

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

FOLDOUT PAGE FOR PROCEDURE 3-ONOP-041.1

1. **RCP VIBRATION ASSESSMENT CRITERIA**

   IF motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10), is greater than or equal to 3 mils but less than 5 mils, **THEN** contact Engineering to evaluate the condition.

2. **RCP STOPPING CRITERIA**

   IF any of the following RCP limits are reached, **THEN** manually trip the reactor and verify reactor trip using the EOP network **AND** then stop the affected RCP, and close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

   - RCP number one seal ΔP - LESS THAN 200 psid.
   - RCP number one seal leakoff temperatures on DCS - GREATER THAN OR EQUAL TO 235°F.
   - RCP pump bearing temperature on DCS - GREATER THAN OR EQUAL TO 225°F.
   - RCP motor bearing temperature on DCS - GREATER THAN OR EQUAL TO 195°F.
   - RCP stator winding temperature on DCS - GREATER THAN OR EQUAL TO 248°F. Note exception in Foldout Page Item 4.
   - Motor frame vibration, R-3-369 (Points 1, 2, 5, 6, 9, 10) - GREATER THAN OR EQUAL TO 5 MILS. Note exception in Foldout Page Item 4.
   - RCP shaft vibration, R-3-369 (Points 3, 4, 7, 8, 11, 12) - GREATER THAN OR EQUAL TO 20 MILS. Note exception in Foldout Page Item 4.

3. **RCP SEAL CRITERIA FOR STOPPING RCP**

   WHEN the RCP number one seal leakoff flow exceeds 6 gpm, **THEN** perform the following:

   a. Trip the reactor **AND** verify the reactor tripped using the EOP network.

   b. Stop the affected RCP.

   c. Close the applicable RCP Seal Leakoff Isolation Valve 303A, 303B, or 303C.

   d. Close PCV-3-455A, PZR Spray Valve Loop C, or PCV-3-455B, PZR Spray Valve Loop B, if applicable.

4. **EXCEEDING VIBRATION OR STATOR TEMPERATURE LIMITS**

   For the basis of obtaining data for startup, for balancing an RCP, or for shutdown operations; the Electrical Maintenance Supervisor or Component Engineering Supervisor may authorize continued RCP operations with vibration level or stator winding temperature above stopping criteria noted in Foldout Page Item 2. This authorization is required to be obtained prior to starting the RCP.

ENCLOSURE 1
(Page 1 of 9)

NRC NOTIFICATION TABLE

Plant Condition or Event	Notification Time Limit	Notes
Deviation from Tech Specs allowed by 10 CFR 50.54(x) and 10 CFR 72.32(d)	One Hour	Perform actions immediately necessary to protect public health and safety. [NUREG-1022] 10 CFR 50.72(b)(1)
Initiation of any nuclear plant shutdown required by Tech Specs.	Four Hours	"Initiation of any Nuclear Plant Shutdown" is the performance of any action to start reducing reactor power to achieve an operational condition or mode that requires the reactor to be subcritical as a result of a Tech Spec requirement. This includes any means of power reduction, such as control rod insertion, boration, or turbine load reduction [NUREG-1022]. Examples are exceeding an LCO action statement, Tech Spec 3.0.3, Safety Limit violation [10 CFR 50.36]. 10 CFR 50.72(b)(2)(i)
An event that results or should have resulted in Emergency Core Cooling System discharge into the RCS as a result of a valid signal.	Four Hours	Actuation from a pre-planned sequence during testing or reactor operation is not reportable. [NUREG-1022] 10 CFR 50.72(b)(2)(iv)(A)
An event that results in actuation of the Reactor Protection System (RPS) when the reactor is critical.	Four Hours	Manual or automatic RPS actuation not part of a pre-planned sequence is reportable. Actuation from a pre-planned sequence during testing or reactor operation is not reportable. Turbine runbacks are not part of RPS and therefore not reportable. [NUREG-1022] 10 CFR 50.72(b)(2)(iv)(B)
One startup transformer and one of the required Emergency Diesel Generators inoperable.	Four Hours	Tech Spec 3.8.1.1.
Both startup transformers inoperable.	Four Hours	Tech Spec 3.8.1.1.
Both standby feedwater pumps inoperable for 24 hours.	Four Hours	Tech Spec 3.7.1.6.

QUESTION 87

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	005	2.2.40
	Importance Rating		3.4

Equipment Control: Ability to apply technical specifications for a system.
Proposed Question: SRO Question # 87

Unit 3 was operating at 100% power with the following conditions:

Time	**Log Entry**
1300	3-OSP-050.2, Residual Heat Removal System Inservice Test has been started for 3A RHR Pump.
1330	3A RHR Pump was declared INOPERABLE due to a problem occurring during 3-OSP-050.2.
1357	3B RHR Pump was declared INOPERABLE due to an excessive pump casing leak.
1444	Unit 3 shutdown was commenced.
1551	3B RHR Pump was returned to OPERABLE status after pump casing repair.
1604	3A RHR Pump was returned to OPERABLE status after a minor repair.

Which ONE of the following describes the Technical Specification requirements for operation of the plant?

Plant conditions.......

REFERENCE PROVIDED

A. allow the plant shutdown to be terminated no earlier than 1551.

B. allow the plant shutdown to be terminated no earlier than 1604.

C. require that a shutdown to Mode 3 is completed by 1457.

D. require that the shutdown to Mode 3 is completed by 2044.

Proposed Answer: A

Explanation (Optional):

A. Correct. With two RHR Pumps no longer inoperable, TS 3.0.3 no longer applies.

B. Incorrect. The plant can be terminated earlier than 1604. Plausible because the applicant may require the 2nd RHR to be returned to service prior to exiting TS 3.0.3.

C. Incorrect. Plausible since action has to be taken within 1 hour to shutdown. The applicant applies this knowledge to the transition to MODE 3.

D. Incorrect. Plausible since this is the time to be in MODE 3 if the shutdown were to continue.

Technical Reference(s): TS 3.5.2, SR 4.5.2 (Attach if not previously provided)

TS 3.0.3

Proposed References to be provided to applicants during examination: TS 3.5.2 with SR 4.5.2

Learning Objective: 6900121A, Obj 12c (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

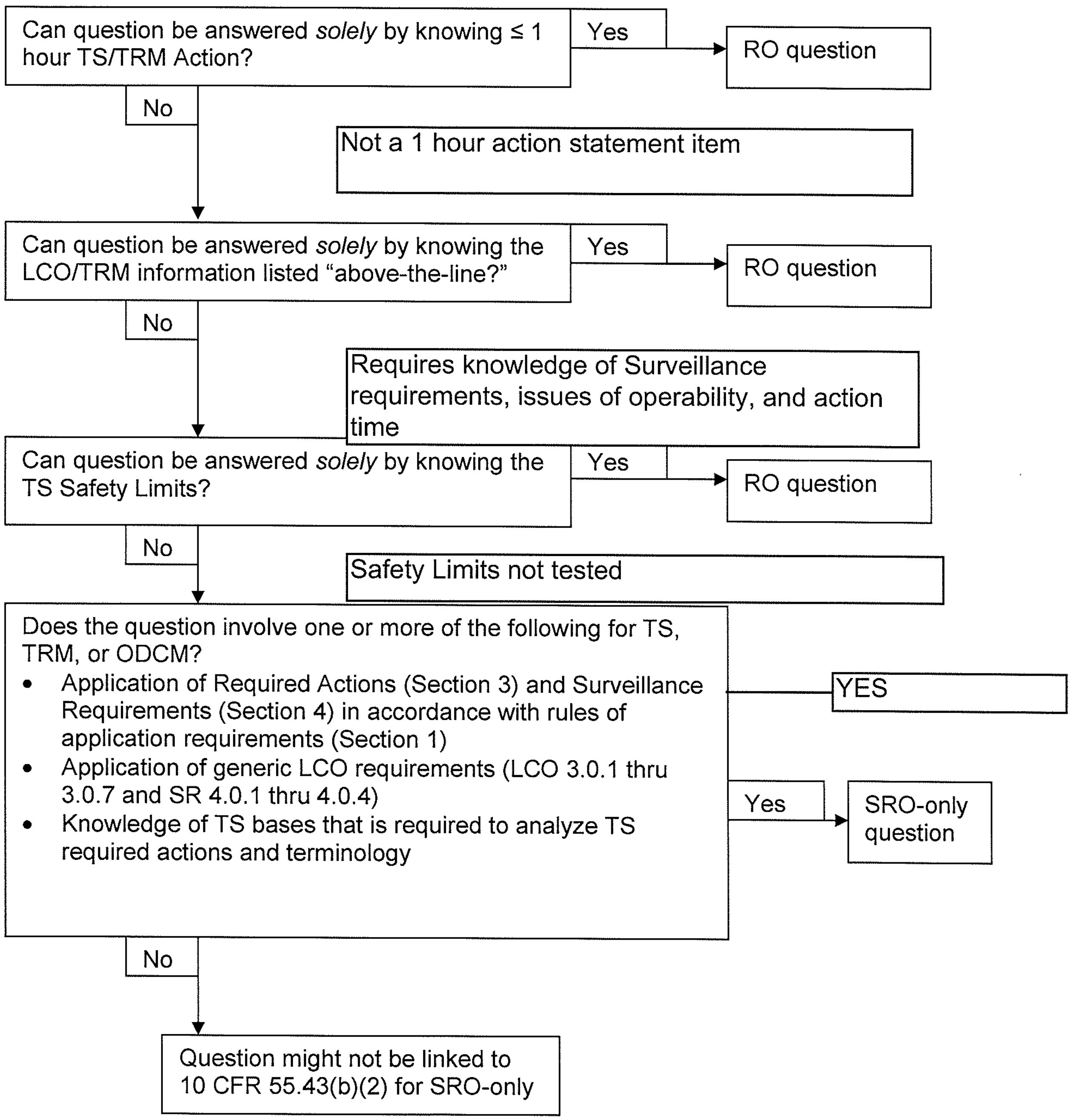
10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

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



3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specification 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

Q#87

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 The following Emergency Core Cooling System (ECCS) equipment and flow paths shall be OPERABLE:

a. Four OPERABLE Safety Injection (SI) pumps, each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold legs,*

b. Two OPERABLE RHR heat exchangers,

c. Two OPERABLE RHR pumps with discharge aligned to the RCS cold legs,

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank as defined in Specification 3.5.4, and

e. Two OPERABLE flow paths capable of taking suction from the containment sump.

APPLICABILITY: MODES 1, 2, and 3**.

ACTION:

a. With any one of the required ECCS components or flow paths inoperable, except for inoperable Safety Injection Pump(s) or an inoperable RHR pump, restore the inoperable component or flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date since January 1, 1990.

c. With one of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours.***

---

*Only three OPERABLE Safety Injection (SI) pumps (two associated with the unit and one from the opposite unit), each capable of being powered from its associated OPERABLE diesel generator#, with discharge aligned to the RCS cold leg are required if the opposite unit is in MODE 4, 5, or 6.

**The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the Safety Injection flow paths isolated pursuant to Specification 3.4.9.3 provided that the Safety Injection flow paths are restored to OPERABLE status prior to Tavg exceeding 380°F. Safety Injection flow paths may be isolated when Tavg is less than 380°F.

***The provisions of Specifications 3.0.4 and 4.0.4 are not applicable.

#Inoperability of the required EDG's does not constitute inoperability of the associated Safety Injection pumps.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

d. With two of the four required Safety Injection pumps inoperable and the opposite unit in MODE 1, 2, or 3, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With one of the three required Safety Injection pumps inoperable and the opposite unit in MODE 4, 5, or 6, restore the pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

f. With a required Safety Injection pump OPERABLE but not capable of being powered from its associated diesel generator, restore the capability within 14 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With an ECCS subsystem inoperable due to an RHR pump being inoperable, restore the inoperable RHR pump to OPERABLE status within 7 days or be in as least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS component and flow path shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying by control room indication that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
864A and B	Supply from RWST to ECCS	Open
862A and B	RWST Supply to RHR pumps	Open
863A and B	RHR Recirculation	Closed
866A and B	H.H.S.I. to Hot Legs	Closed
HCV-758*	RHR HX Outlet	Open

To permit temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping,

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

3) Verifying that each RHR Pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.

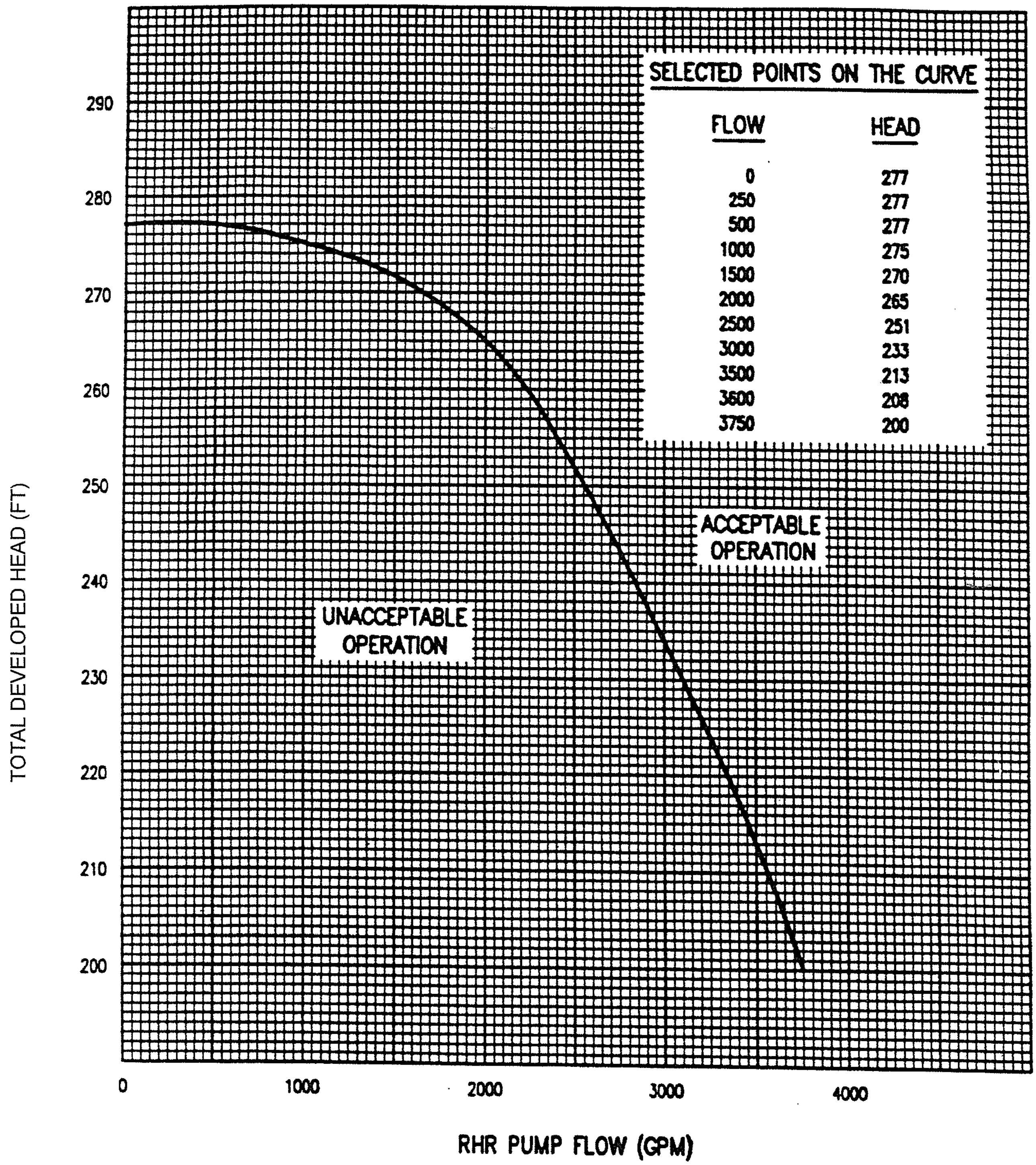
c. At least once per 92 days by:

1) Verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5.

SI pump $\geq$ 1083 psid at a metered flowrate $\geq$ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

$\geq$ 1113 psid at a metered flowrate $\geq$ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

*Air Supply to HCV-758 shall be verified shut off and sealed closed once per 31 days.



FLOW	HEAD
0	277
250	277
500	277
1000	275
1500	270
2000	265
2500	251
3000	233
3500	213
3600	208
3750	200

Figure 3.5-1
RHR Pump Curve

d. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. The visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) At least once daily of the areas affected within containment by containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

e. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 525 psig the interlocks cause the valves to automatically close and prevent the valves from being opened, and

2) Verifying correct interlock action to ensure that the RWST is isolated from the RHR System during RHR System operation and to ensure that the RHR System cannot be pressurized from the Reactor Coolant System unless the above RWST Isolation Valves are closed.

3) A visual inspection of the containment sump and verifying that the suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

f. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on Safety Injection actuation test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Safety Injection pump, and

b) RHR pump.

SURVEILLANCE REQUIREMENTS

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS components are required to be OPERABLE, and

2) At least once per 18 months.

RHR System
Valve Number

HCV-*-758
MOV-*-872

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specification 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

TABLE 1.2

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	≥ 0.99	> 5%	≥ 350°F
2.	STARTUP	≥ 0.99	≤ 5%	≥ 350°F
3.	HOT STANDBY	< 0.99	0	≥ 350°F
4.	HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN	< 0.99	0	≤ 200°F
6.	REFUELING**	≤ 0.95	0	≤ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

QUESTION 88

Facility: Turkey Point
Vendor: WEC
Exam Date: 2011
Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	062	2.2.22
	Importance Rating		2.1

Equipment Control: Knowledge of limiting conditions for operations and safety limits.
Proposed Question: SRO Question # 88

The following conditions exist:

- Unit 3 and 4 are at 100 % power.
- Unit 3 Startup Transformer is INOPERABLE.
- Estimated time for restoration is unknown.

Which ONE of the following actions is required in accordance with Technical Specifications and 0-ADM-536, Technical Specifications Bases Control Program for this limiting condition of operation?

REFERENCE PROVIDED

A. (1) Reduce power operation ≤ 30% on Unit 3 ≤ 48 hours, OR place the associated unit in at least HOT STANDBY within the next 12 hours.
(2) The time period, bounded by online risk assessment models, allows for a reasonable restoration of the Startup Transformer.

B. (1) Reduce power operation ≤ 30% on Unit 3 ≤ 24 hours, then power operation for Unit 3 may continue for 30 days
(2) The time period, bounded by online risk assessment models, allows for a reasonable restoration of the Startup Transformer.

C. (1) Reduce power operation ≤ 30% on Unit 3 ≤ 48 hours, OR place the associated unit in at least HOT STANDBY within the next 12 hours.
(2) < 30% power operations reduces the decay heat level and allows for automatic Feedwater Control.

D. (1) Reduce power operation ≤ 30% on Unit 3 ≤ 24 hours, then power operation for Unit 3 may continue for 30 days
(2) < 30% power operations reduces the decay heat level and allows for automatic Feedwater Control.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausibility – Power reduction to $\leq$ 30% is correct, but in 24 hrs not 48 hrs. However, 48 hrs is the limited time requirement prior to plant shutdown. Also, the associated unit is taken to Hot Stby in 6 hrs vice 12 hrs. 12 hrs is a value required for a dual unit shutdown to Mode 3. The basis is incorrect and something that is used in the work planning and execution process.

B. Incorrect. Plausibility – The first part is the correct action per Tech Specs. Also, the basis is incorrect and something that is used in the work planning and execution process.

C. Incorrect. Plausibility – Power reduction to $\leq$ 30% is correct, but in 24 hrs not 48 hrs. However, 48 hrs is the limited time requirement prior to plant shutdown. Also, the associated unit is taken to Hot Stby in 6 hrs vice 12 hrs. 12 hrs is a value required for a dual unit shutdown to Mode 3. The basis is correct < 30% power operations reduces the decay heat level and allows for automatic Feedwater Control.

D. Correct.

Technical Reference(s): TS 3.8.1.1, 3.8.1.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.8.1.1

Learning Objective: 6902136 Obj 15 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Q#88

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two startup transformers and their associated circuits, and

b. Three separate and independent diesel generators* including,

1) For Unit 3, two (3A and 3B); for Unit 4, one (3A or 3B) each with:

a) A separate skid-mounted fuel tank and a separate day fuel tank with an OPERABLE solenoid valve to permit gravity flow from the day tank to the skid mounted tank, and with the two tanks together containing a minimum of 2000 gallons of fuel oil.

b) A common Fuel Storage System containing a minimum volume of 38,000 gallons of fuel,**

c) A separate fuel transfer pump,**

d) Lubricating oil storage containing a minimum volume of 120 gallons of lubricating oil,

e) Capability to transfer lubricating oil from storage to the diesel generator unit, and

f) Energized MCC bus (MCC 3A vital section for EDG 3A, MCC 3K for EDG 3B).

2) For Unit 3, one (4A or 4B); for Unit 4, two (4A and 4B) each with:

a) A separate day fuel tank containing a minimum volume of 230 gallons of fuel,

b) A separate Fuel Storage System containing a minimum volume of 34,700 gallons of fuel,

c) A separate fuel transfer pump, and

d) Energized MCC bus (MCC 4J for EDG 4A, MCC 4K for EDG 4B).

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*Whenever one or more of the four EDG's is out-of-service, ensure compliance with the EDG requirements specified in Specifications 3.5.2 and 3.8.2.1.

**A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated circuits to OPERABLE status within 24 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c.

ACTION (Continued)

startup transformer and associated circuits within one hour and at least once per 8 hours thereafter; and if the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable diesel generator does not exist on the remaining required diesel generators, unless the diesel generators are already operating; restore one of the inoperable sources to OPERABLE status in accordance with Action Statements a and b, as appropriate. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Notify the NRC within 4 hours of declaring both a start-up transformer and diesel generator inoperable. Restore the other A.C. power source (startup transformer or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

d. With one diesel generator inoperable, in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices (except safety injection pumps) that depend on the remaining required OPERABLE diesel generators as a source of emergency power are also OPERABLE.

      If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   2. At least two Safety Injection pumps are OPERABLE and capable of being powered from their associated OPERABLE diesel generators.

      If this condition is not satisfied within 2 hours, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

e. With two of the above required startup transformers or their associated circuits inoperable notify the NRC within 4 hours; restore at least one of the inoperable startup transformers to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours* and in COLD

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*If the opposite unit is shutdown first, this time can be extended to 42 hours.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously. With only one startup transformer and associated circuits restored, perform Surveillance Requirement 4.8.1.1.1a on the OPERABLE Startup transformer at least once per 8 hours, and restore the other startup transformer and its associated circuits to OPERABLE status or shutdown in accordance with the provisions of Action Statement 3.8.1.1a with time requirements of that Action Statement based on the time of initial loss of a startup transformer. This ACTION applies to both units simultaneously.

f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two startup transformers and their associated circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all required diesel generators to OPERABLE status within 14 days from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

g. Following the addition of the new fuel oil* to the Diesel Fuel Oil Storage Tanks, with one or more diesel generators with new fuel oil properties outside the required Diesel Fuel Oil Testing Program limits, restore the stored fuel oil properties to within the required limits within 30 days.

h. With one or more diesel generators with stored fuel oil total particulates outside the required Diesel Fuel Oil Testing Program limits, restore the fuel oil total particulates to within the required limits within 7 days.

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* The properties of API Gravity, specific gravity or an absolute specific gravity; kinematic viscosity; clear and bright appearance; and flash point shall be confirmed to be within the Diesel Fuel Oil Testing Program limits, prior to the addition of the new fuel oil to the Diesel Fuel Oil Storage Tanks.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required startup transformers and their associated circuits shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months while shutting down, by transferring manually unit power supply from the auxiliary transformer to the startup transformer.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying the fuel volume in the day and skid-mounted fuel tanks (Unit 4-day tank only),

2) Verifying the fuel volume in the fuel storage tank,

3) Verifying the lubricating oil inventory in storage,

4) Verifying the diesel starts and accelerates to reach a generator voltage and frequency of 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz. Once per 184 days, these conditions shall be reached within 15 seconds after the start signal from normal conditions. For all other starts, warmup procedures, such as idling and gradual acceleration as recommended by the manufacturer may be used. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded** to 2300 - 2500 kW (Unit 3), 2650-2850 kW (Unit 4)***, operates at this loaded condition for at least 60 minutes and for Unit 3 until automatic transfer of fuel from the day tank to the skid mounted tank is demonstrated, and the cooling system is demonstrated OPERABLE.

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

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* All diesel generator starts for the purpose of these surveillances may be proceeded by a prelube period as recommended by the manufacturer.

** May include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

***Momentary transients outside these load bands do not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

b. Demonstrating at least once per 92 days that a fuel transfer pump starts automatically and transfers fuel from the storage system to the day tank,

c. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day and skid-mounted fuel tanks (Unit 4-day tank only);

d. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

e. By verifying fuel oil properties of new fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

f. By verifying fuel oil properties of stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

g. At least once per 18 months, during shutdown (applicable to only the two diesel generators associated with the unit):

   1) Deleted

   2)* Verifying the generator capability to reject a load of greater than or equal to 380 kw while maintaining voltage at 4160 ± 420 volts and frequency at 60 ±1.2 Hz;

   3)* Verifying the generator capability to reject a load of greater than or equal to 2500 kW (Unit 3), 2874 kW (Unit 4) without tripping. The generator voltage shall return to less than or equal to 4784 volts within 2 seconds following the load rejection;

   4) Simulating a loss-of-offsite power by itself, and:

      a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

      b. Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently

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* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

SURVEILLANCE REQUIREMENTS (Continued)

connected loads within 15 seconds, energizes the auto–connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the auto-connected shutdown loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 15 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently connected loads within 15 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test; and

   c) Verifying that diesel generator trips that are made operable during the test mode of diesel operation are inoperable.

7)* # Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 2550-2750 kW (Unit 3), 2950-3150 kW (Unit 4)** and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2300-2500 kW (Unit 3), 2650-2850 kW (Unit 4)**. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 15 seconds after the start signal; the steady-state generator voltage and frequency

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* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

** Momentary transients outside these load bands do not invalidate this test.

\# This test may be performed during POWER OPERATION

shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, verify the diesel starts and accelerates to reach a generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz within 15 seconds after the start signal.**

8) Verifying that the auto-connected loads to each diesel generator do not exceed 2500 kW (Unit 3), 2874 kW (Unit 4);

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from the fuel storage tank (Unit 3), fuel storage tanks (Unit 4) to the day tanks of each diesel associated with the unit via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

13) Verifying that the diesel generator lockout relay prevents the diesel generator from starting;

** If verification of the diesel's ability to restart and accelerate to a generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz within 15 seconds following the 24 hour operation test of Specification 4.8.1.1.2.g.7) is not satisfactorily completed, it is not necessary to repeat the 24-hour test. Instead, the diesel generator may be operated between 2300-2500 kW Unit 3, 2650-2850 kW (Unit 4) for 2 hours or until operating temperature has stabilized (whichever is greater). Following the 2 hours/operating temperature stabilization run, the EDG is to be secured and restarted within 5 minutes to confirm its ability to achieve the required voltage and frequency within 15 seconds.

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all required diesel generators simultaneously and verifying that all required diesel generators provide 60 ±1.2 Hz frequency and 4160 ± 420 volts in less than or equal to 15 seconds: and

I. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.*

2) For Unit 4 only, performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

4.8.1.1.3 Reports - (Not Used)

* A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5. 6. or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days. Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

TABLE 4.8-1

(Not Used)

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One startup transformer and associated circuits, or an alternate circuit, between the offsite transmission network and the 4160 volt bus, A or B, and

b. One diesel generator with:

1) For Unit 3 (3A or 3B)

A skid-mounted fuel tank and a day fuel tank, with an OPERABLE solenoid valve to permit gravity flow from the day tank to the skid mounted tank, with the two tanks together containing a minimum of 2000 gallons of fuel oil

For Unit 4 (4A or 4B)

A day fuel tank containing a minimum volume of 230 gallons of fuel

2) A fuel storage system containing a minimum volume of fuel of 38,000 gallons (Unit 3). 34,700 gallons (Unit 4)**

3) An associated fuel transfer pump**

4) For Unit 3 only, lubricating oil storage containing a minimum volume of 120 gallons of lubricating oil

5) For Unit 3 only capability to transfer lubricating oil from storage to the diesel generator unit and

6) Energized MCC bus (as identified by Specification 3.8.1.1.b.).

APPLICABILITY: MODES 5* and 6*.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 2.2 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible and increase RCS inventory as soon as possible.

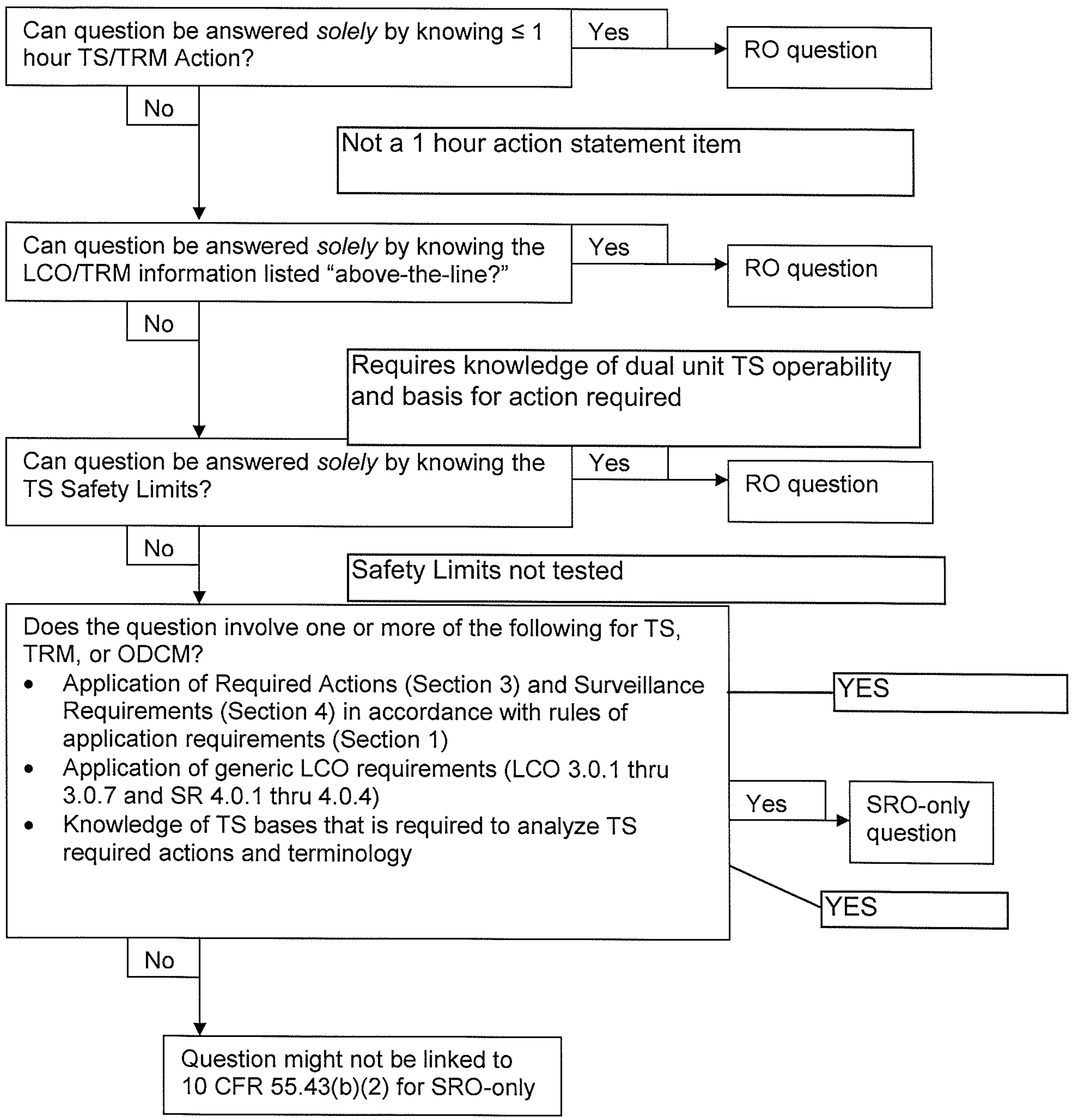
* CAUTION - If the opposite unit is in MODES 1. 2. 3, or 4 see Specification 3.8.1.1

** A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1.a and 4.8.1.1.2 (except for Specification 4.8.1.1.2a.5).

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



3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two startup transformers and their associated circuits, and

b. Three separate and independent diesel generators* including,

1) For Unit 3, two (3A and 3B); for Unit 4, one (3A or 3B) each with:

a) A separate skid-mounted fuel tank and a separate day fuel tank with an OPERABLE solenoid valve to permit gravity flow from the day tank to the skid mounted tank, and with the two tanks together containing a minimum of 2000 gallons of fuel oil.

b) A common Fuel Storage System containing a minimum volume of 38,000 gallons of fuel,**

c) A separate fuel transfer pump,**

d) Lubricating oil storage containing a minimum volume of 120 gallons of lubricating oil,

e) Capability to transfer lubricating oil from storage to the diesel generator unit, and

f) Energized MCC bus (MCC 3A vital section for EDG 3A, MCC 3K for EDG 3B).

2) For Unit 3, one (4A or 4B); for Unit 4, two (4A and 4B) each with:

a) A separate day fuel tank containing a minimum volume of 230 gallons of fuel,

b) A separate Fuel Storage System containing a minimum volume of 34,700 gallons of fuel,

c) A separate fuel transfer pump, and

d) Energized MCC bus (MCC 4J for EDG 4A, MCC 4K for EDG 4B).

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*Whenever one or more of the four EDG's is out-of-service, ensure compliance with the EDG requirements specified in Specifications 3.5.2 and 3.8.2.1.

**A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5, 6, or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days, Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

ATTACHMENT 1

TECHNICAL SPECIFICATION BASES

3/4.8.1, 3/4.8.2, and 3/4.8.3 (Cont'd)

Due to the shared nature of numerous electrical components between Turkey Point Units 3&4, the inoperability of a component on an associated unit will often affect the operation of the opposite unit. These shared electrical components consist primarily of both startup transformers, three out of four 4160 volt busses, and associated 480 volt motor control centers, all four 125 volt D.C. busses, all eight 120 volt vital A.C. panels and eight out of twelve vital A.C. inverters, four out of eight battery chargers, and all four battery banks. Depending on the components which is (are) determined inoperable, the resulting ACTION can range from the eventual shutdown of the opposite unit long after the associated unit has been shutdown (30 days) to an immediate shutdown of both units. Therefore, ACTION times allow for an orderly sequential shutdown of both units when the inoperability of a components affects both units with equal severity. When a single unit is affected, the time to be in HOT STANDBY is 6 hours. When an ACTION statement requires a dual unit shutdown, the time to be in HOT STANDBY is 12 hours. This is to allow the orderly shutdown of one unit at a time and not jeopardize the stability of the electrical grid by imposing a dual unit shutdown.

As each startup transformer only provides the limited equivalent power of approximately one EDG to the opposite Units A-train 4160 volt bus, the allowable out-of-service time of 30 days has been applied before the opposite unit is required to be shutdown. Within 24 hours, a unit with an inoperable startup transformer must reduce THERMAL POWER to less than or equal to 30% RATED THERMAL POWER. The 30% RATED THERMAL POWER limit was chosen because at this power level the decay heat and fission product production has been reduced and the operators are still able to maintain automatic control of the feedwater trains and other unit equipment. At lower power levels the operators must use manual control with the feedwater bypass lines. By not requiring a complete unit shutdown, the plant avoids a condition requiring natural circulation and avoids intentionally relying on engineered safety features for non-accident conditions.

With one startup transformer and one of the three required EDGs inoperable, the unit with the inoperable transformer must reduce THERMAL POWER to less than or equal to 30% RATED THERMAL POWER within 24 hours, based on the loss of its associated startup transformer, whereas operation of the unit with the OPERABLE transformer is controlled by the limits for inoperability of the EDG. The notification of a loss of startup transformers to the NRC (ACTION STATEMENT 3.8.1.1.c) is not a 10 CFR 50.72/50.73 requirement and as such will be made for information purposes only to the NRC Operations Center via commercial lines.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of two startup transformers or an associated circuit inoperable, demonstrate the OPERABILITY of the other startup transformer and its associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit is in MODE 1, reduce THERMAL POWER to ≤30% RATED THERMAL POWER within 24 hours, or restore the inoperable startup transformer and associated circuits to OPERABLE status within the next 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. If THERMAL POWER is reduced to ≤30% RATED THERMAL POWER within 24 hours or if the inoperable startup transformer is associated with the opposite unit, restore the startup transformer and its associated circuits to OPERABLE status within 30 days of the loss of OPERABILITY, or be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. If the inoperable startup transformer is the associated startup transformer and became inoperable while the unit was in MODE 2, 3, or 4 restore the startup transformer and its associated circuits to OPERABLE status within 24 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION applies to both units simultaneously.

b. With one of the required diesel generators inoperable, demonstrate the OPERABILITY of the above required startup transformers and their associated circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining required diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within 24 hours, unless the absence of any potential common mode failure for the remaining diesel generators is determined. If testing of remaining required diesel generators is required, this testing must be performed regardless of when the inoperable diesel generator is restored to OPERABILITY. Restore the inoperable diesel generator to OPERABLE status within 14 days** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. |

c. With one startup transformer and one of the required diesel generators inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a on the remaining

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** 72 hours if inoperability is associated with Action Statement 3.8.1.1.c. |

QUESTION 89

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	073	A2.02
Importance Rating		3.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question: SRO Question # 89

The following conditions exist:

- A Waste Gas Decay Tank E release was in progress.
- R-14, Plant Vent Gaseous Monitor, FAIL light illuminates with indication pegged low.

In accordance with 3-ONOP-067, Radioactive Effluent Release, which ONE of the following describes (1) the required action, if any is required, during a release when R-14, Plant Vent Gaseous Monitor, fails low and (2) in accordance with the Offsite Dose Calculation Manual (ODCM), any additional action(s) associated with this failure?

A. (1) Continue the release based on initial sample results
(2) Ensure R-14 is OPERABLE prior to releasing the next Waste Gas Decay Tank

B. (1) Continue the release based on initial sample results
(2) Ensure Chemistry performs TWO independent Waste Gas Decay Tank E samples every 4 hours during the release

C. (1) Terminate the release
(2) Ensure re-initiation of Waste Gas Decay Tank E ONLY when R-14 is OPERABLE

D. (1) Terminate the release
(2) Ensure Chemistry performs TWO independent Waste Gas Decay Tank E samples and TWO independent calculations prior to restarting the release

Proposed Answer: D

Explanation (Optional):

A. Incorrect due to the release MUST be terminated. Plausible – the applicant understands the tank was sample prior to release and nothing was added. Also, it is plausible to not release until R-14 is repaired.

B. Incorrect due to the release MUST be terminated. Plausible – the applicant understands the tank was sample prior to release and nothing was added. Also, it is plausible to require Chemistry to sample without a process radiation monitor in service. When a flow monitor is inoperable, there is a requirement to estimate flow every 4 hrs.

C. Incorrect due to release is allowed with special requirements. Plausible – The applicant assumes the only method to allow a release is with an operable process radiation monitor.

D. Correct. The release is required to be terminated per 3-ONOP-067 and the ODCM. Also, the ODCM Table 3.1-1 requires TWO independent: discharge valve lineups; Waste Gas Decay Tank E samples; AND release calculations prior to initiating a release with R-14 inoperable.

Technical Reference(s): ODCM Table 3.1-1
3-ONOP-067 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 6900150 EO12 (As available)

Question Source: Bank #
Modified Bank # WTSI 71329 (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010-2 Turkey Point

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41
55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance

activities and various contamination conditions.

Comments:

Reworded question and changed correct answer.

This item is SRO level because it meets the requirements of 10CFR55.43(b) 4 as shown below:

D. Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
[10 CFR 55.43(b)(4)]
Some examples of SRO exam items for this topic include:
□ Process for gaseous/liquid release approvals, i.e., release permits.
□ Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.
Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.
SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Step 8 is NOT applicable to a channel failure of R-11, R-12, R-15, or R-19.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8	**Check For PRMS Channel Failure** • Check Fail indicator - OFF • Display and recorder reading – **NOT** FAILED LOW	Perform the following: a. **IF** R-14 fails low **AND** a gas decay tank release is in progress, **THEN** stop the release. b. **IF** R-18 fails low **AND** a liquid release is in progress, **THEN** stop the release. c. Notify the Shift Manager to refer to Tech Specs **AND** take all required actions for the failed channel(s). d. Notify I&C of the PRMS failure.
9	**Check R11/12 RM-80 Green Monitor Light - ON**	Identify and correct the cause of failure using applicable Steps 18 through 28.
10	**Check If Effluent Radiation Monitors ALARMS - OFF**	
	a. Check the following radiation monitor alarms - OFF • RAD-3-6417 (SJAE SPING) • RAD-3-6426 (DAM-1 Monitor)	a. Go to 3-ONOP-071.2, STEAM GENERATOR TUBE LEAKAGE.
	b. Check RAD-6304 (Plant Vent SPING) alarm - OFF	b. Perform the following as applicable: 1) 4-ONOP-033.1, SFP COOLING SYSTEM MALFUNCTION 2) **IF** Steps 42 through 45 **NOT** previously performed, **THEN** go to Step 42.
	c. Check RAD-6418 (SFP Vent SPING) alarm - OFF	c. Perform 3-ONOP-033.3, ACCIDENT INVOLVING NEW OR SPENT FUEL.

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1 Radioactive Gaseous Effluent Monitoring Instrumentation, Operability and Alarm/Trip Setpoints, (Cont'd)

TABLE 3.1-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GAS DECAY TANK SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	3.1.1
b. Effluent System Flow Rate Measuring Device	1	*	3.1.2
2. Condenser Air Ejector Vent System			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	#	3.1.3
b. Iodine Sampler	1	##	3.1.6
c. Particulate Sampler	1	##	3.1.6
d. Effluent System Flow Rate Measuring Device		##	3.1.2
e. Sampler Flow Rate Measuring Device	1	##	3.1.5

3.0 RADIOACTIVE GASEOUS EFFLUENT

CONTROL 3.1: Radioactive Gaseous Effluent Monitoring Instrumentation; Operability and Alarm/Trip Setpoints, (Cont'd)

TABLE 3.1-1 (Cont'd)
TABLE NOTATION

* At all times.
# Applies during Mode 1, 2, 3, and 4.
Applies during Mode 1, 2, 3 and 4 when primary to secondary leakage is detected.

ACTION 3.1.1 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, **and**

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 3.1.2 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 3.1.3 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 3.1.4 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided continuous sample collection with auxiliary equipment as required by Table 3.2-1 is installed within 4 hours of the channel being declared inoperable, and analyzed at least weekly.

Facility:	Turkey Point		Original Item for question 89
Vendor:	WEC		
Exam Date:	2010-2		
Exam Type:	S		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	073	A2.02
	Importance Rating	2.7	3.2

Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure

Proposed Question:

- R-14, Plant Vent Gaseous Monitor, is inoperable.
- It is desired to release Waste Gas Decay Tank F.

In accordance with the ODCM Table 3.1-1, Radioactive Gaseous Effluent Monitoring Instrumentation, which ONE of the following completes both of the following statements?

The contents of Waste Gas Decay Tank F may be released to the environment provided that:

At least two independent samples of ___(1)___, and

At least two technically qualified members of the facility staff independently verify the (2)___

A: (1) Waste Gas Decay Tank F are analyzed prior to initiating the release
(2) release rate calculations only

B: (1) Waste Gas Decay Tank F are analyzed prior to initiating the release
(2) release rate calculations and discharge valve lineup

C: (1) Plant Vent Stack effluent are analyzed during the release
(2) release rate calculations only7

D: (1) Plant Vent Stack effluent are analyzed during the release

(2) release rate calculations and discharge valve lineup

Proposed Answer: B

Explanation (Optional):

A: Incorrect IAW above discussion. Plausible - this is part of the requirement

B: Correct IAW above discussion

C: Incorrect IAW above discussion. Plausible - R-14 monitors the effluent and this is part of the requirement

D: Incorrect IAW above discussion. Plausible - R-14 monitors the effluent

Technical Reference(s):

1. ODCM Table 3.1-1 rev. 12/29/09
2. 0-NOP-061.14F attachment 1 rev. 0
3. 0-NCOP-004 steps 7.1.2 & 7.1.21 rev. 10/5/07

Required to perform action 3.1.1, which requires at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup. IAW NOP-061.14F, the SNPO performs the lineup. IAW NCOP-004, Chemistry samples and performs the release rate calculation. Action 3.1.3 is not required

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: Yes

Learning Objective: 6900150 EO12 (As available)

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History:

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	

10 CFR Part 55 Content:	55.41
	55.43

Comments:

QUESTION 90

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	076	A2.01
Importance Rating		3.7

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS

Proposed Question: SRO Question # 90

The following conditions exist:

- Unit 3 is at 100%.
- 3C ICW Pump out of service.
- Traveling Screen D/P is 8.0" H_2O.
- ICW/CCW and ICW/TPCW Basket Strainer clogging due to grass influx.
- CCW Heat Exchanger Outlet temperature is currently 104°F and stable.
- 3A and 3B TPCW Heat Exchangers - ICW flows are at 4200 gpm each.
- 3A, 3B, and 3C CCW Heat Exchangers - ICW flows are at 3000 gpm each.
- Annunciator E 2/2, TURB BEARING HI TEMP, is in alarm.
- Turbine Bearing temperatures are 171°F and slowly increasing after TPCW flow has been adjusted.

Which ONE of the following describes the procedure mitigation strategies in accordance with 3-ONOP-019, Intake Cooling Water Malfunction?

A. Trip the reactor and turbine in accordance with 3-ONOP-019, Intake Cooling Water Malfunction, Foldout Page and enter 3-EOP-E-0, Reactor Trip or Safety Injection.

B. Reduce Turbine load to maintain temperatures and continue with 3-ONOP-019. Implement, 3-ONOP-011, Screen Wash System/Intake Malfunction in parallel.

C. Continue to increase cooling water flow to the Turbine Oil Coolers, then implement 3-ONOP-008, Turbine Plant Cooling Water Malfunction.

D. Shutdown the unit using 3-GOP-100, Fast Load Reduction, within the next 6 hours due to INOPERABLE ICW Headers.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Turbine Trip setpoint is being approached but has not been exceeded on high bearing temperature.

B. Correct.

C. Incorrect. Cooling water flow is sufficient and transition is logical for current plant conditions of temperatures rising

D. Incorrect. This would be performed if ICW flow to a CCW heat exchanger was less than minimum for less than 5 minutes as per FOP criteria

Technical Reference(s): 3-ONOP-019 (Attach if not previously provided)

3-ARP-097.E E2/2

Proposed References to be provided to applicants during examination: None

Learning Objective: 6902277 Obj 7 (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPK)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

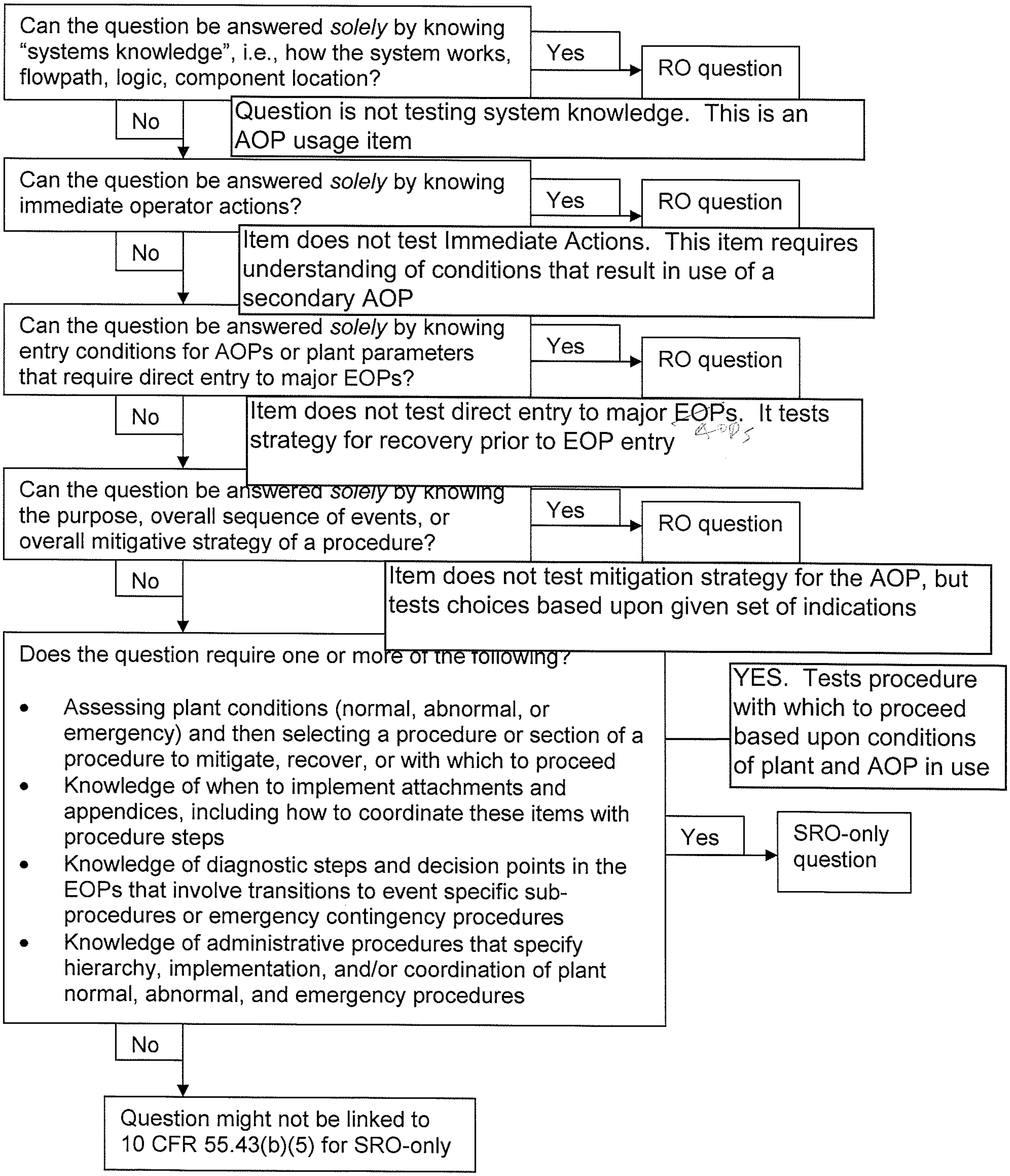
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Mod from ILC 25 #53

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



CAUSES:
1. Insufficient oil to bearing
2. Insufficient cooling water to lube oil cooler
3. Bearing failure

E11

TURB BEARING HI TEMP

DEVICE:	SETPOINT:	LOCATION:
Bearing drain temp elements	170°F	N/A

ALARM CONFIRMATION

NOTE

TSA 09-008 turned off the Channel 3 input (TE-3-3423 - No. 3 bearing oil drain) for recorder R-3-340, such that indication and alarm capability for this input has been disabled.

1. **CHECK** bearing temperatures on recorder R-340 on VPA.

OPERATOR ACTIONS

1. IF all bearings temps are increasing, THEN **DISPATCH** operator to check lube oil temperature using local indication at TC-3-2200.
2. **DETERMINE** method of reducing Turbine Lube Oil Temperature:
   - **ADJUST** TPCW flow as indicated on FI-3-1410 by adjusting hand loader for CV-3-2200 and **MONITOR** for temperature change. IF CV-3-2200 does **NOT** respond as desired, THEN **DISPATCH** operator to manually position 3-60-116, CV-2200 Bypass, as required.
   - **ADJUST** ICW flow to TPCW heat exchangers by throttling TPCW HX combined ICW Outlet Isolation Valve, 3-50-401 or CCW/ICW Outlet Spool Piece Downstream Isolation, 3-50-407. **CHECK** ICW flows within specifications after adjusting.
   - **CHECK** H/A-3-2203 is operating in HAND with proper flow through CV-3-2203.

CORRECT

3. IF lube oil temperature remains high after adjusting cooling water flow, THEN **REDUCE** load as necessary to return temperature to normal.
4. IF alarm is associated with a TPCW malfunction, THEN **REFER TO** 3-ONOP-008, Turbine Plant Cooling Water Malfunction.
5. IF one bearing temp is high, THEN **PERFORM** the following:
   A. **DISPATCH** operator to locally check temp.
   B. **MONITOR** bearing temp closely.

   CAUTION

   Do **NOT** lower lube oil temp to less than 100°F.

   C. **INCREASE** cooling water flow to lube oil cooler to lower bearing temp.
   D. IF bearing vibration starts to **INCREASE**, THEN **STOP** lowering lube oil temp.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTIONS

- *If the cause of the Intake Cooling Water Malfunction is determined to be due to high differential pressure on the traveling screens, then 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION, should be used.*
- *If an Intake Cooling Water Pump is stopped in this procedure and the reason for stopping the pump has not been corrected, that pump is not available for starting in subsequent procedure steps.*
- *Monitoring Main Generator RTDs is required if TPCW flow or temperature is changed due to the effect on Main Generator hydrogen leakage. An increase in hydrogen leakage is expected if the gas temperature to rotor temperature gradient increases. (Reference CR 2008-803)*

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	**Verify All Intake Cooling Water Pump Alarms - OFF** • I 4/1, ICWP A/B/C MOTOR OVERLOAD • I 4/2, ICWP A/B/C TRIP • I 4/3, ICWP A/B/C MOTOR BRG HI TEMP	Perform the following: 1. Have operator check pump(s) locally 2. Determine affected intake cooling water pump. 3. Start standby intake cooling water pump. 4. Stop affected intake cooling water pump.
2	**Check Traveling Screens - CLEAN** • Alarm I 3/3, Traveling Screen HI ΔP - OFF • Traveling Screen DP - LESS THAN 7.5 INCHES OF WATER	Go to 3-ONOP-011, SCREEN WASH SYSTEM/INTAKE MALFUNCTION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	**Maintain Cooling For Turbine Plant Cooling Water Heat Exchangers:** a. Check alarm I 5/4, TPCW HI TEMP/LO PRESS-OFF b. Locally check Turbine Plant Cooling Water Supply Header Temperature, TI-3-1432 - LESS THAN 110°F c. Locally check Turbine Plant Cooling Water Supply Header Temperature, TI-3-1432 - STABLE OR DECREASING	Perform the following: 1. Remove reactive load from main generator. 2. Decrease turbine load as necessary to maintain Turbine Plant Cooling Water Supply Header Temperature less than 110°F. 3. **IF** any component cooled by turbine plant cooling water is overheating, **THEN** restore cooling to component using 3-ONOP-008, TURBINE PLANT COOLING WATER MALFUNCTION, while continuing with this procedure.
11	**Maintain Cooling For Component Cooling Water Heat Exchangers:** a. Check alarm H 8/5, CCW HX OUTLET HI TEMP-OFF b. Check Component Cooling Water Supply Header Temperatures - LESS THAN 120°F • TI-3-607A • TI-3-607B c. Check Component Cooling Water Supply Header Temperatures - STABLE OR DECREASING • TI-3-607A • TI-3-607B	**IF** component cooling water temperature can **NOT** be maintained, **THEN** restore component cooling water temperatures using 3-ONOP-030, COMPONENT COOLING WATER MALFUNCTION while continuing with this procedure.

FOLDOUT PAGE FOR 3-ONOP-019

1. **TRIP CRITERIA**

   - Component Cooling Water temperature as read on TI-3-607A and TI-3-607B cannot be maintained less than 120°F.

   - Turbine or Generator bearing temperatures cannot be maintained less than 180°F.

2. **MINIMUM FLOW REQUIREMENTS FOR CCW HXs**

   While isolating a CCW/ICW strainer, ICW flow less than minimum required through the CCW HXs can be tolerated without entry into Technical Specification Action 3.0.3, provided flow is restored to the minimum allowable, as determined by 3-NOP-019, Intake Cooling Water System, in less than 5 minutes by reopening the strainer isolation valves. If flow is below the minimum allowable value for greater than 5 minutes, then entry into Technical Specification Action 3.0.3 is started at the point where flow first fell below the minimum value. [Reference 3.1.4]

Facility:	Turkey Point
Vendor:	WEC
Exam Date:	2003
Exam Type:	R

Original Item for Question 90

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	2
	Group #	1	1
	K/A #	076	A2.01
	Importance Rating	3.5	3.7

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS

Proposed Question:

Unit 3 is at 100% power with the '3C' ICW Pump out of service. Annunciator E-2/2, TURB BEARING HI TEMP, is in alarm.

The SNPO reports a massive grass influx has resulted in ICW/CCW and ICWITPCW basket strainer clogging.

The following conditions exist:

- Component Cooling Water heat exchanger outlet temperature is currently 118 °F and stable.
- Turbine bearing temperatures are 181°F and slowly increasing.
- '3A' and '3B' TPCW heat exchangers are at 4200 gpm ICW flow.
- '3A', '3B', and '3C' CCW heat exchangers are at 3000 gpm ICW flow each.

Which ONE of the following describes the actions that should be taken due to the above conditions?

A: Trip the reactor and turbine and enter EOP-E0, Reactor Trip or Safety Injection.

B: Reduce turbine load as necessary to return temperatures within normal bands and implement ONOP-011, Screen Wash System/Intake Malfunction

C: Increase cooling water flow to the turbine lube oil cooler to reduce bearing

temperatures and implement ONOP-011, Screen Wash System/Intake Malfunction.

D: Enter into Technical Specification 3.0.3 and commence a reactor shutdown per GOP-103, Power Operation to Hot Standby

Proposed Answer: A

Explanation (Optional):

A: Correct; Conditions on the turbine bearing are degrading and the Annunciator E-2/2 trip criteria for has been met due to bearing temperatures unable to be maintained below 180 °F.

B: Incorrect; Correct action if temperature problem was with the TPCW and maintainable below 180°F.

C: Incorrect; This is the action to reduce the turbine bearing temperature and should be initiated if the temperature were able to be maintained less than the trip criteria.

D: Incorrect, Correct for total flow of ICW dropping below minimum flow rate of 12,400 gpm for more than **5** minutes.

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # X (Note changes or attach parent)

New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	

Comments:

QUESTION 91

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	014	A2.02
Importance Rating		3.6

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: Loss of power to the RPIS

Proposed Question: SRO Question # 91

The following conditions exist:

- Unit 3 is operating at 100% power.
- Annunciator F-4/6, RPIS POWER TROUBLE is lit.
- Unit 3 Turbine Operator reports the RPI Inverter normal power supply breaker is open.

Which ONE of the following (1) predicts what Control Rod/Group Demand Position indication that is lost including the method to restore power in accordance with 3-NOP-028.01, Rod Position Power Supply and (2) determines the required Technical Specification ACTION?

A. (1) Analog Rod Position Indications are extinguished until RPI power is manually transferred to the CVT

(2) Reenergize the RPIS Loads within 2 hours OR be in at least HOT STANDBY in next 12 hours

B. 1) Group Step Counter Demanded Position Indicators have lost power until re-energized by the normal power supply

(2) Reenergize the RPIS Loads within 2 hours OR be in at least HOT STANDBY in next 12 hours

C. (1) Analog Rod Position Indications are extinguished until RPI power is manually transferred to the CVT

(2) Reenergize the RPIS Loads within 1 hour OR be in at least HOT STANDBY in next 6 hours

D. (1) Group Step Counter Demanded Position Indicators have lost power until re-energized by the normal power supply

(2) Reenergize the RPIS Loads within 1 hour OR be in at least HOT STANDBY in next 6 hours

3.0.3.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because ALL Analog Rod Position Indications are extinguished until RPI power is manually transferred to the CVT is a correct statement. Also, the second part is plausible because 2 hours is the amount of time used for an out of service vital inverter.

B. Incorrect since the Group Step Counter Demanded Position Indicators maintain power. Plausible because it is logical with Annunciator RPIS POWER TROUBLE to assume they have lost power. Also, the second part is plausible because 2 hours is the amount of time used for an out of service vital inverter.

C. Correct. ALL Analog Rod Position Indications are extinguished until RPI power is manually transferred to the CVT. The unit is in a 3.0.3 condition.

D. Incorrect since the Group Step Counter Demanded Position Indicators maintain power. Plausible because it is logical with Annunciator RPIS POWER TROUBLE to assume they have lost power. Also, the second part is plausible because it is correct.

Technical Reference(s): TS 3.1.3.2, TS LCO 3.0.3 (Attach if not previously provided)
3-ARP-097.CR.F, F-4/9
3-NOP-028.01

Proposed References to be provided to applicants during examination: None

Learning Objective: 6900106, Obj 8c (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam:

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	(2DR)
Question Difficulty Level: B		
10 CFR Part 55 Content:	55.41	
	55.43 2	

Facility operating limitations in the technical specifications and their bases.

Comments:

F49

RPIS POWER TROUBLE

CAUSES:
1. Low AC voltage from CVT
2. Low DC input voltage
3. Static switch transfer to CVT
4. Manual Bypass Sw in Alt Source
5. Inverter off frequency
6. AC output Line 1 Ground
7. AC output Line 2 Ground
8. Inverter fan failure
9. Inverter out of synch
10. Inverter high temperature

DEVICE: Alarm relays

SETPOINT: N/A

LOCATION: In RPI inverter cabinet

ALARM CONFIRMATION

1. **CHECK** the following:
   - Loss of RPI indication on console
   - Inaccurate RPI indication on console

OPERATOR ACTIONS

1. **ENSURE** auto transfer from normal RPI power supply inverter to CVT upon inverter failure.
2. **DIRECT** operator to check RPI inverter.
3. **CHECK** power supply breakers:
   - 3D01-49
   - 30766
4. IF necessary, THEN **TRANSFER** RPI power supply using 3-NOP-028.01, RPI Power Supply System.
5. **ENSURE** RPI positions on console meet acceptance criteria contained in 3-OSP-201.1, RO Daily Log.
6. **REFER TO** T.S. 3.1.3 or 3.10 (as applicable) for additional actions.

REFERENCES:
1. FPL EWD 5613-E-28, Sh 94B
2. Tech Spec Sections 3.1.3 and 3.10

5.2 <u>Manual Transfer from RPI Inverter to CVT Bypassing Static Transfer Switch</u> **<u>INITIAL</u>**

1. At the RPI INVERTER, **PERFORM** the following:

A. **CHECK** the following:

- ALTERNATE SOURCE voltage is 115 to 125 VAC
- IN SYNC light ON

Concurrent Verification

B. **PRESS** the ALTERNATE SOURCE TO LOAD button. ______

______ CV

C. **CHECK** the following:

- ALTERNATE SOURCE SUPPLYING LOAD light ON
- INVERTER SUPPLYING LOAD light OFF

D. **OPEN** the INVERTER OUTPUT breaker.

Concurrent Verification

E. **PLACE** MANUAL BYPASS SWITCH to ALTERNATE SOURCE TO LOAD position. ______

______ CV

F. **OPEN** the DC INPUT breaker.

G. **CHECK** the INVERTER OUTPUT voltage is 0 VAC.

H. **OPEN** ALTERNATE SOURCE breaker.

I. **CHECK** ALTERNATE SOURCE voltage is 0 VAC.

2. **COMPARE** the RPI positions against the acceptance criteria contained in 3-OSP-201.1, RO Daily Logs.

AC Supply rectified to DC for RPI module use.

LVDT mounted on the CRDM rod travel housing

o 45 rod position detectors

o Dedicated 5KVA inverter from DC Bus 3D01 (4D01) all power from 3C(4B) MCC via CVT.

"Step Counters"

digital count the actuating pulse which displays the bank demand position

12, 3 digit Step Counters

→ RIL Computer

3.1.3.2 The Analog Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the respective actual and demanded shutdown and control rod positions as follows:

a. Analog rod position indicators, within one hour after rod motion (allowance for thermal soak);

<u>All Shutdown Banks</u>: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal ranges of 0-30 steps and 200-All Rods Out as defined in the Core Operating Limits Report.

<u>Control Bank A and B</u>: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal ranges of 0-30 steps and 200-All Rods Out as defined in the Core Operating Limits Report.

<u>Control Banks C and D</u>: within the Allowed Rod Misalignment of Specification 3.1.3.1 of the group demand counters for withdrawal range of 0-All Rods Out as defined in the Core Operating Limits Report.

b. Group demand counters; ± 2 steps.

<u>APPLICABILITY</u>: MODES 1 and 2.

<u>ACTION</u>:

a. With a maximum of one analog rod position indicator per bank inoperable either:

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2** a). Determine the position of the non-indicating rod indirectly by the movable incore detectors within 8 hours and once every 31 Effective Full Power Days thereafter, and within 1 hour if rod control system parameters indicate unintended movement, or if the rod with an inoperable position indicator is moved greater than 12 steps, and

b). Review the parameters of the rod control system for indications of unintended rod movement for the rod with an inoperable indicator within 8 hours and once per 8 hours thereafter, and

c). Determine the position of the non-indicating rod indirectly by the movable incore detectors prior to increasing THERMAL POWER above 50% RATED THERMAL POWER and within 8 hours of reaching 100% RATED THERMAL POWER, or

3. Reduce THERMAL POWER to less than 75% of RATED THERMAL POWER within 8 hours.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued):

b. With a maximum of one demand position indicator per bank inoperable either:

1. Verify that all analog rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within the Allowed Rod Misalignment of Specification 3.1.3.1 at least once per 8 hours, or

2. Reduce THERMAL POWER to less than 75% of RATED THERMAL POWER within 8 hours.

**Rod position monitoring by Actions a.2.a), a.2.b), and a.2.c) may only be applied to one inoperable rod position indicator per unit and shall only be allowed until an entry into MODE 3.

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met, except as provided in Specification 3.0.6.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals, except as provided in Specification 3.0.6. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODES 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires a shutdown if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or specified condition may be made in accordance with ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

QUESTION 92

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	068	A2.04
	Importance Rating		3.3

Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation

Proposed Question: SRO Question # 92

Given the following conditions:

- A Liquid Release is in progress from Recycle Monitor Tank A to Discharge Canal using Monitor Tank Pump A.
- Annunciator WASTE LIQUID HI RADIATION (WB.B 5/3) is received from R-18, Waste Disposal System Liquid Effluent Monitor.
- RCV-018, Liquid Waste Discharge Isolation Valve, fails to close automatically due to the high radiation signal and CANNOT be closed manually from the Waste Boron Recycle Panel.

Which ONE of the following identifies (1) the NEXT action that should be taken to mitigate the situation in accordance with 0-NOP-061.11A, Controlled Liquid Release from Recycle Monitor Tank A, and (2) the Regulatory impact of this condition?

A. (1) Stop the Monitor Tank Pump A
(2) The continued release may have exceeded the Co-60 effluent limits of the Offsite Dose Calculation Manual (ODCM).

B. (1) Stop the Monitor Tank Pump A
(2) The continued release may have exceeded the tritium effluent limits of the National Pollutant Elimination Discharge System (NPDES).

C. (1) Close 1282, Monitor Tank A Outlet Valve
(2) The continued release may have exceeded the Co-60 effluent limits of the Offsite Dose Calculation Manual (ODCM).

D. (1) Close 1282, Monitor Tank A Outlet Valve
(2) The continued release may have exceeded the tritium effluent limits of the National Pollutant Elimination Discharge System (NPDES).

Proposed Answer: A

Explanation (Optional):

A. CORRECT. The ODCM limits are the Regulatory concern.

Per 0-NOP-061.11A, Page 25:

17. IF at any time, R-18 count rate exceeds the Warning Limit, THEN:
A. STOP Recycle Monitor Tank Pump used for release.
B. GO TO Attachment 1 Step 19 to terminate the release.

B. Incorrect. Plausible because first part is correct. Also, plausible because it could be thought that tritium content of the tank may have exceeded the limits of the NPDES for Tritium, thereby creating a regulatory issue.

C. Incorrect since the NEXT step is to stop the Monitor Tank Pump, not close tank outlet. Plausible because second part is correct and closing the valve may stop the discharge, but could damage the pump.

D. Incorrect since the NEXT step is to stop the Monitor Tank Pump, not close tank outlet. Plausible because second part is correct and closing the valve may stop the discharge, but could damage the pump. Also, plausible because it could be thought that tritium content of the tank may have exceeded the limits of the NPDES for Tritium, thereby creating a regulatory issue.

Technical Reference(s): 0-NOP-061.11A, Controlled Liquid Release from Recycle Monitor Tank A (Attach if not previously provided)

ODCM

Proposed References to be provided to applicants during examination: None

Learning Objective: LP 6902242, Obj. 7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

This item meets requirements of 10CFR55.43(b) 4 based upon the criteria below:

D. Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
[10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

□ Process for gaseous/liquid release approvals, i.e., release permits.

□ Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

2.0 RADIOACTIVE LIQUID EFFLUENTS

OBJECTIVES & SYSTEM DESCRIPTION

A. Objectives

To provide calculational methodology needed to assure compliance with 10CFR20 and 10CFR50, which require the following determinations and surveillance:

- o The concentration of radioactive materials released in liquid effluents.
- o The concentrations of radioactive materials released are maintained within the limits of Control 2.2.
- o Quarterly and annual cumulative dose contributions to a member of the public from radioactivity in liquid effluents released from each unit to unrestricted areas are maintained within the limits of Control 2.3.
- o Projected doses due to liquid releases to unrestricted areas are maintained within the limits of Control 2.4.
- o Operation of appropriate portions of the Liquid Radwaste Treatment System when projected doses exceed limits of Control 2.4.
- o The operability of Liquid Radwaste System is verified by meeting Controls 2.2 and 2.3

B. Bases

Radioactive liquid effluents from Turkey Point Units 3 and 4 are released through radiation monitors which provide an alarm and automatic termination of radioactive releases. There are three discharge points from the units: steam generator blowdown from each unit and a common radwaste monitor tank discharge. The liquid effluent monitoring instrumentation and controls at Turkey Point for controlling and monitoring normal radioactive releases in accordance with Turkey Point Technical Specification 6.8.4.f consist of the following :

1 Liquid Radwaste System

Potentially radioactive liquid waste from Units 3 and 4 Chemistry laboratories, containment sumps, floor drains, showers and miscellaneous sources are collected in waste hold up tanks. These wastes are processed through a demineralizer system and the effluent stored in one of the three waste monitor tanks or one of two monitor tanks (Refer to Figure 2-1). Laundry wastes are normally segregated and sent to one of two monitor tanks

2.0 RADIOACTIVE LIQUID EFFLUENTS

OBJECTIVES & SYSTEM DESCRIPTION (continued)

Bases, Liquid Radwaste System (continued)

5. Radioactivity Concentration in Water at the Restricted Area Boundary

   Control 2.2 requires that the concentration of radioactive material, other than noble gases, in liquid effluent released into an unrestricted area not exceed 10 times the effluent concentration specified in 10 CFR Part 20, Appendix B, Table 2, Column 2.

   A maximum concentration, 2×10^{-4} μCi/ml, for noble gas entrained in aqueous releases into an unrestricted area applies separately since the potential exposure route, immersion in water, differs from that upon which Part 20, Appendix B is based.

   Radioactive material in liquid effluent from Turkey Point is diluted by condenser cooling water from fossil units 1 and 2 and from nuclear units 3 and 4 in the condenser cooling water mixing basin. Water in the basin flows into an on site closed cooling canal system. Liquid effluent does not actually leave the site in a surface discharge. For the purpose of compliance with Control 2.2, the total condenser cooling water flow from operating condenser cooling water pumps at the four units is assumed for dilution and the restricted area boundary is assumed to be at the end of the condenser cooling water mixing basin where water enters the cooling canal system.

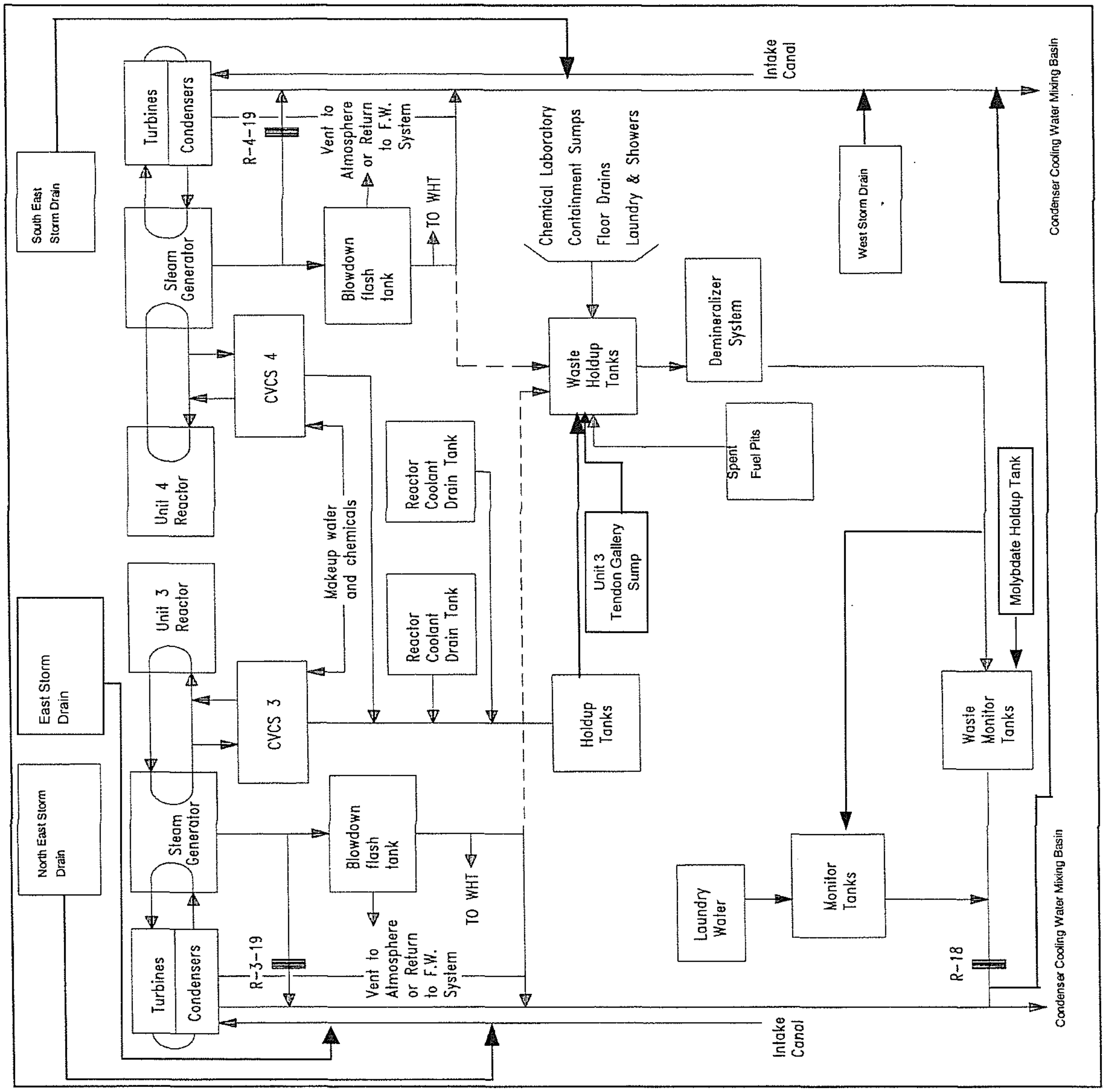
   Methods 2.2.1 and 2.2.2 describe methods used to assess compliance with Control 2.2. Effluent monitor alarm/trip setpoints are computed on the same basis as described in Methods 2.1.1 and 2.1.2. If an alarm/trip setpoint is not exceeded, aqueous effluents are deemed to comply with Control 2.2.

   The operability of the Liquid Radwaste System is considered verified by virtue of meeting Controls 2.1 and 2.2. Normally, batch releases from the Laundry, and continuous releases from Steam Generator Blowdown are not processed. When necessary, the Laundry and the Steam Generator Blowdown can be diverted to the Liquid Radwaste System for processing.

2.0 RADIOACTIVE LIQUID EFFLUENTS

FIGURE 2-1

RADIOACTIVE LIQUID WASTE



ATTACHMENT 1
Controlled Liquid Release from Recycle Monitor Tank A with R-18 OPERABLE
(Page 7 of 10)

NOTE

An R-18 Warning Limit may be exceeded due to a miscalculation on the Liquid Release Permit, an incorrect thumbwheel adjustment, rapid manipulation of Valve 1296, or other factors associated with high activity. All data should be evaluated prior to the re-issuance of the Liquid Release Permit.

14. **REQUEST** RCO monitor R-18 to ensure both the following:
   - R-18 responds to the liquid release.
   - R-18 count rate does **NOT** exceed the Warning Limit.

NOTE

Release time for Recycle Monitor Tanks should be approximately 2 hours.

15. **MONITOR** waste tank levels to ensure Recycle Monitor Tank A level lowers, and all other waste tank levels **NOT** changing unexpectedly.

NOTE

Valve 1279, MONITOR TANK PUMP RECIRC TO MONITOR TANK A STOP, throttled ½ turn OPEN, should prevent Recycle Monitor Tank Pump cavitation, however, if the pump exhibits cavitation characteristics, 1279 should be throttled while maintaining a recirculation flowpath.

16. **MONITOR** operating Recycle Monitor Tank Pump for cavitation.
   A. IF cavitation is evident, THEN **ADJUST** 1279, MONITOR TANK PUMP RECIRC TO MONITOR TANK A STOP position, while maintaining a recirculation flowpath.

17. IF at any time, R-18 count rate exceeds the Warning Limit, THEN:
   A. **STOP** Recycle Monitor Tank Pump used for release.
   B. **GO TO** Attachment 1 Step 19 to terminate the release.

18. WHEN Recycle Monitor Tank A level lowers to between 15% and 10%, THEN **STOP** Recycle Monitor Tank Pump used for release.

ATTACHMENT 1
Controlled Liquid Release from Recycle Monitor Tank A with R-18 OPERABLE
(Page 8 of 10)

INITIAL

19. **RECORD** the following on the Radioactive Liquid Release Permit:
    - Stop time
    - Tank level
    - Required R-18 count rate data

20. **CLOSE** and **LOCK** 4749, STOP VALVE BEFORE RCV-018 WASTE TO DISCHARGE. ______

    ______ IV

21. **CLOSE** RCV-018, WASTE DISCHARGE LINE VALVE. ______

    ______ IV

22. **CLOSE** 1296, MONITOR TANK PUMP TO WASTE DISPOSAL SYSTEM. ______

    ______ IV

23. **ENSURE** Recycle Monitor Tank Pump used for release, STOPPED.

24. **CLOSE** 1279, MONITOR TANK PUMP RECIRC TO MONITOR TANK A STOP.

25. **ENSURE** 1282, MONITOR TANK A OUTLET VALVE, CLOSED.

26. **ENSURE** 1303, MONITOR TANK CROSS CONNECTION, CLOSED.

27. **ENSURE** 1290, MONITOR TANK PUMP A DISCHARGE, OPEN.

QUESTION 93

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	017	2.4.3
	Importance Rating		3.9

Emergency Procedures / Plan: Ability to identify post-accident instrumentation.

017 – Incore Temperature Monitoring System

Proposed Question: SRO Question # 93

Unit 3 is in MODE 3 during a reactor startup.

- Core Exit Thermocouples from B QSPDS Train are out of service.
- The Train A QSPDS readings are provided on the DCS printout from QSPDS CET/HJTC Channel A display.

Which ONE of the following describes (1) the ACTION(s) required for Core Exit TCs in accordance with TS 3.3.3.3, Accident Monitoring Instrumentation, and (2) the impact to the reactor startup?

REFERENCES PROVIDED

A. (1) ACTION is required by Action Statement 31 of TS 3.3.3.3, Accident Monitoring Instrumentation
(2) The startup may NOT continue. Unit 3 shall remain in MODE 3.

B. (1) ACTION is required by Action Statement 32 per TS 3.3.3.3, Accident Monitoring Instrumentation
(2) The startup may NOT continue. Unit 3 shall remain in MODE 3.

C. (1) ACTION is required by Action Statement 32 per TS 3.3.3.3, Accident Monitoring Instrumentation
(2) The startup may continue and Reactor Power may be raised to 100% power.

D. (1) ACTION is required by Action Statement 31 per TS 3.3.3.3, Accident Monitoring Instrumentation
(2) The startup may continue and Reactor Power may be raised to 100% power.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausible because this is the correct Action Statement. On Part 2, the student believes a Mode change cannot be made while in an Action Statement which is a true fact if SR 3.04 is not applicable.

B. Incorrect. Plausible because this Action Statement applies to out of service CETs. However, this is an incorrect Action Statement. On Part 2, the student believes a Mode change cannot be made while in an Action Statement which is a true fact if SR 3.04 is not applicable.

C. Incorrect. Plausible because this Action Statement applies to out of service CETs. However, this is an incorrect Action Statement. On Part 2, the student believes a Mode change cannot be made while in an Action Statement which is a true fact if SR 3.04 is not applicable.

D. Correct.

Technical Reference(s): TS 3.3.3.3 amendments 227/223 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS 3.3.3.3
QSPDS DCS Printout

Learning Objective: 6902103 Obj 10a (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2RI)

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:
SRO because TS action on Mode change is ‘below the line’ knowledge exclusive to SRO

Q # 93

System | Process

DATE – TODAY

TIME - NOW

PRIMARY
SECONDARY
POWER
ESF
SUPPORT SYS
EMERG RESP
UTILITIES
LEGEND
ALARMS
TRENDS
PRINT
PREVIOUS

PTN	QSPDS CET/HJTC CHANNEL A
UNIT 3	

QSPDS MENU

HEATED JUNCTION THERMOCOUPLES

HEATER POWER		THERMOCOUPLE TEMPERATURES			
HEATER	%	SENSOR	HEATED °F	UNHEATED °F	DIFFERENTIAL °F
1	80	1	595	544	50
2	80	2	601	548	52
3	80	3	595	544	51
4	80	4	601	545	56
5	80	5	607	546	61
6	80	6	613	547	66
7	80	7	619	547	71
8	80	8	625	548	77

CALCULATED CET DATA

HIGHEST TEMPERATURE

QUAD	ID	°F
1	3	546
2	5	547
3	4	547
4	3	546

NEXT HIGHEST TEMPERATURE

QUAD	ID	°F
1	4	546
2	2	546
3	1	546
4	4	546

FIVE HIGHEST CET TEMPERATURES

	QUAD	ID	°F
1	2	5	547
2	3	4	547
3	1	3	546
4	1	4	546
5	1	5	546

CORE EXIT THERMOCOUPLES

QUADRANT 1

CET	LOC	°F
1	P7	2300
2	N10	2300
3	N8	546
4	L6	546
5	K8	546

QUADRANT 2

CET	LOC	°F
1	M3	2300
2	H5	546
3	H3	2300
4	G2	546
5	E4	547
6	D3	2300

QUADRANT 4

CET	LOC	°F
1	L14	2300
2	L12	2300
3	J12	546
4	J10	546
5	H11	546
6	G15	546
7	F13	546
8	F11	546

QUADRANT 3

CET	LOC	°F
1	G8	546
2	E10	2300
3	E7	2300
4	D5	547
5	C12	546
6	C8	2300
7	A8	546

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS

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TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermo-couples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
d. Main Steam Lines	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 38 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, restore the inoperable channel(s) to OPERABLE status within 7 days. If repairs are not feasible without shutting down:

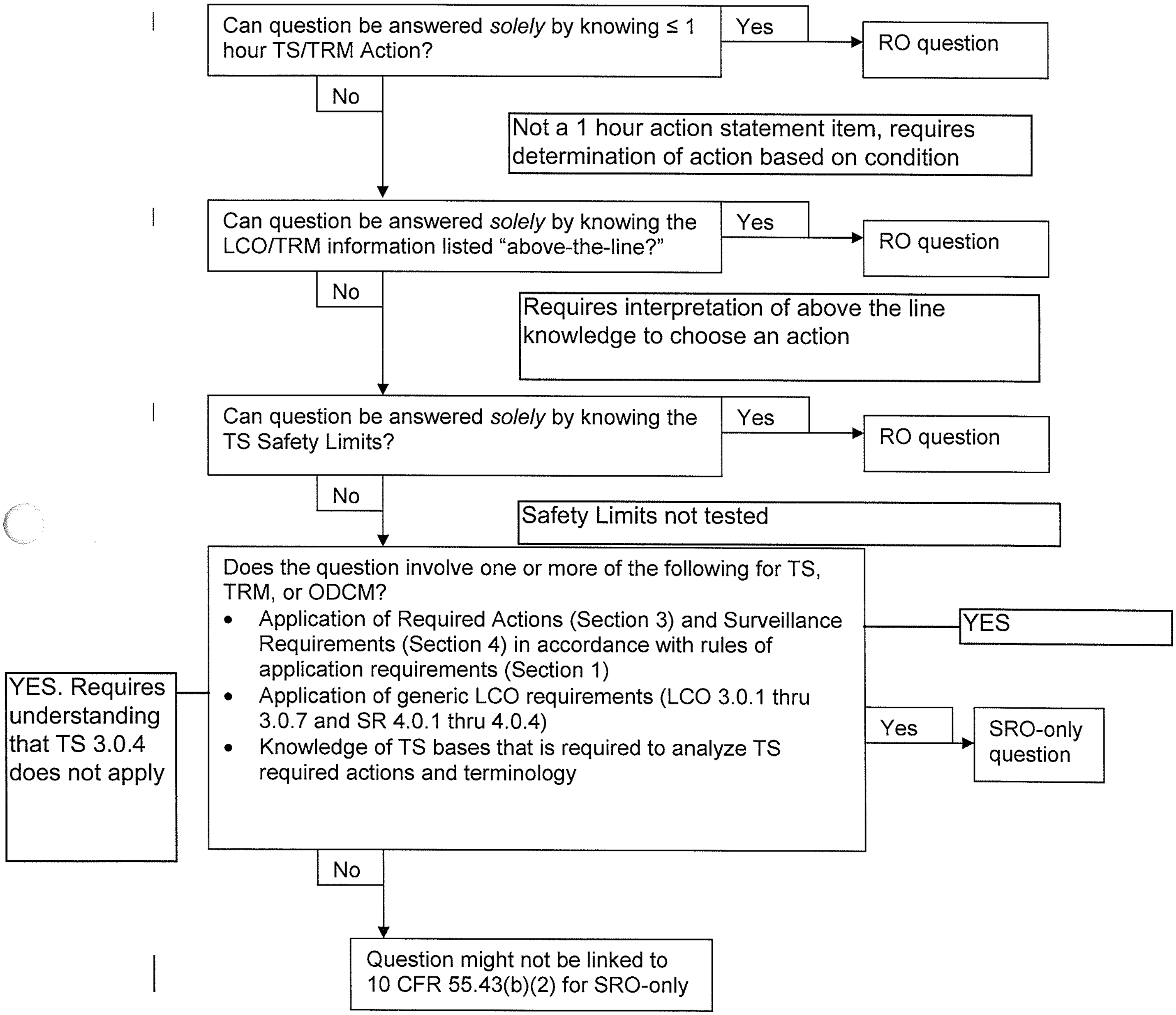
1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore at least one channel to OPERABLE status at the next scheduled refueling.

ACTION 39 With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, verify position by an alternate means (e.g. administrative controls, ERDADS, alternate position indication, or visual observation) within 2 hours, and restore the inoperable channel(s) within 7 days, or comply with the provisions of Specification 3.6.4 for an inoperable containment isolation valve.

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



Plant Computer (DCS) – Simulated QSPDS Display

System | Process

DATE – TODAY

TIME - NOW

PRIMARY | SECONDARY | POWER | ESF | SUPPORT SYS | EMERG RESP | UTILITIES | LEGEND | ALARMS | TRENDS | PRINT | PREVIOUS

PTN UNIT 3

QSPDS CET/HJTC CHANNEL A

QSPDS MENU

HEATED JUNCTION THERMOCOUPLES

HEATER POWER

HEATER	%
1	80
2	80
3	80
4	80
5	80
6	80
7	80
8	80

THERMOCOUPLE TEMPERATURES

SENSOR	HEATED °F	UNHEATED °F	DIFFERENTIAL °F
1	595	544	50
2	601	548	52
3	595	544	51
4	601	545	56
5	607	546	61
6	613	547	66
7	619	547	71
8	625	548	77

CALCULATED CET DATA

HIGHEST TEMPERATURE

QUAD	ID	°F
1	3	546
2	5	547
3	4	547
4	3	546

NEXT HIGHEST TEMPERATURE

QUAD	ID	°F
1	4	546
2	2	546
3	1	546
4	4	546

FIVE HIGHEST CET TEMPERATURES

	QUAD	ID	°F
1	2	5	547
2	3	4	547
3	1	3	546
4	1	4	546
5	1	5	546

CORE EXIT THERMOCOUPLES

QUADRANT 1

CET	LOC		°F
1	P7	Failed	2300
2	N10	Failed	2300
3	N8	1	546
4	L6	2	546
5	K8	3	546

QUADRANT 2

CET	LOC		°F
1	M3	Failed	2300
2	H5	1	546
3	H3	Failed	2300
4	G2	2	546
5	E4	3	547
6	D3	Failed	2300

QUADRANT 4

CET	LOC		°F
1	L14	Failed	2300
2	L12	Failed	2300
3	J12	1	546
4	J10	2	546
5	H11	3	546
6	G15	4	546
7	F13	5	546
8	F11	6	546

QUADRANT 3

CET	LOC		°F
1	G8	1	546
2	E10	Failed	2300
3	E7	Failed	2300
4	D5	2	547
5	C12	3	546
6	C8	Failed	2300
7	A8	4	546

B Train is OOS per initial conditions.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The accident monitoring instrumentation channels shown in Table 3.3-5 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5.

b. The provisions of Specification 3.0.4 are not applicable to ACTIONS in Table 3.3-5 that require a shutdown.

c. Separate Action entry is allowed for each Instrument.

SURVEILLANCE REQUIREMENTS

4.3.3.3 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-4.

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS

INTENTIONALLY LEFT BLANK

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermo-couples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32 Not Met
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. DELETED				
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
d. Main Steam Lines	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. Gen.	1/stm. Gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.

2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

* Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 30 days, or submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 33 Close the associated block valve and open its circuit breaker.

ACTION 34 With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) Either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 35 DELETED

ACTION 36 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channel OPERABLE, either restore the inoperable channel to OPERABLE status within 30 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 30 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 14 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

QUESTION 94

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.36
	Importance Rating		4.1

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations.

Proposed Question: SRO Question # 94

The following conditions exist:

- Unit 3 is operating in MODE 6 with RCS temperature at 105°F and stable.
- Core reload activities were temporarily suspended with 7 fuel assemblies loaded in the core.
- The Unit Supervisor directed Attachment 5, Restart Minimum Equipment Checklist, of 3-NOP-040.02, Refueling Core Shuffle, to be performed before restarting fuel reload.
- The last items checked on the Restart Minimum Equipment Checklist were:
  - Emergency Air Lock Doors – ACTUAL STATUS: one of two doors is closed
  - NIS Channels – ACTUAL STATUS: N32 is OOS
    N31 is available
    Both Gamma Metric Channels are available
  - R-3-11 and R-3-12 – ACTUAL STATUS: R-3-11 and R-3-12 are available without Normal Containment Coolers running

Which ONE of the following describes the required action(s), if any, prior to the Refueling SRO recommencing fuel reload?

A. Ensure Emergency Air Lock Doors are both closed.

B. Ensure NIS Channel N32 is returned to service.

C. Ensure one Normal Containment Cooler is running.

D. No action is required.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because this is an item checked to restart moving fuel. The novice applicant assumes both Emergency Air Lock Doors will have to be closed. However, the minimum is met.
Tech Spec 3.9.4

B. Incorrect. Plausible because this is an item checked to restart moving fuel. The novice applicant assumes all NIS Channel and Gamma Metrics are required to be available. However, the minimum is met.
Tech Spec 3.9.2

C. CORRECT. R-3-11and R-3-12 require at least one NCC to be running for operation.
Tech Spec 3.9.13

D. Incorrect. Plausible because all items are checked prior to restarting fuel movement. Also, the majority of the conditions are met to move fuel.

Technical Reference(s): 3-NOP-040.02, Rev 7, Attachment 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 7

Comments:

SRO because it evaluates TS requirements applied by SROs as well as administrative controls related to Fuel Handling Activities. See below.

Fuel handling facilities and procedures. [10 CFR 55.43(b)(7)]
Some examples of SRO exam items for this topic include:

- Refuel floor SRO responsibilities.
- Assessment of fuel handling equipment surveillance requirement acceptance criteria.
- Prerequisites for vessel disassembly and reassembly.
- Decay heat assessment.
- Assessment of surveillance requirements for the refueling mode.
- Reporting requirements.
- Emergency classifications.

ATTACHMENT 5
Restart Minimum Equipment Checklist
(Page 1 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
Containment Equipment Door	CAPABLE OF BEING CLOSED		3.9.4	**ENSURE** a designated individual is available to close the Equipment Hatch per0-GMM-051.02, Containment Equipment Hatch and that they are documented in 0-ADM-051, Outage Risk Assessment And Control.
Containment Personnel Air Lock Doors	See Remarks		3.9.4	One of two doors closed or both doors of the Containment Personnel Air Lock may be open if: 1) The plant is in MODE 6 with at least 23 feet of water above the flange, AND 2) At least one Personnel Air Lock Door is capable of being closed in accordance with 3-NOP-051, Containment Personnel Air Lock, AND 3) A designated individual is available outside the Personnel Air Lock to close the door and knows the requirements of 3-NOP-051, Containment Personnel Air Lock.
With both Containment Personnel Air Lock Doors Open	See Remarks		3.9.4	**ENSURE** a designated individual is available outside the Personnel Air Lock to close the door in accordance with 3-NOP-051 and that they are documented in 0-ADM-051, Outage Risk Assessment And Control.
Emergency Air Lock Doors	1/2 CLOSED		3.9.4	

MEJ

ATTACHMENT 5

Restart Minimum Equipment Checklist

(Page 2 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
Refueling Integrity Alignment Satisfied Per 3-OSP-051.12, Refueling Containment Penetration Alignment	ALL PENETRATIONS		3.9.4	
Containment Ventilation Isolation System	OPERABLE		3.9.9	**CHECK** purge valves have been tested per 3-OSP-067.1, Process Radiation Monitoring Operability Test or purge valves are closed on an admin clearance to the Shift Manager.
Plant Vent SPING High Range Noble Gas Monitor	OPERABLE		Table 3.3-5 Item 16.b	
Process Rad. Monitors that Initiate a Containment and Control Room Ventilation Isolation: R-3-11, R-3-12 (Requires ≥ 1 NCC running.)	OPERABLE		3.9.13	IF one or both monitors **NOT** operable, THEN **REFER TO** Technical Specifications for actions.
Control Room Ventilation Isol. Two Channels Actuation Logic and Relays (3QR50 and 3QR51)	OPERABLE		Table 3.3-2 Item 9a Action 16	Both Racks 3QR50 and 3QR51 operable OR comply with TS 3.3.3.1, Item 1a of Table 3.3-4, Action 27, which is to comply with actions of TS 3.9.9 and 3.9.13.
Process Rad. Monitor R-14	OPERABLE		Table 3.3-8 Item 4	

NOT MET ANSWER

ATTACHMENT 5
Restart Minimum Equipment Checklist
(Page 3 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
Area Radiation Monitors Operating: R-2, R-7, R-19 & R-21 Remote/Local Indications and Alarms	REMOTE OPERABLE LOCAL OPERABLE			IF local and remote indicators and alarms for area monitors are **NOT** operable, THEN **USE** a portable monitor with alarm functions.
NIS Channels N31, N32 and Gamma Metrics	2/4 one with audible in Control Room and containment		3.9.2	IF **NOT** Operable, THEN **REFER TO** 3-ONOP-059.5, Source Range Nuclear Instrumentation Malfunction or 3-ONOP-059.6, Backup NIS (Gamma Metrics) Malfunction.
AC Electrical Power Sources	See Remarks		3.8.1.2	One startup transformer and associated circuits, or an alternate circuit, between the off-site transmission network and the 4 KV bus, A or B, and one emergency diesel generator.

NET

ATTACHMENT 5
<u>Restart Minimum Equipment Checklist</u>
(Page 4 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
On-site Power Distribution	See Remarks		3.8.3.2	Minimum of the following: 1. At least one train of emergency buses associated with the unit consisting of one 4 KV bus, three 480 volt load centers, and three (four for Unit 4 train A) vital MCCs. 2. Two 120 volt AC vital buses for the unit energized from their respective DC buses. 3. Three 125 volt DC buses energized from their associated battery banks.
DC Electrical Sources	See Remarks		3.8.2.2	Minimum of three 125 volt battery banks each with their associated full capacity charger capable of being powered by an operable diesel generator.
Control Room Emergency Ventilation System	Operable		3.7.5	Requires 3 air handlers and 2 compressors.
RHR Pumps in Operation and RCS less than 140°F	1/2 MINIMUM		3.9.8.1	RHR loop may be removed from operation for up to 1 hour per 8 hour period, provided **NO** operations are permitted that would cause a reduction of the Reactor Coolant System boron concentration.
Boron Concentration in Vessel per 3-OP-038.1, Preparation For Refueling Activities	greater than or equal to ________		3.9.1	IF vessel head is removed and fuel is in the vessel, THEN **CHECK** daily.

ATTACHMENT 5
Restart Minimum Equipment Checklist
(Page 5 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
Boron Concentration in Refueling Cavity per 3-OP-038.1, Preparation For Refueling Activities	greater than or equal to ________		3.9.1	IF water is in the Refueling Cavity, THEN **CHECK** daily.
Boron Concentration in the SFP per 3-OP-038.1, Preparation For Refueling Activities	greater than or equal to ________		3.9.1, 3.9.14	IF tube gate is open, THEN **CHECK** daily.
Communications Headsets/Other Reliable Communication System – Manipulator Crane to Control Room	CONTINUOUS		3.9.5	**PERFORM** a communications check between all required stations to verify Communication System functioning properly. IF Communications System is **NOT** functional, THEN **PERFORM** Attachment 7, Refueling Communication Verification Checklist. At least one communication system shall be functional.
Spent Fuel Pit Ventilation System	OPERABLE			IF the SFP Ventilation System is inoperable, THEN **STOP** Refueling Operations.
Unit 3 SFP SPING	OPERABLE		Table 3.3-4 Item 2a	
Spent Fuel Pool Demin aligned to SFP	In Service			May be waived by Radiation Protection supervision if clarity and dose rates permit.

ATTACHMENT 5
<u>Restart Minimum Equipment Checklist</u>
(Page 6 of 7)

Equipment/Conditions	Minimum Requirement for Refueling	Checked Initials	Applicable Tech. Spec.	Remarks
Spent Fuel Pit and Refueling Water Cavity water surface and subsurface are clear enough to allow good visibility during refueling operations	CLEAR			**CHECK** Daily – IF **NOT** clear, THEN request RP place Trinuke in service or change filters IF surface clarity is a problem, THEN **REQUEST** RP skim the pool surface.
Refueling cavity water level	56'10" See Remarks Level greater than or equal to 23 feet		3.9.10	During movement of irradiated fuel assemblies or intentional movement of control rods within containment.
Boron Injection Flowpath	ONE		3.1.2.1	VCT or RWST boron concentration must be greater than or equal to the required refueling boron concentration.
A Licensed Senior Reactor Operator or Refueling SRO Supervising Fuel Movement	MAINTAINED		6.2.2.e	
Radiation Protection Coverage for Reactor Cavity Access and SFP	CONTINUOUS			
FME Monitor for Reactor Cavity Access and SFP	CONTINUOUS			FME monitor.

ATTACHMENT 5
Restart Minimum Equipment Checklist
(Page 7 of 7)

Reason for stopping refueling or core shuffle. ______________________________

____

____

____

	Name	Date	Time	Signature
Performed By				
Unit Supervisor confirms conditions are met for restart.				
Shift Manager gives permission for restart.				

sensitive to small increases in activity. This is usually the first indication of a primary to secondary leak.

Per the FSAR, the alarm setpoint is set equivalent to 2.7 x 10^{-4} μCi/ml discharge flow (400 GPM blowdown and 313,000 GPM of circulating water flow).

COMPONENT COOLING WATER MONITORS, CHANNELS R3-17A (R4-17A) AND R3-17B (R4-17B)

The component cooling water return headers each have an in-line monitor to monitor for reactor coolant leakage to the component cooling system. The monitors employ sodium-iodide scintillation detectors which are lead shielded against background radiation to increase the monitor's sensitivity. The alarm setpoint is determined by the radiochemist and set by the I&C Department. Per the FSAR, the setpoint is set equivalent to 5.1 x 10^{-4} μCi/cc given 1,000 gallons of surge volume (X/Q = 1.5 x 10^{-4} sec/m^3). Indication is provided on the PRMS cabinet for the respective Unit. A check source is provided to verify detector response.

The component cooling water system is normally operated as a parallel system. Therefore, the monitors on the return headers should read approximately the same count rate levels.

If R-17A or B alarms, CCW Surge Tank Vent Valve RCV-*-609 closes.

WASTE DISPOSAL SYSTEM LIQUID EFFLUENT MONITOR, CHANNEL R-18

This in-line monitor is located in the plant liquid waste discharge header upstream of the liquid release isolation valve RCV-018. Channel R-18 continuously monitors the plant liquid releases to ensure they stay below 10 CFR 20 liquid release limits. The monitor employs a sodium-iodide scintillation detector which is lead shielded against background to increase its sensitivity. The alarm setpoint is determined by the radiochemist or designee with NPS approval and set by the Unit 3 RCO on shift. Indication is provided on the Unit 3 PRMS cabinet and on the waste/boron panel in the Auxiliary Building. Per the FSAR, the setpoint is set equivalent to 1.3 x 10^{-4} μCi/cc given 20 GPM effluent flow rate into 157,000 GPM of circulating water. A check source is provided to verify detector response. In the alarm condition, Channel R-18 automatically closes the liquid waste discharge isolation valve at the North wall of the Monitor

Tank Room, RCV-018. An alarm on the waste/boron south panel, WASTE LIQUID HI RADIATION on window 5/3, is also annunciated by this channel high alarm. RCV-018 can not be operated from the waste/boron panel until the channel high alarm has been reset.

STEAM GENERATOR LIQUID SAMPLE MONITOR, CHANNEL R3-19 (R4-19)

The R-19 steam generator blowdown liquid sample monitor is provided to continuously monitor a combined and average sample of the three steam generator blowdown streams. A high activity is an indication of a primary to secondary leak. Figure 16 illustrates the blowdown radiation monitor sample piping. The sample line for each steam generator is taken from the blowdown line inside of the containment, and penetrates the containment to the pipe and valve room. Two manual isolation valves and a motor-operated isolation valve are located in each line in this room. The motor operated valves, MOV-1425, 1426 and 1427, are operated by OPEN-CLOSE switches on VPB. An auxiliary feedwater pump auto start or containment isolation Phase A signal will automatically close the motor operated valves. These MOVs also have Phase A isolation "White Light" position indication on VPB, and are interlocked with blowdown control valves CV-6275-A, 6275-B, and 6275-C so that unless the sample MOVs are opened the blowdown control valves cannot be opened. This interlock is to prevent blowing down without continuous S/G sampling as required by Technical Specifications 3.3.3.5. The manual flow control valves downstream of the MOVs in the Primary Sample Rooms are used to throttle and balance the sample flows from the three steam generators to the R-19 detector.

The sample line in the sample room taps off to route a sample to the sample sink in the Chemistry Cold Chem Lab. These lines have air operated control valves, CV-2800, 2801, and 2802. They are automatically closed when the R-19 high alarm is actuated to prevent contaminated liquid from being sent to the Cold Chem Lab sample sink.

In the Sample Room the R-19 sample from each steam generator is then cooled by two parallel sample coolers. A local flow indicator downstream of the sample coolers is provided to indicate the individual sample flows for flow balancing. The flow is adjusted to 2.5 to 7.5 GPH. This flow rate is sufficient to prevent plugging of the throttle valves and the tubing through R-19. A second tap is taken from each sample line to route a cooled sample to the primary sample sink. This provides a location for drawing a grab sample as a means of identifying which S/G is the primary to secondary leak source. Finally, the individual samples pass through a check valve

is processed through a series of filters to maintain the control room environment acceptable during adverse radiological conditions.

DETAILED DESCRIPTION

The Control Room HVAC Envelope consists of the Control Room and the Mechanical Equipment Room (located in the southwest corner of the Cable Spreading Room) including the Control Room's offices, rack area, kitchen, and lavatory. Both rooms are considered part of the envelope because both are serviced and pressurized by the control room's air handlers through common ductwork. The boundaries of the envelope are the floors, walls, ceilings, dampers, doors and ductwork of the two rooms.

During normal operation, fresh makeup air is admitted to this system through an intake louver and two dampers in series (D-1A and D-1B) located in the east wall of the Control Building. This system maintains a positive pressure in the Control Room Envelope greater than that in the cable spreading room in order to prevent smoke from a potential fire in the cable spreading room from entering the Control Room. All Control Room penetrations, including doors, are designed for leak tightness standards. Since the Control Room is maintained at slightly more than atmospheric pressure, the infiltration of contaminated air into the Control Room is negligible. Two radiation monitors located in the air intake downstream of dampers D-1A and D-1B continuously monitor for radiation in the incoming air. Refer to Figure 1 and control drawing 5610-T-E-4535. In the unlikely event of a maximum hypothetical accident (MHA), the Control Room ventilation will automatically be placed in a recirculation mode. The normal outside air intake dampers D-1A and D-1B will close and emergency recirculation fan SF-1B will start. Outside air dampers D-2 and D-3 will open and recirculation dampers D-11A and D-11B will open. Kitchen and lavatory exhaust fans V-56 and V-28 will automatically turn off and their associated dampers will close.

The following signals will automatically activate the recirculation mode:

1. Control Room HVAC test switches (two)
2. High radiation from Control Room intake radiation monitors (6642 or 6643)
3. High containment radiation from R11 or R12
4. Automatic or manual initiation of safety injection
5. Manual containment isolation phase 'A' (1/2 pushbuttons)
6. Manual containment isolation phase 'B' (2/2 pushbuttons)

QUESTION 95

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.35
	Importance Rating		4.5

Equipment Control: Ability to determine Technical Specification Mode of Operation.

Proposed Question: SRO Question # 95

Which ONE of the following describes (1) the required Technical Specification MODE to place Overpressure Mitigating Systems (OMS) in service and (2) in accordance with 0-ADM-536, Technical Specification Bases Control Program, the bases for the required HHSI flow path alignment when OMS is in service?

A. (1) MODE 3
(2) High Pressure Safety Injections flow paths to the RCS shall be **isolated** to limit the mass input into the RCS during low temperature conditions

B. (1) MODE 3
(2) High Pressure Safety Injections flow paths are **unisolated** to provide sufficient core cooling following a LOCA

C. (1) MODE 4
(2) High Pressure Safety Injections flow paths to the RCS shall be **isolated** to limit the mass input into the RCS during low temperature conditions

D. (1) MODE 4
(2) High Pressure Safety Injections flow paths are **unisolated** to provide sufficient core cooling following a LOCA

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Plausible because the unit passes through Mode 3 during a cooldown for refueling. The applicant believes that HPSI is isolated at 350°F. Actual requirement is before 275°F.

B. Incorrect. Plausible because the unit passes through Mode 3 during a cooldown for refueling. The applicant believes that HPSI is unisolated since it provides SI flow as required in 3-ONOP-41.7 when a LOCA develops in Mode 3/4.

C. Correct. See TS 3.4.9.3 and bases.

D. Incorrect. Plausible because the MODE is correct. Also, the applicant believes that HPSI is unisolated since it provides SI flow as required in 3-ONOP-41.7 when a LOCA develops in Mode 3/4. However, TS 3.4.9.3 requires the initial lineup to be isolated in this MODE with the RCS intact to limit the mass input into the RCS during low temperature conditions

Technical Reference(s): TS 3.4.9.3 (Attach if not previously provided)

TS Definitions

3-GOP-305

Proposed References to be provided to applicants during examination: None

Learning Objective: 6900121B Obj 8 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (2DR)

Question Difficulty Level: B

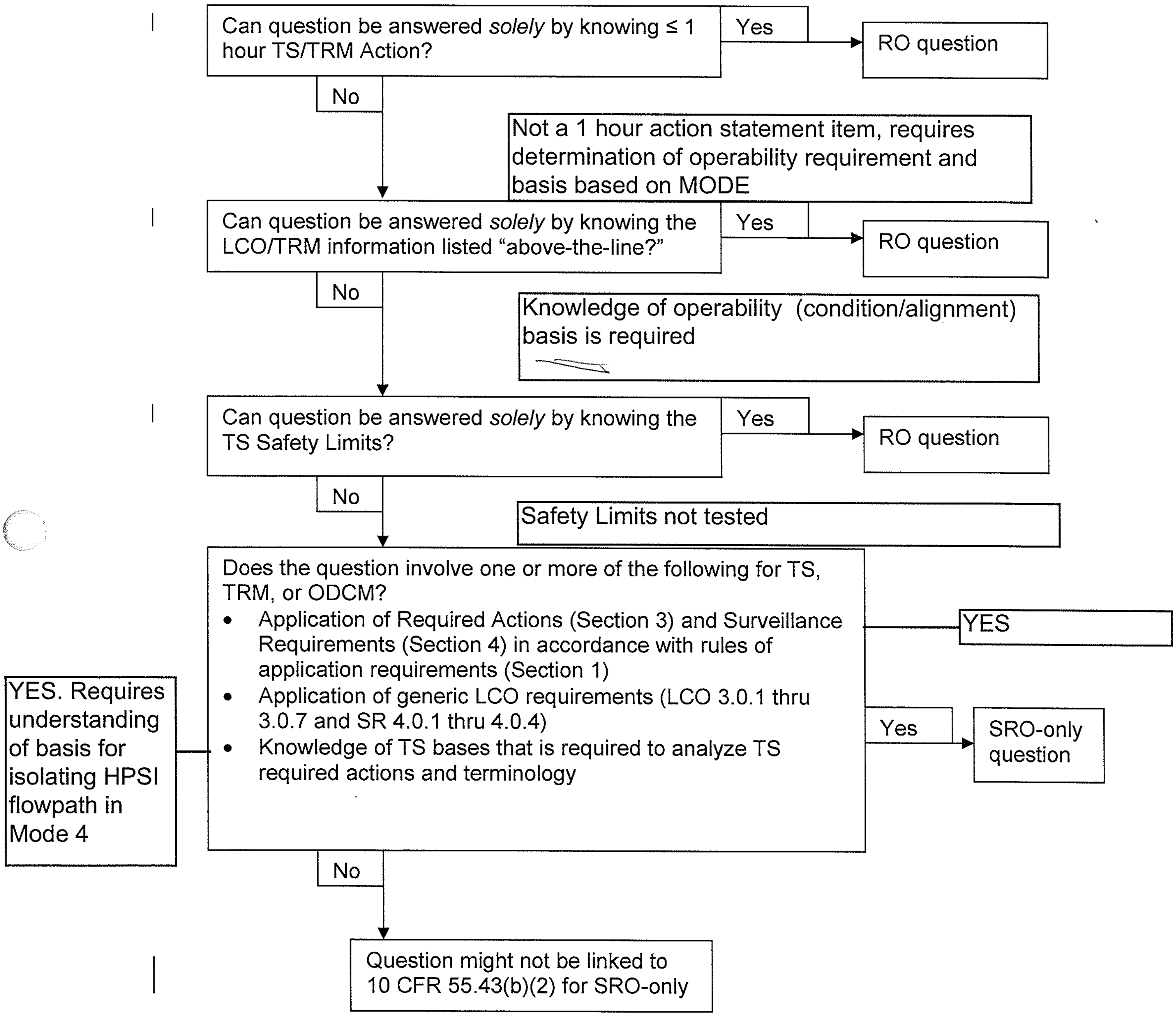
10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

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



REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 468 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

ATTACHMENT 1
(Page 69 of 114)

TECHNICAL SPECIFICATION BASES

3/4.4.9 (Cont'd)

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Overpressure Mitigating System

The Technical Specifications provide requirements to isolate High Pressure Safety Injection from the RCS and to prevent the start of an idle RCP if secondary temperature is more than 50°F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients more severe than those assumed in the low temperature overpressurization protection analysis cannot occur.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.20 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) The start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures including margin for instrument error, or (2) The start of a HPSI pump and its injection into a water-solid RCS. When the PORVs or 2.2 square inch area vent is used to mitigate a plant transient, a Special Report is submitted. However, minor increases in pressure resulting from planned plant actions, which are relieved by designated openings in the system, need not be reported.

Associated requirements for accomplishing specific tests and verifications in SR 4.4.9.3.1.a and 4.4.9.3.1.d allow a 12 hour delay after decreasing RCS cold leg temperature to ≤ 275°F. The bases for the 12 hour relief in completing the analog channel operation test (ACOT) and verifying the OPERABILITY of the backup Nitrogen supply are provided in the proposed license amendment correspondence L-2000-146 and in the NRC Safety Evaluation Report provided in the associated Technical Specification Amendments 208/202 effective October 30, 2000.

Based on the justifications provided therein and the discussion provided in NUREG-1431, Volume 1, Rev.2 (Westinghouse Standard Technical Specifications. Section B3.4.12), the 12 hour delay allowed for completing SR 4.4.9.3.1.a and 4.4.9.3.1.d is considered to start coincident with the enabling of OMS, regardless of RCS cold leg temperature. For example, if OMS is enabled at RCS cold leg temperature of 298°F, the ACOT must be completed within 12 hours of placing OMS in service (not 12 hours after decreasing RCS cold leg temperature to ≤ 275°F). (Reference: PTN-ENG-SENS-03-0046 approved 9/12/03.)

QUESTION 96

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.7
	Importance Rating		3.6

Equipment Control: Knowledge of the process for conducting special or infrequent tests.

Proposed Question: SRO Question # 96

4-OSP-040.12, At Power Measurement of Moderator Temperature Coefficient, is in progress.

- Unit 4 is at 100% Power at the End of Core Life.
- All members of the Control Room Staff and a Reactor Engineer are briefed per 0-ADM-217, Conduct of Infrequently Performed Tests or Evolutions.
- The Reactor Engineer will serve as the Test Director.

In accordance with 0-ADM-217, ___(1)___ is (are) allowed to give direction to licensed operators to operate Control Rods and if this evolution extends into shift turnover, the Shift Manager shall received agreement with the ___(2)___ that turnover can be conducted safely at this time.

A. (1) the Test Director
(2) Management Designee

B. (1) the Test Director
(2) Test Director

C. (1) ONLY On-Shift Senior Reactor Operators
(2) Test Director

D. (1) ONLY On-Shift Senior Reactor Operators
(2) Management Designee

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Plausibility – The second part is correct. Also, it is plausible to have a SRO Test Director give permission to operate unit controls. However, the Reactor Engineer does not have a SRO License.

B. Incorrect. Plausibility – It is plausible to have a SRO Test Director give permission to operate unit controls. However, the Reactor Engineer does not have a SRO License. Also, the Test Director will coordinate certain test aspects with the Unit Supervisor and Shift Manager, but the responsibility to relieve on station for turnover requires joint agreement with the SM and Management Designee.

C. Incorrect. Plausibility – Part one is correct. Also, it is plausible to have a SRO Test Director give permission to operate unit controls. However, the Reactor Engineer does not have a SRO License.

D. Correct. See 0-ADM-217

Technical Reference(s): 0-ADM-217 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 6902045 Obj 2 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

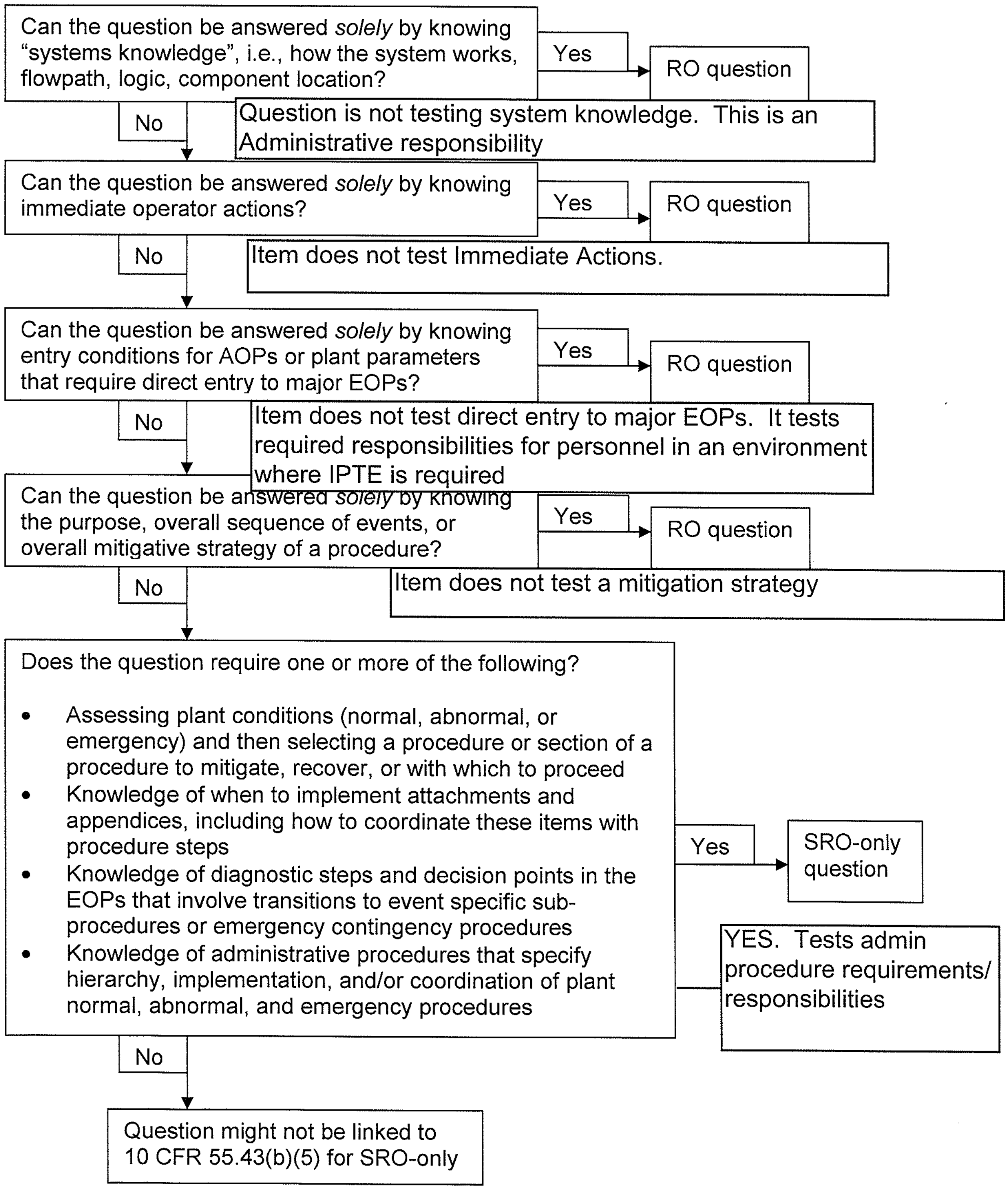
10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

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



5.0 PROCEDURE

CAUTIONS

- ***Direction to licensed operators regarding operation of the unit shall be given ONLY by On-Shift Senior Reactor Operators.***
- ***The highest margin of safety shall be maintained throughout the test or evolution exercising caution and conservatism, particularly when uncertainties or unexpected plant behavior is encountered.***

NOTE

Infrequently performed tests or evolutions often place plant equipment and operators outside the bounds of normal operating procedures and training. Under these conditions, inadequate controls and direction can have more serious consequences and result in significant challenges to reactor safety. Because of this, these tests or evolutions require procedures with enhanced development and review, management oversight with clear direction and expectations, and execution with expected results or actions that should be performed in the event of unexpected plant response.

5.1 Prior to the conduct of an infrequently performed test or evolution, the Site Vice President or the Plant General Manager shall designate a Management Designee.

5.1.1 The level of seniority of the Management Designee is based on the potential risk or complexity of the infrequently performed test or evolution.

5.1.2 The Site Vice President or Plant General Manager may designate more than one individual as the Management Designee for continuous oversight of tests or evolutions that require multiple shifts to complete.

5.1.3 The Management Designee may not re-designate this function to any other individual without prior approval of the Site Vice President or the Plant General Manager.

5.1.4 The Management Designee shall maintain the appropriate level of oversight and independence to ensure a high margin of safety is maintained during performance of the test or evolution.

5.1.5 The Management Designee has the primary responsibility for communicating unexpected or unanticipated conditions encountered during the implementation of this procedure to the Site Vice President or Plant General Manager.

5.1.6 The Management Designee should participate in, observe or spot check pre–evolution activities such as tabletops and training to ensure management expectations for performance of the test or evolution are clearly communicated to all participants.

5.1.7 The Management Designee should ensure the pace of both pre-evolution activities and the test or evolution are appropriate.

5.3.9 Determine any effects if any on the opposite unit to include effects on common equipment required by the Opposite Units Mode of Operation. (i.e., EDGs, HHSI pumps, AFW, CRVS, Battery Chargers, etc.)

5.3.10 Perform in-plant walkdowns as needed.

5.3.11 Conduct simulator training or validation as needed.

5.3.12 Complete pre-evolution section of a form similar to Attachment 1.

5.4 After the pre-evolution section of Attachment 1 is complete, the Test Specialist shall ensure that the prerequisites for the test or evolution are satisfied.

5.5 The Test Specialist shall conduct and document a pre-evolution briefing using a form similar to Attachment 1. Copies of Attachment 1, including all reactor/turbine trip and/or termination criteria, shall be distributed to each individual participating in the evolution.

5.5.1 Record briefing attendance on Attachment 2 and attach to Form 326 used for the briefing.

5.6 Prior to the conduct of the test or evolution, the Test Specialist or management designee shall obtain permission from the Shift Manager.

5.7 If the test or evolution has the potential to cause a plant transient, then the following Control Room positions shall be assumed:

5.7.1 Shift Manager – Oversight position behind the RO desk

5.7.2 Unit Supervisor – Crew direction position at RO desk

5.7.3 Reactor Operator - Reactor Controls Console

5.7.4 BOP - Balance of Plant console

5.8 Communications shall be conducted as follows:

5.8.1 Unit Supervisor - Communicating and directing the actions of the board operators

5.8.2 Shift Manager - Communicating with the Unit Supervisor, maintaining command and control, and limiting excess noise and unnecessary communications

5.9 If a test extends into a shift turnover, the Shift Manager shall ensure the following before turnover is conducted:

5.9.1 The plant is in a stable configuration such that reactor safety will not be jeopardized during the turnover process.

5.9.2 The Shift Manager and the management designee agree that shift turnover can be safely conducted at this time.

QUESTION 97

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		3
K/A #	G3	2.3.5
Importance Rating		2.9

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO Question # 97

A Unit 3 Containment Purge will be conducted in MODE 1 for normal operations.

Which ONE of the following identifies (1) the MINIMUM requirement for radiation monitor operability in accordance with 3-NOP-053, Containment Purge System, and (2) the HIGHEST permission level required to initiate a Containment Purge?

A. (1) R-3-11 (Particulate) and R-3-12 (Gaseous) Containment Radiation Monitors
R-3-14 (Plant Vent) and RAD-6304 (Plant Vent SPING) Radiation Monitors
(2) Plant General Manager

B. (1) R-3-11 (Particulate) or R-3-12 (Gaseous) Containment Radiation Monitors
R-3-14 (Plant Vent) or RAD-6304 (Plant Vent SPING) Radiation Monitors
(2) Plant General Manager

C. (1) R-3-11 (Particulate) or R-3-12 (Gaseous) Containment Radiation Monitors
R-3-14 (Plant Vent) and RAD-6304 (Plant Vent SPING) Radiation Monitors
(2) Shift Manager

D. (1) R-3-11 (Particulate) and R-3-12 (Gaseous) Containment Radiation Monitors
R-3-14 (Plant Vent) or RAD-6304 (Plant Vent SPING) Radiation Monitors
(2) Shift Manager

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausibility – Only one of each radiation monitor is required, not both. Each of the radiation monitors listed provides required monitoring if it was in service as the operable monitor. Also, if these monitors were OOS. There would be compensatory actions. Also, the PGM is required for approval for MODE 1,2, & 3 releases.

B. Correct. Only one of each radiation monitor is correct. The PGM is required for approval for MODE 1, 2, & 3 releases.

C. Incorrect. . Plausibility – One of each monitor group must be operable and capable of performing its function if a purge is initiated. Also, the SM is required for approval for MODE 4-6 releases.

D. Incorrect. Plausibility – Only one of each radiation monitor is required, not both. Each of the radiation monitors listed provides required monitoring if it was in service as the operable monitor. Also, if these monitors were OOS. There would be compensatory actions. The Shift Manager is required for approval in this MODE of operation.

Technical Reference(s): 3-NOP-053 R2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 6902129 Obj 10c (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1P)

Comprehension or Analysis

Question Difficulty Level: B

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.
[10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

- Process for gaseous/liquid release approvals, i.e., release permits.
- Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

3.0 PREREQUISITES

INITIAL

> **NOTE**
>
> Technical Specification 3.3.3.1, Table 3.3-4 specifies the radiation monitoring instrumentation requirements for plant operation.

- **CHECK** the following Process Radiation Monitors are OPERABLE as required by Technical Specifications for purge operation:
  - R-3-11 PARTICULATE or R-3-12, GASEOUS Containment Radiation Monitor ______
  - R-3-14, PLANT VENT or RAD-6304, PLANT VENT SPING radiation monitor ______
- IF R-3-11 PARTICULATE or R-3-12 GASEOUS Radiation Monitor is in operation, THEN **ENSURE** Containment Normal Ventilation and Cooling System is in service with at least one Normal Containment Cooler in operation. ______
- **ENSURE** the Containment Purge System valve and breaker alignment has been verified by completion of the following attachments:
  - Attachment 1, Containment Purge System Valve Alignment ______
  - Attachment 2, Containment Purge System Switch Alignment ______
  - Attachment 3, Containment Purge System Electrical Alignment ______
- **ENSURE** the following electrical systems are energized and capable of supporting operation of the Containment Purge System operation:
  - 3B MCC ______
  - 4B MCC (If using 4V20, U-4 CNTMT PURGE EXHAUST FAN) ______

4.0 NORMAL OPERATIONS

4.1 Startup

4.1.1 Containment Purge Initiation

1. **ENSURE** a completed Radioactive Gas Release Permit has been obtained from Nuclear Chemistry.

2. **ENSURE** Containment Normal Ventilation and Cooling System is in service in accordance with the Prerequisite Section.

NOTE

Technical Specification 3.3.3.1, Table 3.3-4 specifies the radiation monitoring instrumentation requirements for plant operation.

3. IF in MODE 1, 2 OR 3, THEN **ENSURE** permission has been obtained from the Plant General Manager for performance of this section.

4. IF any of the following conditions are met:

   - Plant in MODE 4, 5 or 6
   - Plant DEFUELED
   - Emergency conditions

   THEN **ENSURE** permission has been granted by the Shift Manager for performance of this section.

5. **COMPLETE** Part A, Prestart Data, on Attachment 4, Containment Purge Data.

QUESTION 98

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		3
K/A #	G3	2.3.4
Importance Rating		3.7

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: SRO Question # 98

- The Shift Manager is the Emergency Coordinator.
- Medical Response Workers are briefed to rescue a worker who has suffered a broken leg, non-life threatening condition, during a radiological emergency.
- The worker is located where general area dose rates are 125 mrem/hr.
- The Medical Response Workers will pass through high dose rate areas to reach the worker.
- Health Physics estimates each worker will receive a total dose (TEDE) of 12 rem and a thyroid dose of 3.5 rem while performing this rescue.

Which ONE of the following correctly describes (1) whose approval is required to exceed dose in excess of the Annual Federal Limits and (2) the HP estimated dose (from above) as compared to the TEDE limit of 0-EPIP-20111, Re-entry?

A. (1) the Emergency Coordinator (EC) approval
(2) is within the TEDE limit

B. (1) the OSC Health Physics Supervisor approval
(2) is within the TEDE limit

C. (1) the Emergency Coordinator (EC) approval
(2) exceeds the TEDE limit

D. (1) the OSC Health Physics Supervisor approval
(2) exceeds the TEDE limit

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The EC is the correct position for authorization. This is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

B. Incorrect. The OSC Health Physics Supervisor is the incorrect position for authorization. However, they will give radiological briefings, will issue emergency dosimetry with maximum exposure capabilities, will ensure the Nuclear Division Medical Review Officer notification, and will sign emergency exposure authorization form. Also, this is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

C. Correct because the estimated total dose (12 rem) exceeds its limit (10 rem)

D. Incorrect. The OSC Health Physics Supervisor is the incorrect position for authorization. However, they will give radiological briefings, will issue emergency dosimetry with maximum exposure capabilities, will ensure the Nuclear Division Medical Review Officer notification, and will sign emergency exposure authorization form. Also, this is plausible because the estimated total dose (12 rem) exceeds its limit (10 rem).

Technical Reference(s): 0-EPIP-20111 Steps including Enclosure 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 3200001 Obj. 10 (As available)

Question Source: Bank #

Modified Bank # WTSI 66661 (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty: Moderate (C)

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Modified from Turkey Point 2009 NRC Exam. Changed dose rates to change correct answer.

SRO only because a decision must be made to minimize radiation exposure with an injured person in a high radiation area. This decision would be made by the SRO acting as the E-Plan Emergency Coordinator

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

[10 CFR 55.43(b)(4)]

Some examples of SRO exam items for this topic include:

- Process for gaseous/liquid release approvals, i.e., release permits.
- Analysis and interpretation of radiation and activity readings as they pertain to selection of administrative, normal, abnormal, and emergency procedures.

Analysis and interpretation of coolant activity, including comparison to emergency plan criteria and/or regulatory limits.

SRO-only knowledge should not be claimed for questions that can be answered *solely* based on RO knowledge of radiological safety principles; e.g., RWP requirements, stay-time, DAC-hours, etc.

5.0 PROCEDURE

5.1 General

5.1.1 The following guidelines for emergency exposure of personnel shall be followed during the re-entry operation:

1. Re-entry personnel that have been authorized to exceed regulatory exposure limits should be volunteers, familiar with the risks involved (radiosensitivity of fetuses, effects of acute exposures, etc.), and whose normal duties have trained them for such missions.

2. Declared pregnant adults should not be used as on-site emergency workers.

3. Exposures to emergency workers shall be maintained as low as reasonably achievable (ALARA) and if possible be maintained within site specific radiological exposure guidelines and/or limits identified in 10 CFR 20.

4. Conditions may warrant re-entry into high radiation areas leading to exposure in excess of the regulatory limit. Except for rescue of personnel from a life threatening situation, authorization must be given in advance by the Emergency Coordinator (EC) in consultation with the TSC RP Supervisor (or alternate). If the EOF is operational and as time permits, the EC should obtain concurrence from the Recovery Manager (RM). In any case where regulatory limits have been exceeded, the EC shall notify the RM of the event.

5. If obtaining EC approval for exposure in excess of the regulatory limit will result in leaving the accident scene or decrease the victim(s) chance of survival, life-saving actions may be performed without obtaining EC approval. The EC shall be notified immediately following the rescue operation.

6. Emergency exposures requiring immediate action are not planned and are not controlled as a Planned Special Exposure. Dose received from exposure under emergency conditions will be added to the dose received during the current year, prior to the emergency, to determine compliance with the occupational dose limits in 10 CFR 20.

ENCLOSURE 1
(Page 1 of 3)

EMERGENCY WORKER EXPOSURE LIMITS AND GUIDANCE FOR POTASSIUM IODIDE USE

NOTE

Consult 0-EPIP-20129, Emergency Response Team, Radiological Monitoring for off-site monitoring exposure guidelines.

For the following missions, the exposure limits are (Note 1):	TOTAL DOSE (Note 2) (TEDE)	THYROID (Note 3) (CDE)
Performance of actions that would not directly mitigate the event, minimize escalation, or minimize effluent releases	5 REM	50 REM
Performance of actions that mitigate the escalation of the event, rescue persons from a <u>non-life</u> threatening situation, minimize exposures or minimize effluent releases.	10 REM	100 REM
Performance of actions that: decrease the severity of the event, or terminate the processes causing the event in an attempt to control effluent releases to avoid extensive exposure of large populations. Also rescue of persons from a <u>life-threatening</u> situation.	25 REM	250 REM
Rescue of persons from a life threatening situation. (Volunteers should be above the age of 45.) (Note 4)	(Note 5)	(Note 5)

NOTES

- *Both Total Dose (TEDE) and Thyroid Dose (CDE) should be used for purposes of controlling exposure.*
- *Protective clothing, including respirators, should be used where appropriate.*

(Note 1) Exposure limits to the lens of the eye are three times the Total Dose (TEDE) values listed.

(Note 2) Total Dose (TEDE) is the <u>total</u> dose from both external and internal (weighted) sources - Total Effective Dose Equivalent.

(Note 3) Thyroid dose (CDE) commitment from internal sources - Committed Dose Equivalent. The same dose limits also apply to other organs (CDE), skin (Shallow Dose Equivalent), and extremities (Extremity Dose Equivalent).

(Note 4) Volunteers with full awareness of risks involved, including numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

ATTACHMENT 2
(Page 1 of 1)

EMERGENCY EXPOSURE AUTHORIZATION FORM

Date:____________________

I have been briefed on the radiological consequences and hazards associated with the authorized emergency exposure, and I have volunteered to perform the task described below:

Name of Individual(s)	Social Security Number	TLD Number	Signature	Time

Brief Description of Task:____________________

Authorization Limit:____________________

Briefing Completed By:____________________ Time:________
(Signature)

OSC Radiation
Protection Supervisor:____________________ Time:________
(Signature)

OR

TSC Radiation
Protection Supervisor:____________________ Time:________
(Signature)

Emergency Exposure Authorized by:

Emergency Coordinator:____________________ Time:________
(Signature)

NOTE: Signatures required by TSC personnel may be authorized by phone or fax.

QUESTION 99

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		4
K/A #	G4	2.4.29
Importance Rating		4.4

Emergency Procedures / Plan: Knowledge of the emergency plan.

Proposed Question: SRO Question # 99

Unit 4 experienced an ATWS with a 4C Steam Generator Tube Rupture. The crew is now responding with 4-EOP-E-3, Steam Generator Tube Rupture.

Subsequently the following conditions exist:

- There was an uncontrolled pressure decrease on the 4C S/G due to a stuck open S/G Steam Dump To Atmosphere Valve. The S/G Steam Dump To Atmosphere Valve is now isolated.
- All three Aux Feedwater Steam Supply Valves to Aux Feedwater Pumps are open.
- The Unit 4 Turbine Operator has just been directed to reposition AFW Steam Supply Cross Connect Valves in order to provide steam from an intact S/G(s) to all AFW Pumps.

Which ONE of the following identifies the status of the release and evacuation plan as directed by the Emergency Coordinator in accordance with 0-EPIP-20101, Duties of Emergency Coordinator?

The Emergency Coordinator will declare a release __(1)__ and __(2)__ implement an Owner Controlled Area Evacuation.

	(1)	(2)
A.	has occurred and is stopped	will not
B.	has occurred and is stopped	will
C.	is in progress	will not
D.	is in progress	will

Proposed Answer: D

Explanation (Optional):

A. Incorrect because release is ongoing, however this is plausible after the S/G is isolated in 4-EOP-E-3. If applicant chooses an ALERT classification, it would be logical to say that OCA evacuation isn't required.

B. Incorrect because release is ongoing, however this is plausible after the S/G is isolated in 4-EOP-E-3. With a SAE is declared, the applicant understands an evacuation is required because of the threat of contamination due to the SGTR in progress with a faulted S/G.

C. Incorrect. It is correct that the release is ongoing. However, if applicant chooses an ALERT classification, it would be logical to say that OCA evacuation isn't required.

D. Correct because SAE is declared and because an Owner Controlled Area Evacuation is performed.

Technical Reference(s): 0-EPIP-20101 (Attach if not previously provided)

4-EOP-E-3

Proposed References to be provided to applicants during examination: None

Learning Objective: 3200003 Obj. 4 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis (3SPK)

Question Difficulty: Moderate (B)

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

SRO because the test item evaluates SRO knowledge without any evaluation of RO knowledge items

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

Question is not testing system knowledge. This is an E-Plan responsibility

No ↓

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

Yes → RO question

Item does not test Immediate Actions.

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

Item does not test direct entry to major EOPs. It tests required responsibilities for SROs during E-Plan implementation

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

Item does not test a mitigation strategy

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

YES. Tests procedure requirements/ responsibilities during E-Plan

No ↓

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

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	**Isolate Steam From Ruptured S/G(s) To AFW Pumps**	
	a. Verify SI - RESET	a. Reset SI.
	b. Verify AMSAC - RESET	b. Reset AMSAC.
	c. Verify steam supply aligned to both trains of AFW pumps from intact S/G(s)	c. Reposition AFW steam supply cross-connect valves to provide steam from intact S/G(s) to all AFW pumps. Maintain steam flow to AFW pumps while repositioning cross-connect valves. * AFSS-4-006 * AFSS-4-007
	d. Close AFW pump steam supply MOV on ruptured S/G(s)	
	e. Dispatch an operator to perform the following 1) Open AFW pump steam supply MOV breaker on ruptured S/G(s) 2) Verify AFW pump steam supply MOV on ruptured S/G(s) - CLOSED	
5	**Isolate Miscellaneous Flowpaths From Ruptured S/G(s)**	
	a. Verify blowdown isolation valve(s) from ruptured S/G(s) - CLOSED	a. Locally close associated manual isolation valve: * SGB-4-007 for S/G A * SGB-4-008 for S/G B * SGB-4-009 for S/G C
	b. Check auxiliary steam - SUPPLIED FROM ANOTHER UNIT	b. Perform ATTACHMENT 5 while continuing with Step 6.

5.6.1.3 (Cont'd)

CAUTION

RM approval is required prior to downgrading from a Site Area Emergency or General Emergency.

b. **IF** Downgrading to an Alert, **THEN** make the following announcement twice:

Attention all personnel, attention all personnel. The Emergency has been downgraded to an Alert.

CAUTION

If a significant release (process monitors off scale, or other indications) and/or security related events are in progress (intruders, bomb threat, etc.) inform emergency responders and site evacuees of the best access and egress routes to take on site to minimize hazards. During off hours, dispatch Security to route incoming Emergency Responders away from the hazardous routes.

NOTE

If Plant Events (radiological or security threat considerations) warrant, alternate facilities and/or routes to these facilities may be necessary. Refer to Subsection 5.1, General.

____ TIME

4. Initiate Activation of On-site Emergency Response Facilities (ERFs) per 0-EPIP-20104, Emergency Response Organization Notifications/Staff Augmentation.

CAUTION

The Emergency Coordinator shall use good judgment prior to releasing contractors from the site and clearing those owner controlled areas outside the Protected Area. Such conditions as security events, release status, release duration, plant conditions, and meteorological conditions should be evaluated prior to moving personnel.

____ TIME

5. Determine the need to dismiss non-essential contract personnel from the site **AND** clear those areas outside the Protected Area.

5.6.1 (Cont'd)

6. **IF** a precautionary clearing of personnel outside of the Protected Area is required, **THEN** perform the following:

____ TIME

a. Inform Security to clear personnel from the following areas and implement applicable sections of Security Force Instruction (SFI) 6307:

(1) North Recreational Area

(2) Beach/Boat Ramp Area

(3) Wellness Center

(4) Switchyard

(5) Barge Canal

(6) Sea Survival School

(7) Trailer Areas and other work areas

(8) Land Utilization

____ TIME

b. Contact the Watch Engineers of Units 1, 2, and 5, **AND** inform them of the precautionary clearing of personnel.

7. **IF** there is a localized emergency (fire, high radiation, toxic gas), **THEN** perform the following:

____ TIME

a. Determine an assembly area for personnel evacuate from the affected area.

____ TIME

b. Announce type and location, instruct personnel to stand clear, and to report to the assembly area.

____ TIME

c. Sound applicable alarm, if not previously done.

____ TIME

d. Announce type and location, instruct personnel to stand clear, and to report to the assembly area.

____ TIME

e. Initiate search and rescue, as required.

QUESTION 100

Facility: Turkey Point

Vendor: WEC

Exam Date: 2011

Exam Type: SRO ONLY

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.40
	Importance Rating		4.5

Emergency Procedures / Plan: Knowledge of the SRO's responsibilities in emergency plan implementation.

Proposed Question: SRO Question # 100

In accordance with 0-EPIP-20101, Duties of Emergency Coordinator, which ONE of the following identifies an action that may be delegated by Emergency Coordinator (EC)?

A. Contacting the Duty Call Supervisor

B. Issuing Protective Action Recommendations

C. The Decision to notify Federal, State, and Local authorities

D. The Classification of the Event

Proposed Answer: A

Explanation (Optional):

A. Correct. See 0-EPIP-20101, Step 5.1.1

B. Incorrect. See 0-EPIP-20101, Step 5.1.1

C. Incorrect. See 0-EPIP-20101, Step 5.1.1

D. Incorrect. See 0-EPIP-20101, Step 5.1.1

Technical Reference(s): 0-EPIP-20101, rev 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: None Found (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge (1F)

Comprehension or Analysis

Question Difficulty Level: B

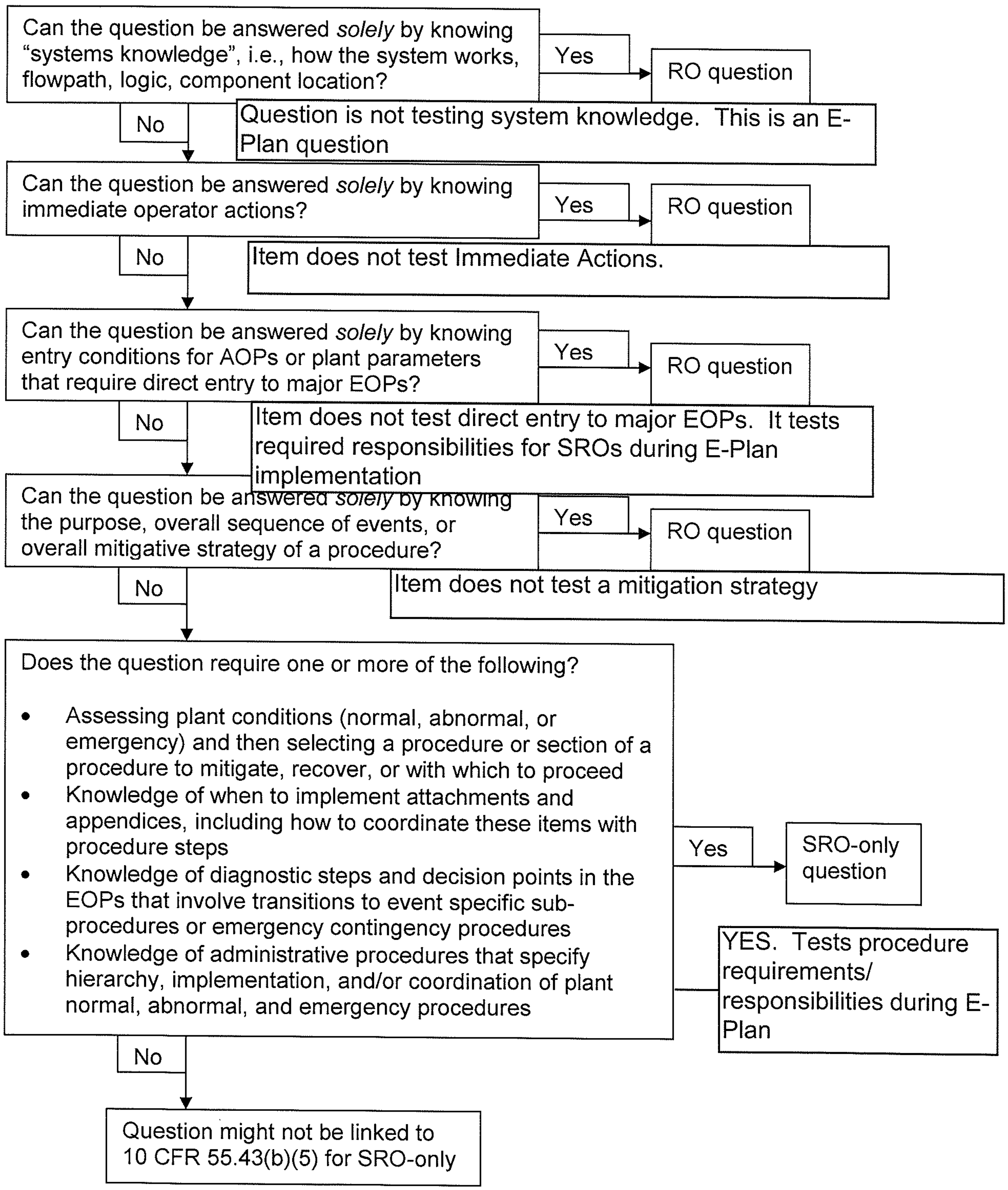
10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

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



5.0 PROCEDURE

5.1 General

5.1.1 The Emergency Coordinator (EC) can delegate his responsibilities to his subordinates with the exception of classification, the decision to notify federal, State, and Local authorities and the issuing of Protective Action Recommendations (PARs). The actual notification can be done by the EC's designee. Notification of off-site agencies and PARs become the responsibility of the Recovery Manager (RM) when the EOF is manned and operational. The EC documents his decision to notify State and Local authorities and his concurrence with PARs by initialing a form similar to F439 (Florida Nuclear Plant Emergency Notification Form).

5.1.2 Classifying the Event

1. The Initiating Conditions (ICs) and Emergency Action Levels (EALs) and their technical bases have been organized as follows:

   Attachment 1, Emergency Classification Hot Conditions Table - Contains all the EALs and bases that apply when the RCS temperature is greater than 200°F. The Fission Product Barrier chart is included to assess the status of the fission product.

   Attachment 2, Emergency Classification Cold Conditions Table - Contains all the EALs and bases, that apply when the RCS temperature is less than or equal to 200°F. Each of these attachments has been converted to a form in the Forms Database that can be used online or printed out as a 11 x 17 color print. The attachments are also maintained in the Unit 3 and Unit 4 Control Rooms and Technical Support Center in a red binder for Hot Conditions EALs and Fission Product Barrier Chart and a blue binder for Cold Conditions EALs.

5.1.3 Procedural notification steps may be performed out of sequence in order to meet State of Florida and/or NRC notification time requirements.

5.1.4 During exercises, drills, or tests, ALL MESSAGES shall begin and end with **THIS IS A DRILL.**

5.1.5 In any case where a **General Emergency** has been declared, the minimum protective action recommendation shall be: **Shelter all people within a 2 mile radius from the plant and 5 miles in the down wind sectors.**

5.1.6 The Emergency Coordinator responsibilities shall reside with the EC in the Control Room until they have been formally transferred to the EC in the TSC.

5.1.7 Emergency notification to State and Local counties is required within15 minutes of declaring an emergency.

5.1.8 Emergency notification to the NRC is required immediately following notification of State and counties, but not later than 1 hour from the declaration of an emergency.