QUESTION 76

On April 27th a series of storms hit the Tennessee Valley, at 0900 Offsite power is lost and the following electrical lineup exists:

- All 500 KV lines are DE-ENERGIZED
- Athens 161KV line is DE-ENERGIZED
- Trinity 161KV line is ENERGIZED
- B and D Emergency Diesel Generators failed to start and CANNOT be manually started
- All other equipment has responded as designed

At 1000, power has been restored to the Browns Ferry transmission yard via two QUALIFIED 500KV lines (Limestone and Union).

Which ONE of the following is the LATEST date & time Units 1 and 2 are required to be in Mode 4 in accordance with Technical Specification 3.8.1, AC Sources- Operating?

[REFERENCE PROVIDED]

A. April 27th at 2200

B. April 28th at 2300

C. April 27th at 2300

D April 28th at 2200

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295003 AA2.05	
	Importance Rating		4.2

295003 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:AA2.05 Whether a partial or complete loss of A.C. power has occurred

Explanation: **B CORRECT** –At 0900 3.8.1 Condition J applies (One or more required offsite circuits and two or more unit 1 and 2 diesel generators inoperable). Required Action J.1 (Enter LCO 3.0.3 Immediately) would require Mode 3 at 2200 on April 27th and Mode 4 on April 28th at 2200. One hour later at 1000 when 2 offsite circuits are restored, Condition J no longer applies and LCO 3.0.3 is exited. TS 3.8.1 condition H (Two or more Unit 1 and 2 diesel generators inoperable) still applies from the original entry time of 0900. After 2 hours (at 1100), TS 3.8.1 condition I (Required completion time not met) will require Mode 3 at 2300 on April 27th and Mode 4 at 2300 on April 28th.

A Incorrect –Plausible because 2200 would be the correct time if 3.0.3 is followed all the way to Mode 4, however the date should be April 28th (the next day).

C Incorrect – Plausible because 2300 is the correct time, however the date should be April 28th (the next day).

D Incorrect – Plausible because 2200 would be the correct time if 3.0.3 is followed all the way to Mode 4, the date is correct.

Technical Reference(s): Unit 2 Tech Spec 3.8.1

Proposed references to be provided to applicants during examination: Unit 2 Tech Spec 3.8.1, without bases.

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.43(b) 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources - Operating

LCO 3.8.1 The following AC electrical power sources shall be OPERABLE:

a. Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System;

b. Unit 1 and 2 diesel generators (DGs) with two divisions of 480 V load shed logic and common accident signal logic OPERABLE; and

c. Unit 3 DG(s) capable of supplying the Unit 3 4.16 kV shutdown board(s) required by LCO 3.8.7, "Distribution Systems - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTE---

LCO 3.0.4.b is not applicable to DGs.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required offsite circuit inoperable.	A.1	Verify power availability from the remaining OPERABLE offsite transmission network.	1 hour <u>AND</u> Once per 8 hours thereafter
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Evaluate availability of both temporary diesel generators (TDGs).	1 hour AND Once per 12 hours thereafter
	AND	
	B.3 Declare required feature(s), supported by the inoperable Unit 1 and 2 DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND	
	B.4.1 Determine OPERABLE Unit 1 and 2 DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	B.4.2 Perform SR 3.8.1.1 for OPERABLE Unit 1 and 2 DG(s).	24 hours
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.5 Restore Unit 1 and 2 DG to OPERABLE status.	7 days from discovery of unavailability of TDG(s) AND 24 hours from discovery of Condition B entry $\geq$ 6 days concurrent with unavailability of TDG(s) AND 14 days AND 21 days from discovery of failure to meet LCO (continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One division of 480 V load shed logic inoperable.	C.1	Restore required division of 480 V load shed logic to OPERABLE status.	7 days
D. One division of common accident signal logic inoperable.	D.1	Restore required division of common accident signal logic to OPERABLE status.	7 days
E. Two required offsite circuits inoperable.	E.1	Declare required feature(s) inoperable when the redundant required feature(s) are inoperable.	12 hours from discovery of Condition E concurrent with inoperability of redundant required feature(s)
	<u>AND</u>		
	E.2	Restore one required offsite circuit to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
----------------NOTE------------ Only applicable when more than one 4.16 kV shutdown board is affected. ------------------------------------- F. One required offsite circuit inoperable. <u>AND</u> One Unit 1 and 2 DG inoperable.	-----------------NOTE--------------- Enter applicable Conditions and Required Actions of LCO 3.8.7, "Distribution Systems - Operating," when Condition F is entered with no AC power source to any 4.16 kV shutdown board. -- F.1 Restore required offsite circuit to OPERABLE status. <u>OR</u> F.2 Restore Unit 1 and 2 DG to OPERABLE status.	 12 hours 12 hours
----------------NOTE------------ Applicable when only one 4.16 kV shutdown board is affected. ------------------------------------- G. One required offsite circuit inoperable. <u>AND</u> One Unit 1 and 2 DG inoperable.	G.1 Declare the affected 4.16 kV shutdown board inoperable.	Immediately

(continued)

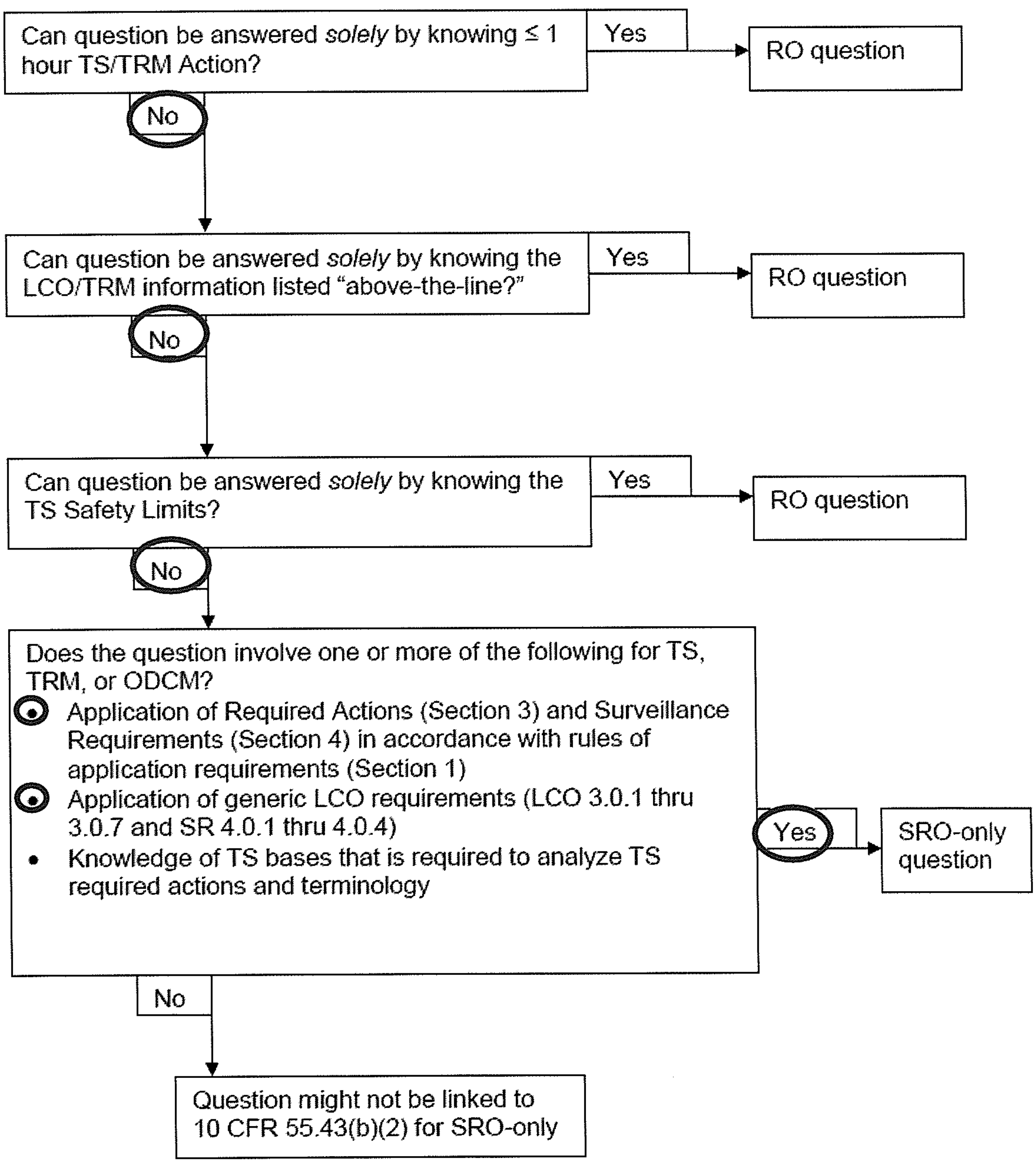
ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Two or more Unit 1 and 2 DGs inoperable.	H.1 Restore all but one Unit 1 and 2 DG to OPERABLE status.	2 hours
I. Required Action and Associated Completion Time of Condition A, B, C, D, E, F, or H not met.	I.1 Be in MODE 3. AND I.2 Be in MODE 4.	12 hours 36 hours
J. One or more required offsite circuits and two or more Unit 1 and 2 DGs inoperable. OR Two required offsite circuits and one or more Unit 1 and 2 DGs inoperable. OR Two divisions of 480 V load shed logic inoperable. OR Two divisions of common accident signal logic inoperable.	J.1 Enter LCO 3.0.3.	Immediately

(continued)

SRO Only Justification: For the sequence of events presented in the stem of this question, the SRO will be required to assess plant conditions and select the correct Technical Specification Condition that applies to determine when the Required Action Completion Time for MODE 4.

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



QUESTION 77

Unit 3 is operating at 100% power when a control air leak occurs on Unit 3.
The Unit 2 to Unit 3 Control Air Crosstie, 2-PCV-032-3901 closes as designed.

Control air pressure continues to drop on Unit 3 and the operators manually scram the reactor.

Not all control rods fully inserted and the APRM DOWNSCALE lights are illuminated.

Which ONE of the following describes the operators' plant control actions?

Attempt to insert control rods IAW 3-EOI APPENDIX-__(1)__ and maintain RPV level __(2)__.

A. (1) 1F, MANUAL SCRAM
(2) (-)50 to (-)100 inches

B. (1) 1F, MANUAL SCRAM
(2) (+)2 to (+)51 inches

C. (1) 1E, MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA
(2) (-)50 to (-)100 inches

D (1) 1E, MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA
(2) (+)2 to (+)51 inches

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295019 AA2.02	
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: AA2.02 Status of safety related instrument air loads.

Explanation: **D** CORRECT – Inserting Control Rods using appendix 1E, MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA, is an available option for inserting control rods. 3-EOI APPENDIX-1F, Manual Scram, resets the scram and drains the scram discharge volume to allow a subsequent scram. Since the CRD Scram discharge volume vent and drain valves (3-FCV-85-83(83A)(82)(82A)), fail CLOSED on loss of air, draining the SDV is not possible. With APRM DOWNSCALE lights are illuminated, the correct level band is +2 to +51 inches per 3-EOI-1.

A Incorrect – First Part: Incorrect. 3-EOI APPENDIX-1F, Manual Scram, resets the scram and drains the scram discharge volume to allow a subsequent scram. Since the CRD Scram discharge volume vent and drain valves (3-FCV-85-83(83A)(82)(82A)), fail CLOSED on loss of air, draining the SDV is not possible. Second part: Incorrect. -50 to -100 inches is plausible because this is the level band for rods out and power >5% (APRM's NOT downscale).

B Incorrect – First Part: Incorrect. 3-EOI APPENDIX-1F, Manual Scram, resets the scram and drains the scram discharge volume to allow a subsequent scram. Since the CRD Scram discharge volume vent and drain valves (3-FCV-85-83(83A)(82)(82A)), fail CLOSED on loss of air, draining the SDV is not possible. Second Part: Correct.

C Incorrect – First Part: Correct. Second part: Incorrect. -50 to -100 inches is plausible because this is the level band for rods out and power >5% (APRM's NOT downscale).

Technical Reference(s): 3-AOI-32-2, 3-EOI APPENDIX-1F, 3-EOI APPENDIX-1E, 3-EOI-1, 3-EOI C-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New :

Question History: Previous NRC: Hope Creek 2009 NRC #78

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

SRO Only Justification: With the conditions presented in the stem of this question, the SRO will be required to assess the impact of the air system on rod insertion and select 1E, MANUAL INSERTION OF CONTROL RODS BY VENTING THE OVER PISTON AREA as an available method for inserting control rods. Additionally, based on the shutdown status of the reactor (<5% as indicated by APRM downscales), the SRO would need be familiar with the 3-EOI-1, RPV Control flowchart to direct the correct RPV Level band.

Attachment 1
(Page 6 of 6)

Expected System Responses

17.0 CORE SPRAY

A. 3-FCV-75-57 and 3-FCV-75-58, PSC PUMP SUCTION INBD & OUTBD ISOL VALVES, fail CLOSED on loss of air. ECCS discharge piping pressure must be maintained greater than TRM 3.5.4 limits by the condensate storage and supply system.

18.0 FUEL POOL COOLING

A. 3-FCV-78-7, DRAIN VLV TO MAIN CONDENSER, fails CLOSED.

B. FUEL POOL F/D C INFLUENT & EFFLUENT VLVs, 3-FCV-78-19 and 3-FCV-78-26, fail CLOSED.

C. FUEL POOL F/D C HOLD PUMP DISCH VLV, 3-FCV-78-33, fails OPEN.

19.0 CRD

A. SCRAM INLET and OUTLET VALVES, 3-FCV-85-39A(B), fail OPEN.

B. EAST & WEST CRD SCRAM DISCH VOL VENT CONT VLVs A & B, 3-FCV-85-83(83A)(82)(82A), fail CLOSED on loss of air.

C. EAST & WEST CRD SCRAM DISCH VOL DRAIN CONT VLVs A & B, 3-FCV-85-37C(D)(E)(F), fail CLOSED on loss of air.

D. CRD SYSTEM FLOW CONTROL VALVES A & B 3-FCV-85-11A and 3-FCV-85-11B, fail CLOSED on loss of air. Valves can be manually opened if required.

RESET ARI.
DEFEAT ARI LOGIC TRIPS IF NECESSARY (APPX 2)

RC/Q-20

INSERT CONTROL RODS USING ONE OR MORE OF THE FOLLOWING METHODS:

PLANT CONDITIONS	AVAILABLE METHODS	APPX
SCRAM VALVES FAILED TO OPEN	**DEENERGIZE** SCRAM SOLENOIDS	1A
	VENT THE SCRAM AIR HEADER	1B
SCRAM VALVES OPENED BUT SDV IS FULL	1. **RESET** SCRAM **DEFEAT** RPS LOGIC TRIPS IF NECESSARY 2. **DRAIN** SDV 3. **RECHARGE** ACCUMULATORS 4. **INITIATE** RX SCRAM	1F
MANUAL CONTROL ROD INSERTION METHODS	**DRIVE** CONTROL RODS. **BYPASS** RWM IF NECESSARY	1D
	RAISE CRD COOLING WATER HEADER DP	1G
	SCRAM INDIVIDUAL CONTROL RODS	1C
	VENT CONTROL ROD OVER PISTON VOLUMES	1E

RC/Q-21

IS RX POWER ABOVE 5% OR UNKNOWN

C5-9

NO

YES

STOP AND **PREVENT** ALL INJ INTO RPV EXCEPT FROM RCIC, CRD, AND SLC (APPX 4)

C5-10

WHEN RPV WATER LEVEL DROPS BELOW -50 IN.

THEN **CONTINUE**

C5-11

D

C5-29

WHILE EXECUTING STEPS C5-15 TO C5-18:	
IF	**THEN**
RX POWER IS ABOVE 5% OR UNKNOWN AND RPV WATER LVL IS ABOVE -50 IN.	**RETURN** TO STEP C5-3 A

C5-14

#5 RAPID RPV INJ MAY CAUSE CORE DAMAGE

#2 PUMP NPSH AND VORTEX LIMITS

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140 °F

MAINTAIN RPV WATER LVL BETWEEN -180 IN. AND +51 IN. WITH THE FOLLOWING INJ SOURCES:

INJ SOURCE	APPX	INJ PRESS
CNDS AND FW	5A	1210 PSIG
CRD	5B	1640 PSIG
RCIC, WITH CST SUCTION IF POSSIBLE	5C	1240 PSIG
HPCI, WITH CST SUCTION IF POSSIBLE	5D	1240 PSIG
CNDS	6A	410 PSIG
LPCI	6B, 6C	320 PSIG
SYSTEMS LISTED IN STEP C5-25 ONLY IF STEP C5-25 HAS BEEN PREVIOUSLY PERFORMED	—	—

C5-15

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

Does the question require one or more of the following?

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

Yes → SRO-only question

No

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

Hope Creek 2009 #78

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295019 AA2.02	
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads

Proposed Question: SRO 78

Hope Creek is operating at 100% power when an Instrument Air line in the Turbine Building ruptures. The air compressors are unable to keep up with the loss of air and Instrument Air pressure is lowering.

The operators insert a manual scram.

What will the Reactor Pressure Vessel (RPV) level control and pressure control strategy be for the loss of Instrument Air?

A. IAW EOP-101 "RPV Control", SRVs for pressure control, HPCI/RCIC for level control.

B. IAW EOP-101 "RPV Control", SRVs for pressure control, Maximize CRD for level control.

C. IAW AB.ZZ-0000 "Reactor SCRAM", Bypass Valves for pressure control, HPCI/RCIC for level control.

D. IAW AB.ZZ-0000 "Reactor SCRAM", Bypass valves for pressure control, Maximize CRD for level control.

Proposed Answer: A

QUESTION 78

Unit 3 is in a refueling outage with the final fuel loading in progress.

Division II work is in progress with the following equipment placed under clearance:

- 3-FCV-74-66, RHR SYS II LPCI OUTBD INJECT VALVE
- RHR Pump 3B

C2 RHRSW Pump is tagged out due to a ground on the motor.

The following annunciators are received:

- FUEL POOL SYSTEM ABNORMAL (9-4C, Window 1)
- FLOOD-UP LEVEL ABNORMAL LT-3-55 (9-3E, Window 29)

A report from the refuel floor indicates that the fuel pool level is noticeably LOWERING.

Which ONE of the following describes the MINIMUM operator actions necessary to comply with Technical Specifications?

[REFERENCE PROVIDED]

A. Verify TWO alternate methods of decay heat removal are available AND verify reactor coolant circulation by an alternate method within ONE (1) hour.

B. Immediately suspend movement of fuel assemblies in the RPV, AND start the 3A RHR Pump in shutdown cooling within TWO (2) hours of securing shutdown cooling.

C. Verify ONE alternate method of decay heat removal is available AND EITHER start the 3A RHR Pump in shutdown cooling OR verify reactor coolant circulation by an alternate method within TWO (2) hours.

D. Immediately suspend movement of fuel assemblies in the RPV, verify ONE alternate method of decay heat removal is available within ONE (1) hour, AND verify reactor coolant circulation by an alternate method within ONE (1) hour.

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295023 G2.2.40	
	Importance Rating		4.7

295023 Refueling Accidents G2.2.40 Ability to apply Technical Specifications for a system

Explanation: **B** CORRECT – The fuel pool system abnormal and floodup abnormal together indicate a low level (per SR-2 the fuel pool abnormal alarm is used to meet the TS requirement to have ≥22 ft above the flange. This requires immediately suspending fuel movement in the vessel per TS 3.9.6. TS 3.9.8 (RHR Low Water Level) requires two shutdown cooling subsystems (A&C are currently available) with one in operation but the operating subsystem may be secured for 2 out of 8 hours.

A Incorrect –Plausible since TS 3.9.8 requires this if the 2 required shutdown cooling subsystems are available and does not pay attention to the note that allows circulation to be secured for 2 out of 8 hours.

C Incorrect – Plausible if the operator believes that the non-availability of C2 RHRSW Pump makes the C subsystem inoperable then an alternate means of decay heat removal would be required.

D Incorrect See A & C

Technical Reference(s): TS 3.9.6, 3.9.7, 3.9.9 ARP 9-4C and 9-3E and 3-SR-2

Proposed references to be provided to applicants during examination: TS 3.9.7 page 3.9-14 & TS 3.9.8 page 3.9-18

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X

10 CFR Part 55 Content: 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

SRO Only Justification: With the conditions presented in the stem of this question, the SRO will be required to assess the impact of the FUEL POOL SYSTEM ABNORMAL (9-4C, Window 1) and FLOOD-UP LEVEL ABNORMAL LT-3-55 (9-3E, Window 29) annunciators on Technical Specifications 3.9.6, RVP Water Level, 3.9.7, RHR- High Water Level, and 3.9.8, RHR- Low Water Level. With knowledge of the surveillance SR-2, the SRO will determine that fuel movement in the vessel is required to be suspended immediately and that two RHR SDC subsystems (as defined in the Tech Spec Bases) are required to be Operable, and one shall be in operation. Placing an RHR SDC subsystem may be delayed up to 2 hours as provided in the NOTE in TS 3.9.8.

FUEL POOL SYSTEM ABNORMAL

3-XA-78-51

1

(Page 1 of 2)

Sensor/Trip Point:

3-PIS-078-0011&0016	Pumps A and B Low Disch Press	100 psi
3-FIS-078-0005	Gate Seal or Drywell to Reactor Well Seal Leakage	5 gpm
3-LS-078-0002B	Fuel Pool Low Level	El 662'8"
3-FS-078-0051	Refueling Bellows Leakage	5 gpm

Sensor Location:	3-PIS-078-0011,-0016	Panel 25-16, El 621', Col R-18 S-LINE
	3-FIS-078-0005	Panel 25-15, El 621', Col R-19 S-LINE
	3-LS-078-0002B	Panel 25-15, El 621', Col R-19 S-LINE
	3-FS-078-0051	Panel 25-15, El 621', Col R-19 S-LINE

Probable Cause:

A. Pump A and B low discharge pressure (it takes low discharge pressure on both pumps to cause this annunciation to alarm.
B. Gate seal or drywell to reactor well seal leakage.
C. Fuel pool level low.
D. Refueling bellows leakage.
E. Using RWCU Blowdown During Refueling, 3-OI-69.
F. Lighting Cabinet 306/Breaker 7 Trip (Supply to Panel 25-15 Fuses)
G. 3A Lighting Board Compartment 2D1 Trip (Supply to Lighting Cabinet 306).

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to Panels 25-15 and 25-16 to determine the cause of the alarm. ☐
B. **IF** Dry Cask activities are being performed in the SFSP, **THEN** **NOTIFY** the Cask Supervisor. ☐
C. **IF** no cause can be determined from Panels 25-15 or 25-16, **THEN** **VERIFY** the following breakers CLOSED:
   - Lighting Cabinet 306/breaker 7 ☐
   - 3A 240V Lighting Board Compartment 2D1 ☐

Operator Action: (Continued)

D. IF fuel pool level is low, THEN
PERFORM the following:
- CHECK weir settings. ☐
- ADD water from other sources, IF required. ☐

E. IF seal leakage alarm is valid, THEN
PERFORM the following:
1. VERIFY CLOSED 3-DRV-078-0558. ☐
2. VERIFY CLOSED 3-DRV-078-0569. ☐
3. DETERMINE leak source. ☐

F. IF FPC Pump discharge pressure low alarm is valid, THEN
SWAP FPC pumps, REFER TO 3-OI-78. ☐

G. [NER/C] DIRECT personnel to make frequent checks at Panel 25-15 and 25-16 until this alarm is reset. [INPO O&MR 363] ☐

H. IF Fuel Pool Cooling System failure has occurred, THEN
REFER TO 3-AOI-78-1. ☐

I. IF this alarm is invalid, THEN
REFER TO 0-OI-55. ☐

References:

GE 730E931 Series	3-45E620-4	3-47E610-78-1	
3-47E855-1	47E832-1	3-45E3631-11	45N3635-17

FSAR Sections 10.5.4, 10.5.6, 13.6.2

Technical Specifications Section 3.7.6

Technical Requirements Manual Sections 3.9.2, 3.9.3

FLOOD-UP LEVEL ABNORMAL LT-3-55	Sensor/Trip Point:	Normal Operation	0-200 Inches (+/-1.5 in.) Alarm Clear Green Band
			201-500 Inches (+/- 1.5 in.) Alarm In Yellow Band
29	3-LT-003-0055	Refueling Operation	475-484 Inches (+/- 1.5 in.) Alarm In Yellow Band
(Page 1 of 1)			485-489 Inches (+/- 1.5 in.) Alarm Clear Green Band
			490-500 Inches (+/- 1.5 in.) Alarm In Red Band

Sensor Location: Unit 3 Rx Bldg El. 593 Panel 3-25-5B Col S-R17

Probable Cause:

A. Refueling Operation
- Make-Up/Dumpback Mismatch
- Loss of Make-up or Dumpback
- System misalignment

B. Normal Operation
- RHR/CS/HPCI/RCIC Injection

Automatic Action:

A. None
B.

Operator Action:

A. Contact refuel floor to verify Rx vessel level and trend. ☐
B. If Rx vessel level is high then adjust make-up/dumpback or secure injection source. ☐
C. If Rx vessel level is low adjust make/up dumpback, align alternate injection source as required and locate cause of lowering level. ☐

References:

3-47E600-58	3-47E610-3-2	3-729E889
3-45N3664-2	3-LCI-L-03-055	

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV water level shall be ≥ 22 ft above the top of the RPV flange.

APPLICABILITY: During movement of irradiated fuel assemblies within the RPV,
During movement of new fuel assemblies or handling of control rods within the RPV, when irradiated fuel assemblies are seated within the RPV.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RPV water level not within limit.	A.1	Suspend movement of fuel assemblies and handling of control rods within the RPV.	Immediately

3.9 REFUELING OPERATIONS

3.9.7 Residual Heat Removal (RHR) - High Water Level

LCO 3.9.7 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----------------------------------NOTE-----------------------------------
The required RHR shutdown cooling subsystem may not be in operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and the water level ≥ 22 ft above the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1	Verify an alternate method of decay heat removal is available.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR) - Low Water Level

LCO 3.9.8 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

--------------------------------------NOTE--------------------------------------
The required operating shutdown cooling subsystem may not be in operation for up to 2 hours per 8 hour period.

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and the water level < 22 ft above the top of the RPV flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1	Initiate action to restore secondary containment to OPERABLE status.	Immediately
	<u>AND</u>		
	B.2	Initiate action to restore two standby gas treatment subsystems to OPERABLE status.	Immediately
	<u>AND</u>		
	B.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. No RHR shutdown cooling subsystem in operation.	C.1	Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND		
	C.2	Monitor reactor coolant temperature.	Once per hour

Surveillance Procedure Data Package - Modes 4 & 5

TABLE 3.18 REACTOR WATER LEVEL DAY SHIFT WEEK: __________ to __________

APPLICABILITY:	During movement of irradiated fuel assemblies within the RPV and during movement of new fuel assemblies or handling of control rods within the RPV, when irradiated fuel assemblies are seated within the RPV. (Refer To P&L Step 3.6A)			
Surveillance Requirements:	3.9.6.1			
LOCATION:	Reactor Building Elevation 639 local observation		Review Initials	
		LIMITS (AC)	UO	Unit Supvr
Friday		≥ 22 ft above the top of the RPV Flange (Note 1)		
Saturday				
Sunday				
Monday				
Tuesday				
Wednesday				
Thursday				

(1) If the gates between the fuel pool and the Reactor well have been removed, verification that the fuel pool low level annunciator (FUEL POOL SYSTEM ABNORMAL 3-XA-55-4C, Window 1) is reset will verify that the level is ≥ 22 ft. above the Top of the RPV flange.

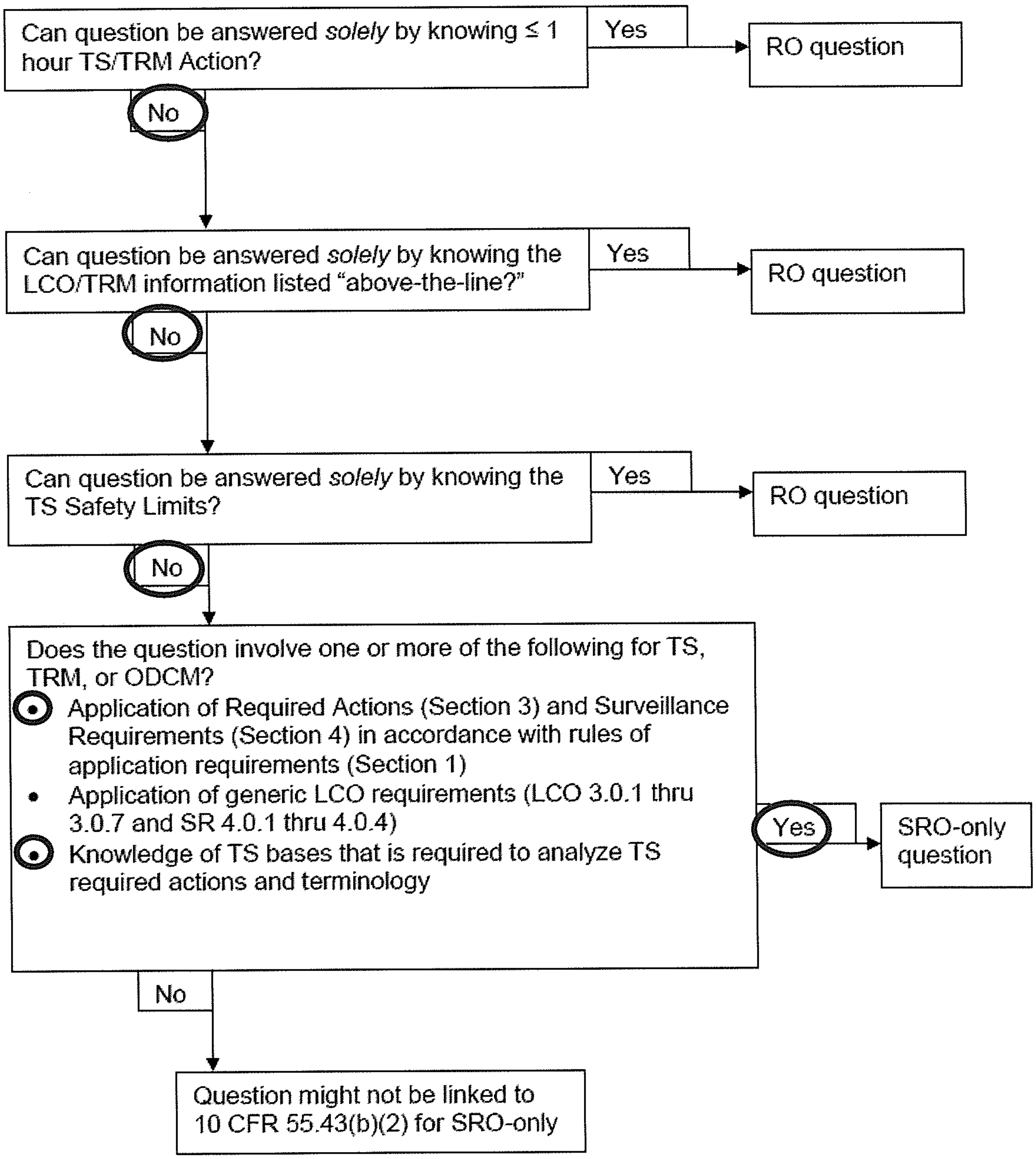
SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

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



QUESTION 79

Unit 1 was in MODE 4 preparing to go to MODE 2. An air leak ~~developed in the Drywell and~~ resulted in Drywell Pressure of 2.5 psig.

The following indications are observed on the Containment Isolation Status System (CISS) on Panel 1-9-4:

- Groups 1, 3, 4, 5 AND 8 PCIS Logic Success Lights are **NOT** illuminated
- Groups 2 AND 6 PCIS Logic Success Lights are illuminated

The leak has since been isolated AND Drywell Pressure has been restored to 1.3 psig and steady.

Based on these current conditions, which ONE of the following completes the statements?

The Unit Supervisor must direct __(1)__ to be closed.

In accordance with Tech Specs, Unit 1 __(2)__ permitted to change Modes to Mode 2.

A. (1) Reactor Water Cleanup (RWCU) supply and return valves (1-FCV-69-1, FCV 69-2 and 1-FCV-69-12)
(2) is NOT

B. (1) Reactor Water Cleanup (RWCU) supply and return valves (1-FCV-69-1, FCV 69-2 and 1-FCV-69-12)
(2) is

C. (1) Traversing Incore Probe Ball **AND** Purge valves
(2) is

D. (1) Traversing Incore Probe Ball **AND** Purge valves
(2) is NOT

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295024 G2.4.21	
	Importance Rating		4.6

295024 High Drywell Pressure G2.4.21 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.

Explanation: **C** CORRECT – High Drywell pressure of 2.45 psig should result in a Group 8 Isolation. CISS indicates the group failed to isolate as required. With the failure of the automatic function, the crew must take the action to isolate the Traversing Incore Probe Ball and Purge valves . Second Part: TS 3.0.4 states that when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. TS 3.6.1.3 Condition A must be entered for the TIP Containment Isolation valves but permits continued operation in the MODE for an unlimited period of time.

A Incorrect – First Part: Incorrect. Group 3 (RWCU) indicates that a successful isolation did not occur. However, this group does not isolate on High Drywell Pressure therefore there is no failed automatic action. Plausible in that several Groups do require isolation on high drywell pressure. Second Part: Incorrect. Plausible in that the plant is going to an applicable mode for PCIVs.

B Incorrect – First Part: Incorrect. Group 3 (RWCU) indicates that a successful isolation did not occur. However, this group does not isolate on High Drywell Pressure therefore there is no failed automatic action. Plausible in that several Groups do require isolation on high drywell pressure. Second Part: Correct.

D Incorrect –First Part: Correct. Second Part: Incorrect. Plausible in that the plant is going to an applicable mode for PCIVs.

Technical Reference(s): U2 TS 3.0.4, TS 3.6.1.3

Proposed references to be provided to applicants during examination: None

Learning Objective (As available): OPL171.017 V.B.4

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Browns Ferry 1006 NRC Exam #81

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

10 CFR Part 55 Content: 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

3.0 LCO APPLICABILITY (continued)

LCO 3.0.4 When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall only be made:

a. When the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time;

b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this Specification are stated in the individual Specifications, or

c. When an allowance is stated in the individual value, parameter, or other Specification.

This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

3.6 CONTAINMENT SYSTEMS

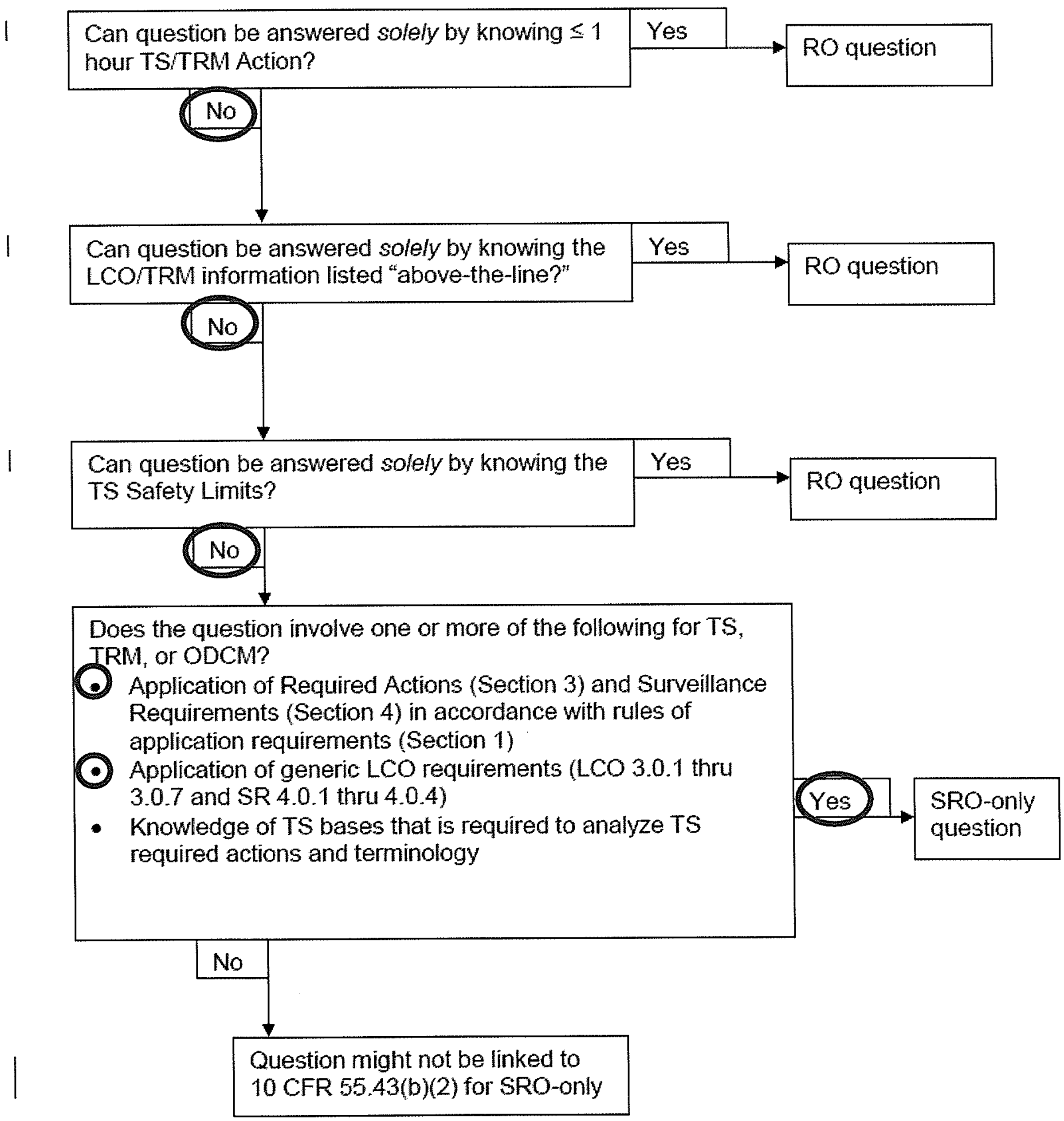
3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

SRO Only Justification: With the conditions presented in the stem of this question, the SRO will be required to recognize and assess the impact of the incomplete Group 8 PCIS isolation. TS 3.6.1.3 Condition A must be entered for the TIP Containment Isolation valves but permits continued operation in the MODE for an unlimited period of time.

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



HLT 0810/1006 Written Exam

81. 295024 G2.1.31

Unit 2 is in Mode 4 preparing to go to Mode 2. An air leak in the Drywell result in Drywell Pressure of 2.5 psig. The following indications are observed on the Containment Isolation Status System (CISS) on Panel 2-9-4:

- Groups 2 **AND** 6 PCIS Logic Success Lights are illuminated
- Groups 1, 3, 4, 5 **AND** 8 PCIS Logic Success Lights are **NOT** illuminated

The leak is subsequently isolated **AND** Drywell Pressure restored to normal.

Based on these indications, which ONE of the following completes the statements?

The Unit Supervisor must direct __**(1)**__ valves be closed.

In accordance with Tech Specs, Unit 2 __**(2)**__ permitted to change Modes to Mode 2.

[REFERENCE PROVIDED]

A. **(1)** Traversing Incore Probe Ball **AND** Purge
(2) is

B. **(1)** Reactor Water Cleanup (RWCU) suction isolation **AND** return isolation
(2) is

C. **(1)** Traversing Incore Probe Ball **AND** Purge
(2) is **NOT**

D. **(1)** Reactor Water Cleanup (RWCU) suction isolation **AND** return isolation
(2) is **NOT**

QUESTION 80

Unit 2 was operating at 100% power when at time 0805 a scram occurred due to a loss of both RPS Bus A and RPS Bus B. **NO** control rods initially inserted, and, after a manual scram and ARI, **NOT** all controls rods are fully inserted.

At 0817, the Shift Manager, as the Site Emergency Director, made an Emergency Plan event declaration in accordance with EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE.

The following conditions exist:

- Reactor power is UNKNOWN
- Reactor pressure is being maintained 800 to 1000 psig with two (2) SRVs OPEN and a third being manually cycled
- Reactor water level is being maintained -100 to -50 inches using HPCI
- Suppression pool temperature is 136°F and rising

Which ONE of the following completes the statement?

The event is a ___(1)___ and the State of Alabama must be notified by ___(2)___.

[REFERENCE PROVIDED]

A. (1) General Emergency
(2) 0832

B. (1) General Emergency
(2) 0835

C. (1) Site Area Emergency
(2) 0835

D. (1) Site Area Emergency
(2) 0832

Answer: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	295037 G2.4.30	
	Importance Rating		4.1

SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown
Knowledge of events relating to system operation/status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator.

Explanation: D CORRECT – The automatic scram, manual scram and ARI have failed to make the reactor subcritical. Power is greater than 5% since greater than 2 SRVs are required to control pressure. Suppression pool temperature has not exceeded HCTL (although it will do so based on present trends) and water level can be maintained above -180 inches, so a general emergency is not required at this time. The time for notification starts when the event classification is declared.

A Incorrect –First Part: Incorrect, plausible if the candidate believes that HCTL will be exceeded in the near future and the projection requires a general emergency. Second Part: Incorrect, plausible if the candidate believes that the notification clock starts with the plant conditions reaching the EAL (15 minutes to classify and 15 minutes to notify = 30minutes, 0805 +30= 0835).

B Incorrect – First Part: Incorrect, plausible if the candidate believes that HCTL will be exceeded in the near future and the projection requires a general emergency. Second Part: Correct.

C Incorrect –First Part: Correct. Second Part: Incorrect, plausible if the candidate believes that the notification clock starts with the plant conditions reaching the EAL(15 minutes to classify and 15 minutes to notify = 30minutes, 0805 +30= 0835).

Technical Reference(s): EPIP-1, EPIP-4

Proposed references to be provided to applicants during examination: EPIP-1

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New X

Question History: Previous NRC : No

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

SRO Only Justification: The SRO will assess plant conditions to determine that the automatic scram, manual scram and ARI have failed to reduce reactor power to < 5% since greater than 2 SRVs are required to control pressure. Additionally the SRO will have to evaluate Suppression pool temperature and Reactor Pressure to determine HCTL not been exceeded (although it will do so based on present trends). Also water level can be maintained above -180 inches, so a general emergency is not required at this time. From these assessments the SRO will use EPIP-1, Emergency Classification Procedure, to select EAL 1.2-S. Knowledge of the Emergency Notification step in EPIP-4 will be required for the SRO to determine that the time for notification starts when the event classification is declared.

SCRAM FAILURE	REACTOR COOLANT ACTIVITY	
Description	Description	
	1.3-U	
	Reactor coolant activity exceeds 26 µCi/gm dose equivalent I-131 (Technical Specification Limits) as determined by chemistry sample. OPERATING CONDITION ALL	UNUSUAL EVENT
1.2-A NOTE	1.3-A	
Failure of RPS automatic scram functions to bring the reactor subcritical AND Manual scram or ARI (automatic or manual) was successful OPERATING CONDITION: Mode 1 or 2	Reactor coolant activity exceeds 300 µCi/gm dose equivalent Iodine-131 as determined by chemistry sample. OPERATING CONDITION: Mode 1 or 2 or 3	ALERT
1.2-S NOTE		
Failure of automatic scram, manual scram, and ARI to bring the reactor subcritical. OPERATING CONDITION: Mode 1 or 2		SITE EMERGENCY
1.2-G CURVE US		
Failure of automatic scram, manual scram, and ARI. Reactor power is above 3% AND Either of the following conditions exists: • Suppression Pool temp exceeds HCTL. Refer to Curve 1.2-G. • Reactor water level can NOT be restored and maintained at or above -180 inches. OPERATING CONDITION: Mode 1 or 2		GENERAL EMERGENCY

3.2 State of Alabama Notification

NOTE

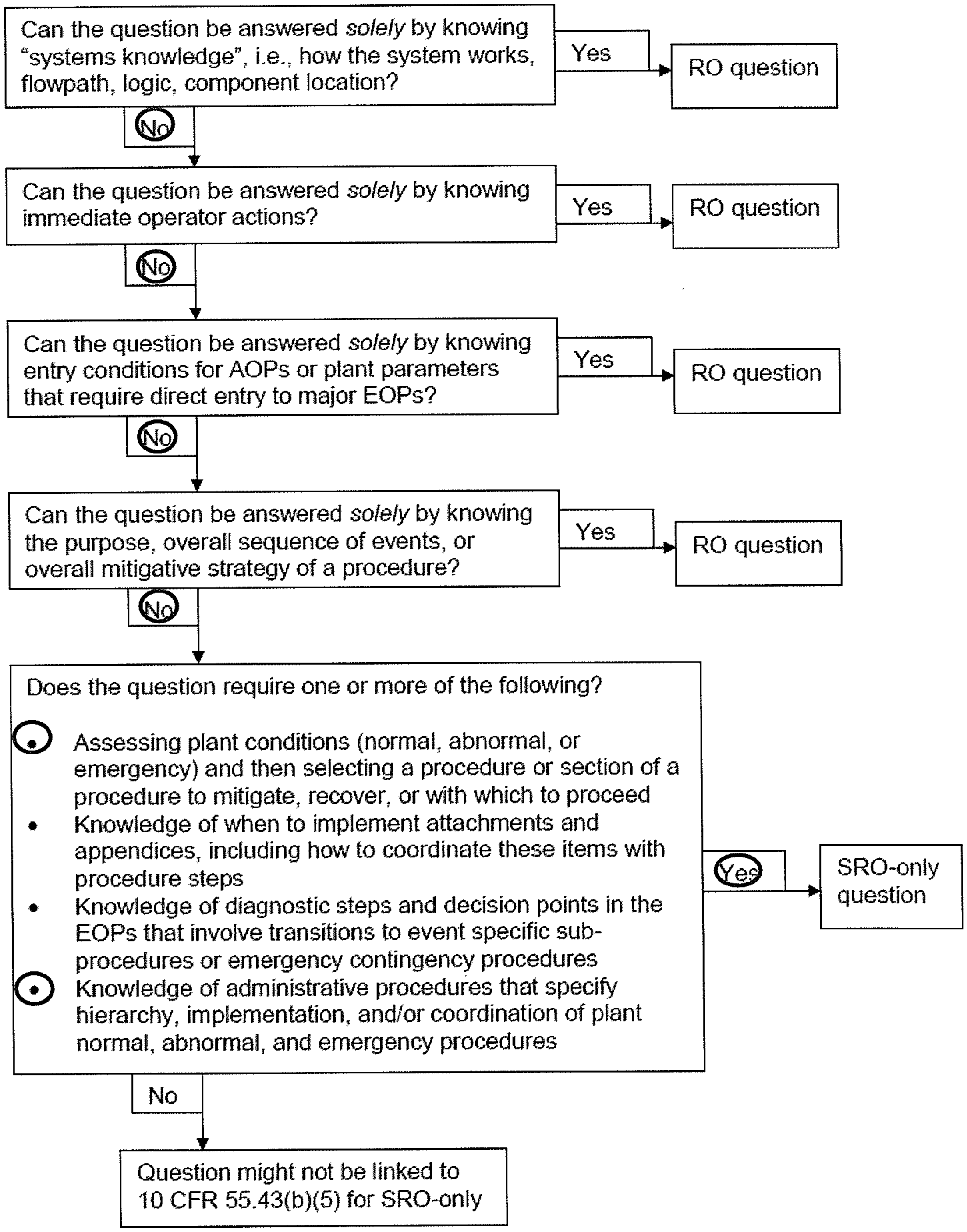
Notification of the State of Alabama is required to be completed within 15 minutes from the time of emergency classification declaration.

3.4 Notification of the Nuclear Regulatory Commission (NRC)

NOTE

Notification of the NRC is required to be completed as soon as possible not to exceed 60 minutes from classification declaration.

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



QUESTION 81

All three units are operating at 100% power when a fire is reported in the 4KV Shutdown Board "A" followed by an explosion.

RHR Pump 2B automatically started and then tripped with no operator action.

Given:

- 1-AOI-100-1, Manual Scram
- 0-SSI-9, Unit 2 Reactor Building Fire 4KV Electrical Board Room 2A
- 3-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations

Which ONE of the following describes the decision the Shift Manager will direct for each Unit's emergency/normal shutdown, and governing procedure for each unit?

A. Unit 1 Manual Scram governed by 1-AOI-100-1
Unit 2 Manual Scram governed by 0-SSI-9
Unit 3 Unit Shutdown governed by 3-GOI-100-12A

B. Unit 1 Manual Scram governed by 1-AOI-100-1
Unit 2 Manual Scram governed by 0-SSI-9
Unit 3 Manual Scram governed by 3-AOI-100-1

C. Unit 1 Manual Scram governed by 0-SSI-9
Unit 2 Manual Scram governed by 0-SSI-9
Unit 3 Manual Scram governed by 0-SSI-9

D. Unit 1 Manual Scram governed by 0-SSI-9
Unit 2 Manual Scram governed by 0-SSI-9
Unit 3 Manual Scram governed by 3-GOI-100-12A

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	600000 AA2.13	
	Importance Rating		3.8

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE
AA2.13 Need for emergency plant shutdown

Explanation: **C CORRECT** – Entry conditions of 0-SSI-1 are met for Unit 2 (i.e. any unit greater than atmospheric pressure AND multiple trains of safety related equipment threatened by fire in 4KV Shutdown Board C followed by an explosion). Unit 2 Control Room Operator Actions of 0-SSI-9 directs Units 1 and 3 to perform section 3.0 of 0-SSI-9 to Scram.

A Incorrect – Unit 1- incorrect for reasons stated above. Plausible in that this would be the appropriate procedure to insert a manual scram if the candidate knew that a manual scram was required from 0-SSI-9. Unit 2- Correct. Unit 3- Incorrect. Plausible if the candidate assumes a shutdown is required but doesn't know that 0-SSI-9 requires a scram.

B Incorrect – Unit 1- incorrect for reasons stated above. Plausible in that this would be the appropriate procedure to insert a manual scram if the candidate knew that a manual scram was required from 0-SSI-9. Unit 2- Correct. Unit 3- Incorrect. Plausible in that this would be the appropriate procedure to insert a manual scram if the candidate knew that a manual scram was required from 0-SSI-9.

D Incorrect –Unit 1 Correct. Unit 2 Correct. Unit 3- Incorrect. Plausible if the candidate assumes a shutdown is required but doesn't know that 0-SSI-9 requires a scram.

Technical Reference(s): 0-SSI-1, 0-SSI-9

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New : X

Question History: Previous NRC : No

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

SRO Only Justification: The Shift Manager is the ONLY one who can direct entry into the SSI's. The SRO will need to assess plant conditions , and select the correct procedure with which to proceed. In this case, the SRO will determine the conditions in the stem of the question indicate that a fire in a Unit 2 4kV board room is affecting safety related equipment associated with Unit 2. The SRO will also need to have a working knowledge of 0-SSI-1, Safe Shutdown Instructions, well enough to know that this procedure will direct entry into 0-SSI-9, Unit 2 Reactor Building Fire 4KV Electrical Board Room 2A, and direct all units to shutdown via manual scram.

NOTE
The decision to trip the unit(s) and declare an Appendix R fire is left to the judgment of the Shift Manager, or designee, and must be based on the magnitude of the fire and potential affect on the safe shutdown equipment necessary to achieve and maintain cold shutdown. Table 2 provides high pressure systems and available reactor water level instrumentation.

3.0 ENTRY CONDITIONS

The sub-instructions are only to be entered when the following conditions are present.

A. Unit 1, Unit 2 or Unit 3 reactor is greater than atmospheric pressure,

AND

B. The Magnitude of the fire has the potential to affect safe shutdown capability as indicated by:

Multiple failures/spurious actuations of systems/components have occurred,

OR

Erratic or questionable indications on numerous MCR instruments have occurred,

OR

Multiple trains/channels of safety related equipment are threatened by the fire.

INSTRUCTOR NOTES

X. Lesson Body

A. Purpose

The purpose of Safe Shutdown Instructions (SSI) is to provide the actions to ensure safe shutdown of Units 1, 2, and 3 in the event of a major disabling fire. 10 CFR 50 Appendix R transient analyses assume the plant is at full power with normal reactor water level and normal Torus temperature equal to or less than 95°F. The decision to trip the units and declare entry into the SSIs is the judgment of the Shift Manager and is based on the magnitude of the fire and potential affect on credited equipment listed in Illustrations 1 and 2 necessary to achieve and maintain cold shutdown. The SSI analysis is based on a single spurious

Step through 0-SSI-001

BFN Unit 0	Unit 2 Reactor Building Fire 4KV Electrical Board Room 2A	0-SSI-9 Rev. 0013 Page 6 of 136

INITIALS

2.0 UNIT 2 CONTROL ROOM OPERATOR ACTIONS

CAUTION
Implementation of Steps 2.0[1] through Step 2.0[21] within 20 minutes is imperative to MINIMIZE the consequences of uncovering the fuel on Unit 2.

TBD-2

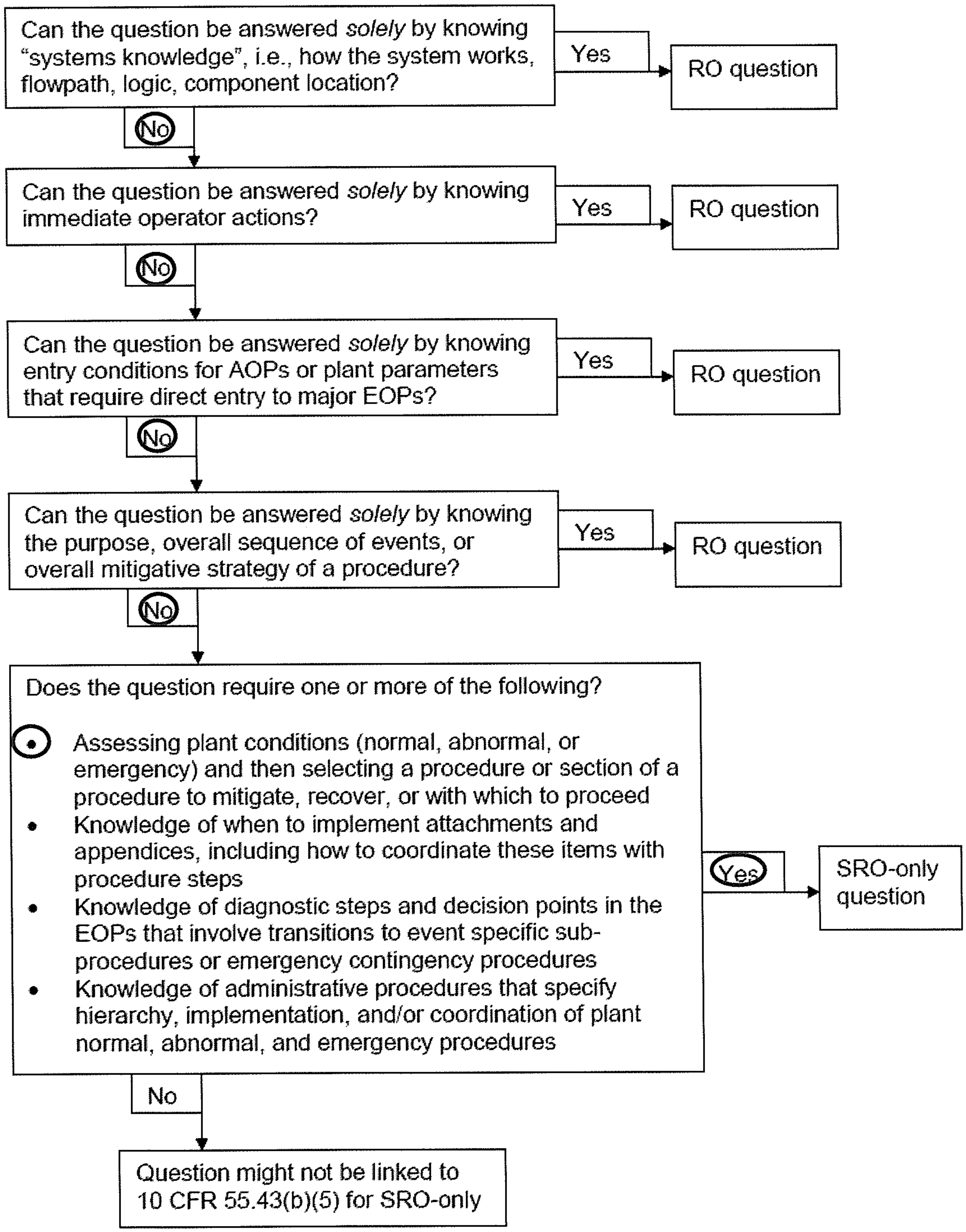
[1] **DIRECT** Unit 3 Unit Supervisor to perform Section 3.0 of 0-SSI-9 to Scram Unit 3, AND **PROCEED TO** cold shutdown. _________

[2] **DIRECT** Unit 1 Unit Supervisor to perform Section 4.0 of 0-SSI-9 to Scram Unit 1, AND **PROCEED TO** cold shutdown. _________

NOTE
The following instruments are those which have been credited for safe shutdown, and must be referenced when executing manual actions for this fire zone:
2-LI-3-58A and 2-PI-3-74A for reactor level and pressure.
TBD-81
2-LI-64-159A and 2-TI-64-161 for the suppression pool level and temperature.

(0 Min)

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



QUESTION 82

Unit 1 is at 90% power, Unit 2 is in MODE 5 performing a refueling outage, and Unit 3 is at 100% power, when severe weather in the area causes grid instabilities.

You are the Unit Supervisor and the following conditions exist on Unit 1:

- Incoming Mvars are 150 MVAR
- Grid voltage is 505 kV on the 500kV bus
- Grid voltage is 161kV on the 161kV bus
- Grid frequency is fluctuating from 59.97Hz to 60.03Hz

The Transmission Operator has notified Browns Ferry that the grid conditions are RED for the 500kV system and YELLOW for the 161kV system.

Which ONE of the following completes the statements?

The required action per 0-AOI-57-1E, Grid Instability, is to ___(1)___.

The required Technical Specification action(s) is/are to ____(2)____.

A. (1) RAISE reactive power until system voltage returns to 510 KV OR UNTIL Generator Reactive Power reaches +300 MVAR
(2) declare ALL offsite circuits Inoperable

B. (1) RAISE reactor power by approximately 1%/minute (10 MW(e)/minute) UNTIL system frequency returns to 59.98 Hz
(2) declare the 161 kV offsite circuits Inoperable, ALL 500 kV offsite circuits remain Operable

C. (1) RAISE reactive power until system voltage returns to 510 KV OR UNTIL Generator Reactive Power reaches +300 MVAR
(2) declare ALL 500kV offsite circuits Inoperable, the 161 kV offsite circuits remain Operable

D. (1) RAISE reactor power by approximately 1%/minute (10 MW(e)/minute) UNTIL system frequency returns to 59.98 Hz
(2) declare ALL offsite circuits Inoperable

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		1
	K/A#	700000 AA2.05	
	Importance Rating		4.7

700000 Generator Voltage and Electric Grid Disturbances AA2.05 Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRICAL GRID DISTURBANCES: Operational status of offsite circuit

Explanation: **C** CORRECT – First Part: The required minimum steady-state operating voltage at the BFN 500-kV bus is provided in TRO-TO-SOP-30.128 applicable Appendix A (distinguished by how many BFN units are tied to the grid). However, the required minimum will be no lower than 515 kV. When a BFN 500-kV bus voltage goes below 515kV and is not corrected within 15 minutes, the TOp shall inform the BFN Generator Operator within 30 minutes from the beginning of the event, that the offsite power source is disqualified (status **RED**).

A Incorrect – Incorrect. The required minimum will be no lower than 515 kV for the 500kV system and 161kV for the 161kV system. Therefore the 161kV system is still qualified. Plausible if the candidate believes that the 161 kV limit is above 161kV just as the 500 kV limit is above 500 kV.

B Incorrect – Incorrect. The required minimum will be no lower than 515 kV for the 500kV system and 161kV for the 161kV system. Therefore the 161kV system is still qualified. Plausible if the candidate believes that the 161 kV limit is above 161kV just as the 500 kV limit is above 500 kV.

D Incorrect – Incorrect. The **300 Mvar maximum outgoing limit** applies to all three units for both 500-kV and 161-kVoffsite power source qualification. However, there is no Mvar absorption (incoming) limit for offsite power source qualification. Plausible since 320Mvars stated in the stem would exceed the outgoing limit.

Technical Reference(s): TRO-TO-SOP-30.128

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

TRO-TO-SOP-30.128 Browns Ferry Nuclear Plant (BFN) Grid Operating Guide

3.2 Program Elements

The grid supplies "offsite power" to BFN at both the 500-kV and 161-kV yards. Offsite power must be capable of safely shutting down the plant in the event of a postulated accident and is legally required for plant operation. If the grid cannot assure the ability to meet the plant's voltage requirements for a postulated accident, the offsite power source is "disqualified," and the plant may be legally compelled to shut down if offsite power qualification is not restored within a limited amount of time.

<u>Transmission Operator</u>

Continually assesses the condition of the BFN offsite power sources and promptly corrects any problems or notifies the plant in accordance with the instructions in this guide and other applicable procedures; informs the BFN Generator Operator of any necessary restrictions on cooling tower load level or alignment; schedules system outages to accommodate plant activities and equipment status to the extent practicable; and informs the BFN Generator Operator of real-time and emerging changes to the offsite power supply grid status color code.

3.2 Program Elements

The grid supplies "offsite power" to BFN at both the 500-kV and 161-kV yards. Offsite power must be capable of safely shutting down the plant in the event of a postulated accident and is legally required for plant operation. If the grid cannot assure the ability to meet the plant's voltage requirements for a postulated accident, the offsite power source is "disqualified," and the plant may be legally compelled to shut down if offsite power qualification is not restored within a limited amount of time.

Three-way communication is required between BFN and the TOp to ensure that the offsite power qualification status is properly assessed and that any necessary corrective actions are promptly initiated. BFN must inform the TOp of any changes in the voltage regulator or capacitor bank status (remote indication is not available). BFN must also inform the TOp of the request to be considered aligned for 161-kV delayed source, see section 5.3.

A color code system is also used to communicate the status of the offsite power sources, both for current conditions and for postulated grid contingencies. Transmission System Operations, Engineering Analysis group uses a web page to communicate the forecasted status in the Operations Planning time horizon. The TOp verbally informs the plant of status color code changes in real-time.

BFN Cooling Tower loading was incorporated into this study such that the loading is not a factor in determining the offsite power grid status from this guide.

3.2.15 Grid Offsite Power Status Color Codes

A color code system is used to communicate the status of each offsite power source, based on the grid's ability to support offsite power requirements for current grid conditions as well as following a worst-case transmission contingency. Figure 1 below illustrates the color code decision process.

A **RED** offsite power grid status means that the source cannot provide qualified offsite power for the given configuration. If the grid offsite power status is **RED** for actual grid conditions and is not resolved within the 15-minute grace period, the Transmission System Operator must notify the BFN Generator Operator within 30 minutes from the beginning of the event, that offsite power requirements cannot be met for that offsite power source.

3.2.11 Voltage Schedules

BFN minimum bus voltage requirements will vary based on system and plant conditions, as determined in the Appendices. There are no upper voltage limits for offsite power qualification.

The required minimum steady-state operating voltage at the BFN 500-kV bus is provided in applicable Appendix A (distinguished by how many BFN units are tied to the grid). The required minimum will be no lower than 515 kV.

The official bus voltage for BFN 500 kV source is from bus 1 at BFN.

The required minimum steady-state operating voltage at the BFN 161-kV bus is provided in applicable Appendices B, C, or D (distinguished by *immediate* or *delayed* source, BFN capacitor bank and 161 kV yard tied or split status). The required minimum will be no lower than 161-kV.

The official bus voltage for BFN 161 kV source is from bus 1 at BFN 161 kV yard.

3.2.7 BFN Generator Mvar Limits

A **300 Mvar maximum outgoing limit** applies to all three units for both 500-kV and 161-kV offsite power source qualification.

There is no Mvar absorption (incoming) limit for offsite power source qualification.

NOTE

If the **300 Mvar** outgoing Mvar limit is exceeded for a unit and is not corrected within 15 minutes, the TOp shall inform the BFN Generator Operator within 30 minutes from the beginning of the event that both offsite power sources are disqualified for the unit that is exceeding the limit (status **RED**) and notify Engineering Analysis in accordance with TRO-TO-SOP-30.301.

Offsite power qualification is not impaired for the unit(s) whose outgoing Mvars are under the limit.

BASES

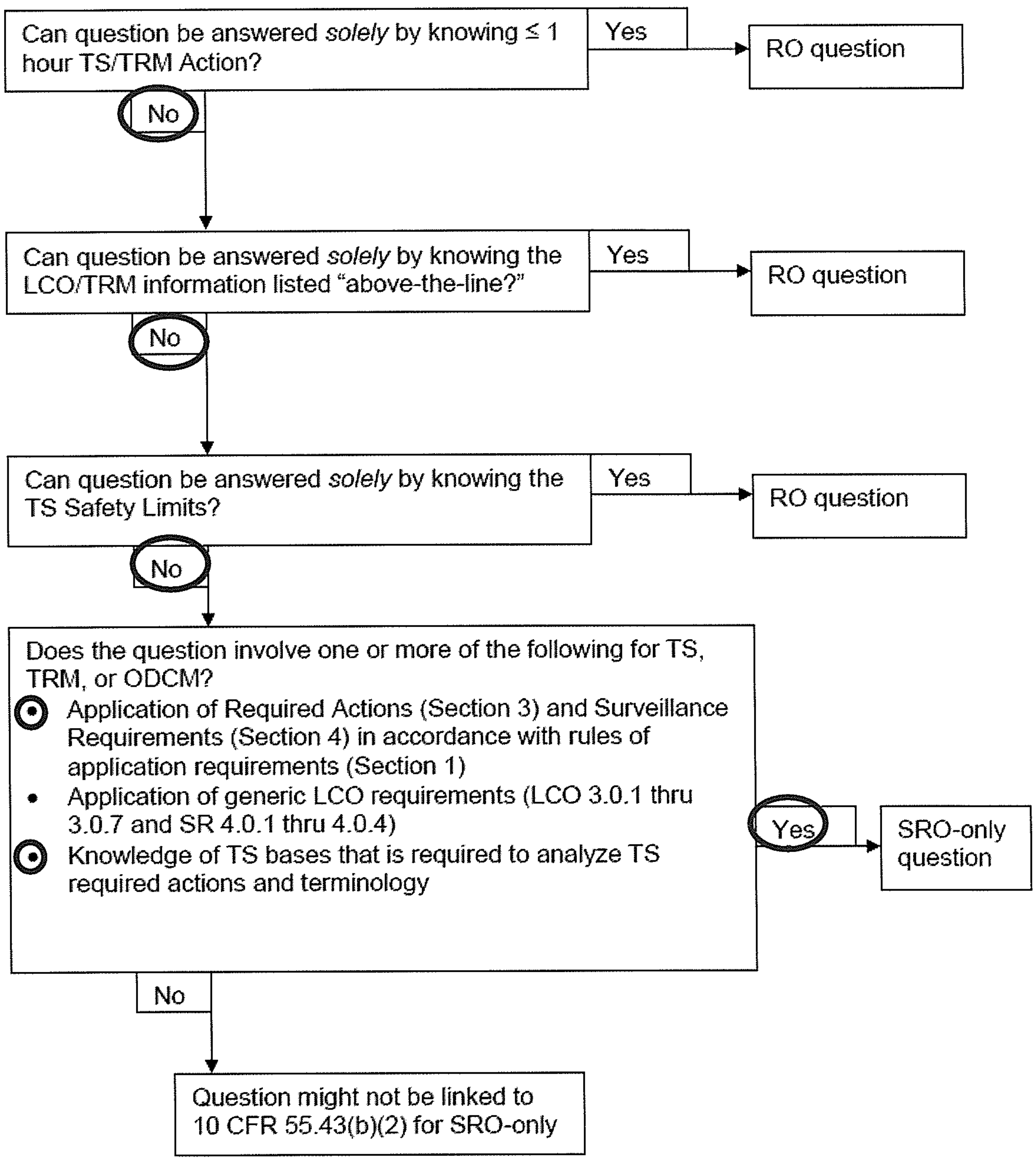
LCO
(continued)

Minimum required switchyard voltages are determined by evaluation of plant accident loading and the associated voltage drops between the transmission network and these loads. These minimum voltage values are provided to TVA's Transmission Operations for use in system studies to support operation of the transmission network in a manner that will maintain the necessary voltages.

Transmission Operations is required to notify BFN Operations if it is determined that the transmission network may not be able to support accident loading or shutdown operations as required by 10 CFR 50, Appendix A, GDC-17. Any offsite power circuits supplied by that transmission network cannot be credited as a qualified offsite circuit and are inoperable.

SRO Only Justification: The SRO will need to assess the current grid conditions to determine why the Transmission Operator is declaring the grid RED for the 500kV system and YELLOW for the 161kV system. He will need to have knowledge of the TRO-TO-SOP-30.128 Browns Ferry Nuclear Plant (BFN) Grid Operating Guide, limits. From that he will need to know the information in the bases of Technical Specification 3.8.1 regarding the qualification of offsite sources based on this information, and determine that only the 500kV system is NOT Operable for TS 3.8.1.

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



QUESTION 83

Unit 2 was operating at 90% when operators inserted a manual reactor scram due to rising secondary containment temperatures and radiation levels as the result of an unisolable Main Steam Line break in the Turbine Building.

The following conditions exist:

- All Rods are fully inserted
- Reactor pressure is 800 psig and slowly lowering
- Reactor Water Level is being maintain +2 to +51 inches with RCIC
- Stack Noble gas (WRGERMS) release rate is 6.1X 10^{10} μCi/sec, and steady for the past 45 minutes
- Field Assessment Teams report the Site Boundary Dose rate is 950 mR/hr gamma, and steady for the past hour

Which ONE of the following completes the statements?

The crew ____(1)____ required to emergency depressurize the reactor.

The crew is required to declare a ___(2)_____ .

[REFERENCE PROVIDED]

A. (1) is
(2) General Emergency

B. (1) is
(2) Site Area Emergency

C. (1) is NOT
(2) General Emergency

D. (1) is NOT
(2) Site Area Emergency

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295017 AA2.01	
	Importance Rating		4.2

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Off-site release rate: Plant-Specific

Explanation: **A** CORRECT -When site boundary dose rates reach EPIP General Emergency classification level, and emergency depressurization will help reduce the discharge into primary and secondary containment, 0-EOI-4 requires 2-EOI-1 be entered and Emergency Depressurization is Required.

B- Incorrect - First part is correct. Second part is incorrect because the General Emergency Limits are met for EAL 4.1-G (Stack Noble gas (WRGERMS) release rate is >5.9 X 10^{10} µCi/sec, and steady for the past 15 minutes)

C- Incorrect - First part is incorrect because site boundary dose rates have reached EPIP General Emergency classification level and emergency depressurization will help reduce the discharge into primary and secondary containment. Therefore, per 0-EOI-4, 2-EOI-1 is entered and Emergency Depressurization is Required. Second part is correct.

D- Incorrect - First part is incorrect because site boundary dose rates have reached EPIP General Emergency classification level and emergency depressurization will help reduce the discharge into primary and secondary containment. Therefore, per 0-EOI-4, 2-EOI-1 is entered and Emergency Depressurization is Required. Second part is incorrect because the General Emergency Limits are met for EAL 4.1-G (Stack Noble gas (WRGERMS) release rate is >5.9 X 10^{10} µCi/sec, and steady for the 15minutes)

Technical Reference(s): 0-EOI-4, EPIP-1

Proposed references to be provided to applicants during examination: EPIP-1

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Hatch 2009 question #86
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

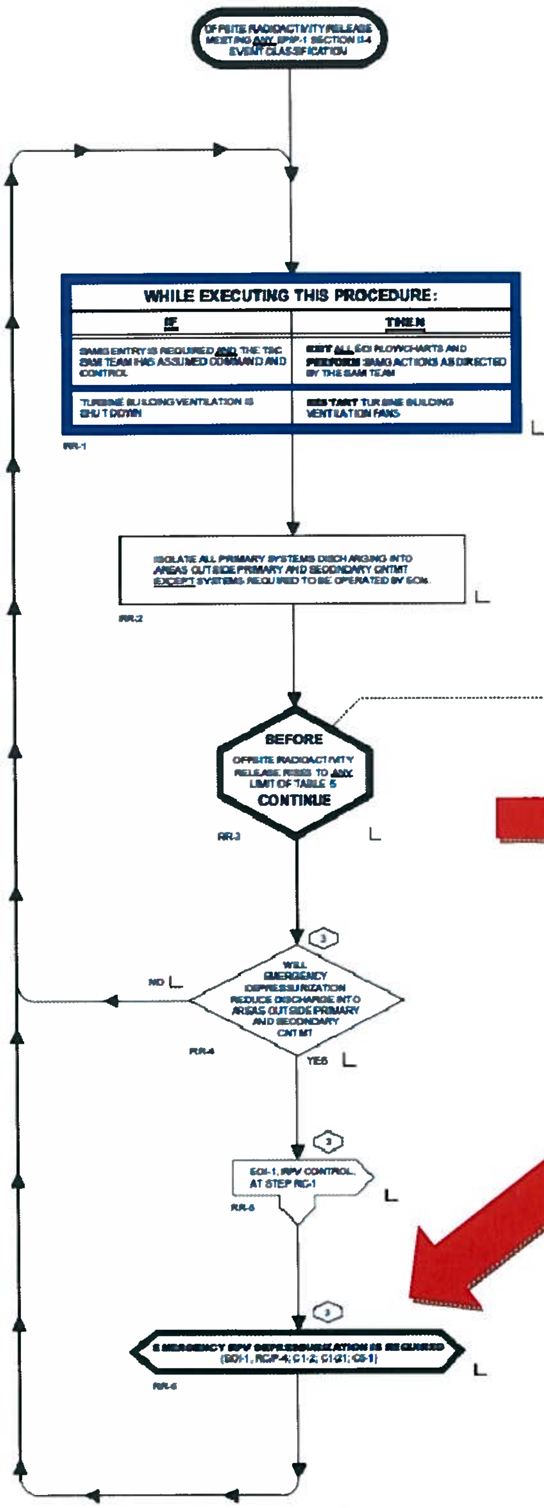


TABLE 5
OFFSITE RADIOACTIVITY RELEASE CLASSIFICATION LIMITS FOR GENERAL EMERGENCY

TYPE	MONITORING METHOD	LIMIT REFERENCE
GASEOUS RELEASE RATE	STACK NOBLE GAS RELEASE (WRGERMS)	BFN EPIP GENERAL EMERGENCY CLASSIFICATION LIMIT
SITE BOUNDARY RADIATION READING	FIELD ASSESSMENT TEAM	BFN EPIP GENERAL EMERGENCY CLASSIFICATION LIMIT
SITE BOUNDARY IODINE-131	FIELD ASSESSMENT TEAM	BFN EPIP GENERAL EMERGENCY CLASSIFICATION LIMIT
ACTUAL OR PROJECTED DOSE CONSEQUENCES AT OR BEYOND THE SITE BOUNDARY (TEDE)	DOSE ASSESSMENT	BFN EPIP GENERAL EMERGENCY CLASSIFICATION LIMIT
ACTUAL OR PROJECTED DOSE CONSEQUENCES AT OR BEYOND THE SITE BOUNDARY (THYROID CDE)	DOSE ASSESSMENT	BFN EPIP GENERAL EMERGENCY CLASSIFICATION LIMIT

NOTES

4.1-U Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-U
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 2.0

If neither assessment can be conducted within 60 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-A Prior to making this emergency classification based upon the WRGERMS indication, assess the release by either of the following:

1. Actual field measurements exceed the limits in table 4.1-A
2. 0-SI 4.8.B.1.a.1 release fraction exceeds 200

If neither assessment can be conducted within 15 minutes then the declaration must be made on the valid WRGERMS reading.

4.1-S Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-S.
2. Projected or actual dose assessments exceed 100 mrem TEDE or 500 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

4.1-G Prior to making this emergency classification based upon the gaseous release rate indication, assess the release by either of the following methods:

1. Actual field measurements exceed the limits in table 4.1-G.
2. Projected or actual dose assessments exceed 1000 mrem TEDE or 5000 mrem CDE.

If neither assessment can be conducted within 15 minutes then the declaration must be made based on the valid WRGERMS reading.

CURVES/TABLES:

Table 4.1-U
RELEASE LIMITS FOR UNUSUAL EVENT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88×10^{7} µCi/sec	1 Hour
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 2.0	1 Hour
Site Boundary Radiation Reading	Field Assessment Team	0.10 MREM/HR Gamma	1 Hour

Table 4.1-A
RELEASE LIMITS FOR ALERT

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	2.88×10^{9} µCi/sec	15 Minutes
Gaseous Release Rate	0-SI 4.8.B.1.a.1	Release Fraction 200	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	10 MREM/HR Gamma	15 Minutes

Table 4.1-S
RELEASE LIMITS FOR SITE AREA EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9×10^{9} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	100 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9×10^{-7} µCI /cm^{3}	1 Hour

Table 4.1-G
RELEASE LIMITS FOR GENERAL EMERGENCY

TYPE	MONITORING METHOD	LIMIT	DURATION
Gaseous Release Rate	Stack Noble Gas (WRGERMS)	5.9×10^{10} µCi/sec	15 Minutes
Site Boundary Radiation Reading	Field Assessment Team	1000 MREM/HR Gamma	1 Hour
Site Boundary Iodine-131	Field Assessment Team	3.9×10^{-6} µCI / cm^{3}	1 Hour

GASEOUS EFFLUENT

Description

UNUSUAL EVENT

4.1-U | NOTE | TABLE

Gaseous release exceeds ANY limit and duration in Table 4.1-U.

OPERATING CONDITION:
ALL

ALERT

4.1-A | NOTE | TABLE

Gaseous release exceeds ANY limit and duration in Table 4.1-A.

OPERATING CONDITION:
ALL

SITE EMERGENCY

4.1-S | NOTE | TABLE

EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-S.
- Dose assessment indicates actual or projected dose consequences above 100 mrem TEDE or 500 mrem thyroid CDE.

OPERATING CONDITION:
ALL

GENERAL EMERGENCY

4.1-G | NOTE | TABLE

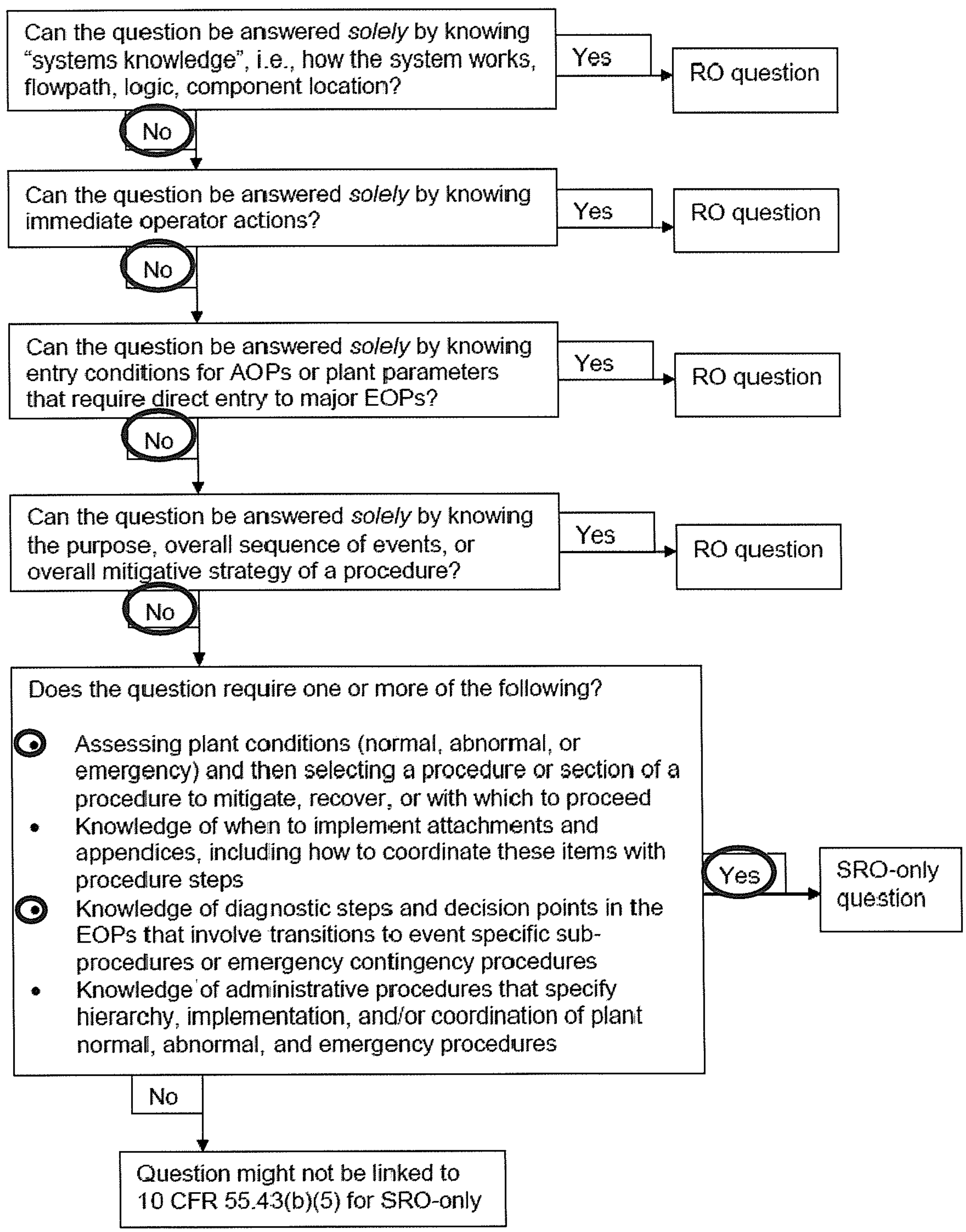
EITHER of the following conditions exists:

- Gaseous release exceeds or is expected to exceed ANY limit and duration in Table 4.1-G.
- Dose assessment indicates actual or projected dose consequences above 1000 mrem TEDE or 5000 mrem thyroid CDE.

OPERATING CONDITION
ALL

SRO Only Justification: The SRO will assess the given plant conditions and with a knowledge of diagnostic steps and decision points in EOI-4 determine that with an unisolable Main Steam Line break in the Turbine Building and therefore emergency depressurization will help reduce the discharge into primary and secondary containment, 0-EOI-4 will reuire 2-EOI-1 be entered and Emergency Depressurization is Required. Also, the SRO will use the Stack Noble gas (WRGERMS) release rate and EPIP-1 to determine that the General Emergency Limits are met for EAL 4.1-G.

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



Hatch 2009 question #86

86. 295017AA2.01 001

Unit 1 was at 80% power when an unisolable Main Steam Line break occurred in the Turbine Building.

The following conditions currently exist:

- o Prompt Offsite Dose results .... 1050 mR/hr peak TEDE
- o Reactor power .... All Rods Full-In
- o Reactor pressure .... 900 psig
- o Reactor Water Level .... 35 inches

Which ONE of the following choices completes both of the following statements?

IAW with 31EO-EOP-014-1, "Radioactivity Release Control" EOP flowchart, the crew __(1)__ required to emergency depressurize the reactor.

IAW 73EP-EIP-001-0, "Emergency Classification and Initial Actions", the crew __(2)__ required to declare a General Emergency.

A✓ (1) is
(2) is

B. (1) is
(2) is NOT

C. (1) is NOT
(2) is

D. (1) is NOT
(2) is NOT

At 1000 mR/hr the RR chart directs scramming and emergency depressing the reactor. EAL RG1 of 73EP-EIP-001-0 requires a General Emergency to be declared.

It is Only a Site Area Emergency on the Fission Product Barrier Chart.

QUESTION 84

Unit 2 was operating at 100% power when a LOCA occurred.

Current plant conditions are as follows:

- All control rods are inserted
- RPV water level is -135 inches and steady
- RPV pressure is 150 psig and lowering
- Core Spray pump 2A is injecting and is the only makeup source
- Drywell pressure is 32 psig and slowly rising
- Suppression Chamber Pressure is 30 psig and slowly rising
- Suppression Pool Water level is 19.5 feet and slowly rising

Based on these conditions, which ONE of the following actions is required to be directed by the Unit Supervisor?

A. Restore Suppression Pool water level to -1 to -6 inches using 2-EOI Appendix-18 Suppression Pool Water Inventory Removal and Makeup

B. Enter EOI C-2, RPV-Emergency Depressurization, and open all 6 ADS SRV's

C. Vent the Drywell irrespective of offsite radioactivity release per 2-EOI Appendix 13, Emergency Venting Primary Containment

D. Vent the Suppression Chamber irrespective of offsite radioactivity release rate per 2-EOI Appendix 13, Emergency Venting Primary Containment

ANSWER: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	295029EA2.03	
	Importance Rating		3.5

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Drywell/containment water level

Explanation: **B** CORRECT – 19.5 feet in the Suppression Pool and Suppression Chamber Pressure at 30 psig is in the action required region of Curve 6, Pressure Suppression Pressure (PSP). EOI-2 step PC/P-10 questions whether Suppression Chamber pressure can be maintained in the safe area of the curve. The answer is NO and Emergency Depressurization is Required (step PC/P-11).

A- Incorrect – This is plausible because 19.5 feet in the Suppression Pool and RPV pressure at 150 psig is in the action required region of Curve 4, SRV Tail Pipe Level Limit. The EOI-2 flowchart, SP/L-21 questions whether parameters can be <u>restored</u> and maintained in the safe region of the curve. In this case suppression pool level can be lowered to <u>restore</u> to the safe region. This is not the correct action for

C- Incorrect – This is plausible if the operator does not identify that Emergency Depressurization is required. Additionally In the previous revision of the EOI-2 flowchart, step PC/P-17 directs venting the DW irrespective of offsite radioactivity release.

D- Incorrect - This is plausible because EOI-2 step PC/P-13 directs venting the suppression chamber prior to 55psig (step PC/P-12).

Technical Reference(s): 2-EOI-2 Flowchart

Proposed references to be provided to applicants during examination: None

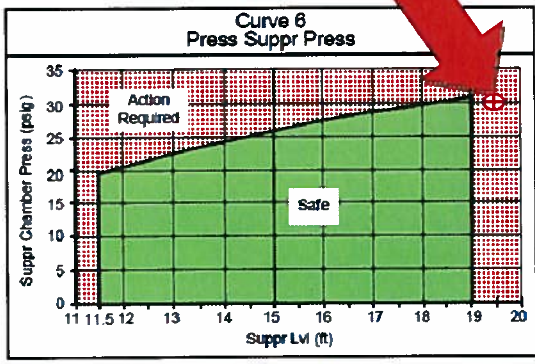
Learning Objective (As available):

Question Source: Bank:
Modified Bank: X
New:

Question History: Previous NRC: Vermont Yankee 2010 NRC #83

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.



2

WHEN suppr chmbr press CANNOT be maintained in the safe area of Curve 6

PC/P-10

EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-1, C1-20; C5-12, C5-14)

PC/P-11

BEFORE
suppr chmbr press rises to 55 psig
CONTINUE

PC/P-12

12

VENT the PC to control suppr chmbr press below 55 psig (APP 13)
(ok to exceed rad release rate limits)

PC/P-13

SRO Only Justification: The SRO will assess plant conditions and determine that 19.5 feet in the Suppression Pool and RPV pressure at 150 psig is in the action required region of Curve 6, Pressure Suppression Pressure (PSP). With klnowledge of the decision steps in 2-EOI-2, Primary Containment Control flowchart, the SRO will recognize that Emergency Depressurization is Required and select entry into EOI C-2, RPV-Emergency Depressurization as the procedure with which to proceed.

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

Does the question require one or more of the following?

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

Yes → SRO-only question

No

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

Vermont Yankee 2010 NRC #83

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295029 EA 2.03	
	Importance Rating		3.5

(K&A Statement) EA2.03- Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level

Proposed Question: SRO 83

Post large break LOCA conditions are as follows:

- All control rods are inserted
- RPV water level is -35 inches and steady
- RPV pressure is 150 psig and lowering
- Core Spray pump "A" is injecting as designed
- Drywell pressure is 32 psig and rising slow
- Torus Pressure is 30 psig and rising slow
- Torus Water level is 11.5 feet and rising slow

Based on these conditions, which ONE of the following actions is required to be directed by the CRS?

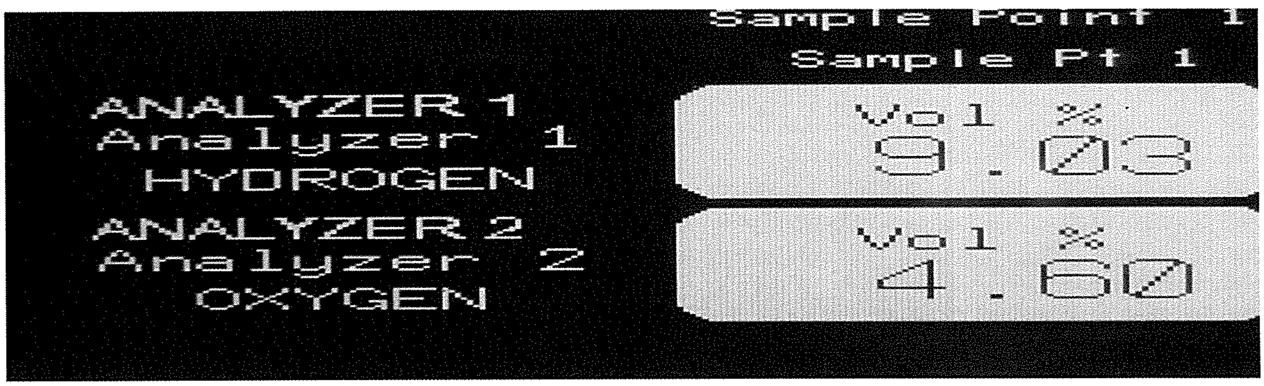
A. Enter the Severe Accident Guideline -1 and establish containment flooding
B. Enter EOP-5, "RPV-Emergency Depressurization" and open all SRVs
C. Vent Primary Containment prior to prevent exceeding PCPL-A IAW OE 3107, Appendix "HH" using the Torus Hardened vent (TVS-86)
D. Vent Primary Containment exceeding Off Site release rates IAW OE 3107, Appendix "HH" using both trains of SBGT vent through the Stack.

Proposed Answer: B

QUESTION 85

A loss of coolant accident has occurred on Unit 1. The following conditions exist:

- Suppression Pool level is 16 feet
- The H_2/O_2 Analyzer is aligned to the Suppression Chamber and has been in operation for 13 minutes
- H_2/O_2 concentrations are as indicated below:



Which ONE of the following completes the statements?

In accordance with 1-EOI-Appendix-19, H_2/O_2 Analyzer Operation, readings from 1-XR-76-110 H_2/O_2 Concentration Recorder (Panel 1-9-54) or from 1-MON-76-110, H_2/O_2 Analyzer (Panel 1-9-55) may only be obtained after ___(1)____.

Based on the current H_2/O_2 readings and in accordance with 1-EOI-2, PC/H leg, the crew is required to purge the Primary Containment with ___(2)____ .

A. (1) 15 minutes
(2) 1-EOI-Appendix 14A, N2 MAKEUP TO PRIMARY CONTAINMENT

B. (1) 15 minutes
(2) 1-EOI-Appendix 14B, CAD OPERATION

C. (1) 10 minutes
(2) 1-EOI Appendix 14A, N2 MAKEUP TO PRIMARY CONTAINMENT

D. (1) 10 minutes
(2) 1-EOI-Appendix 14B, CAD OPERATION

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		1
	Group #		2
	K/A#	500000 G2.2.44	
	Importance Rating		4.4

500000 High Containment Hydrogen Concentration
G2.2.44 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.

Explanation: **C** CORRECT – 1-EOI Appendix 19 requires the analyzer to be in service for at least 10 minutes before accurate readings can be obtained. Then 1-EOI-2, Primary Containment Control flowchart step PC/H-2 directs control of H_2 and O_2 using the Nitrogen Makeup System, 1EOI-Appendix 14A.

A Incorrect - First part is Incorrect. Plausible if the candidate does not know that 1-EOI-appendix 19 directs 10 minutes of operation prior to obtaining accurate concentrations. 15 minutes is a common timeframe for other system operations. Second part is Correct.

B Incorrect - First part is Incorrect. Plausible if the candidate does not know that 1-EOI-appendix 19 directs 10 minutes of operation prior to obtaining accurate concentrations. 15 minutes is a common timeframe for other system operations. Second part is Incorrect, however plausible if the candidate believes that nitrogen addition will be from the CAD system during 1-EOI-2.

D Incorrect - First part is Correct. Plausible if the candidate does not know that 1-EOI-appendix 19 directs 10 minutes of operation prior to obtaining accurate concentrations. Second part is Incorrect, however plausible if the candidate believes that nitrogen addition will be from the CAD system during 1-EOI-2.

Technical Reference(s):1-EOI-2, 1-EOI Appendix 19, 1EOI-Appendix 14A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis :	X

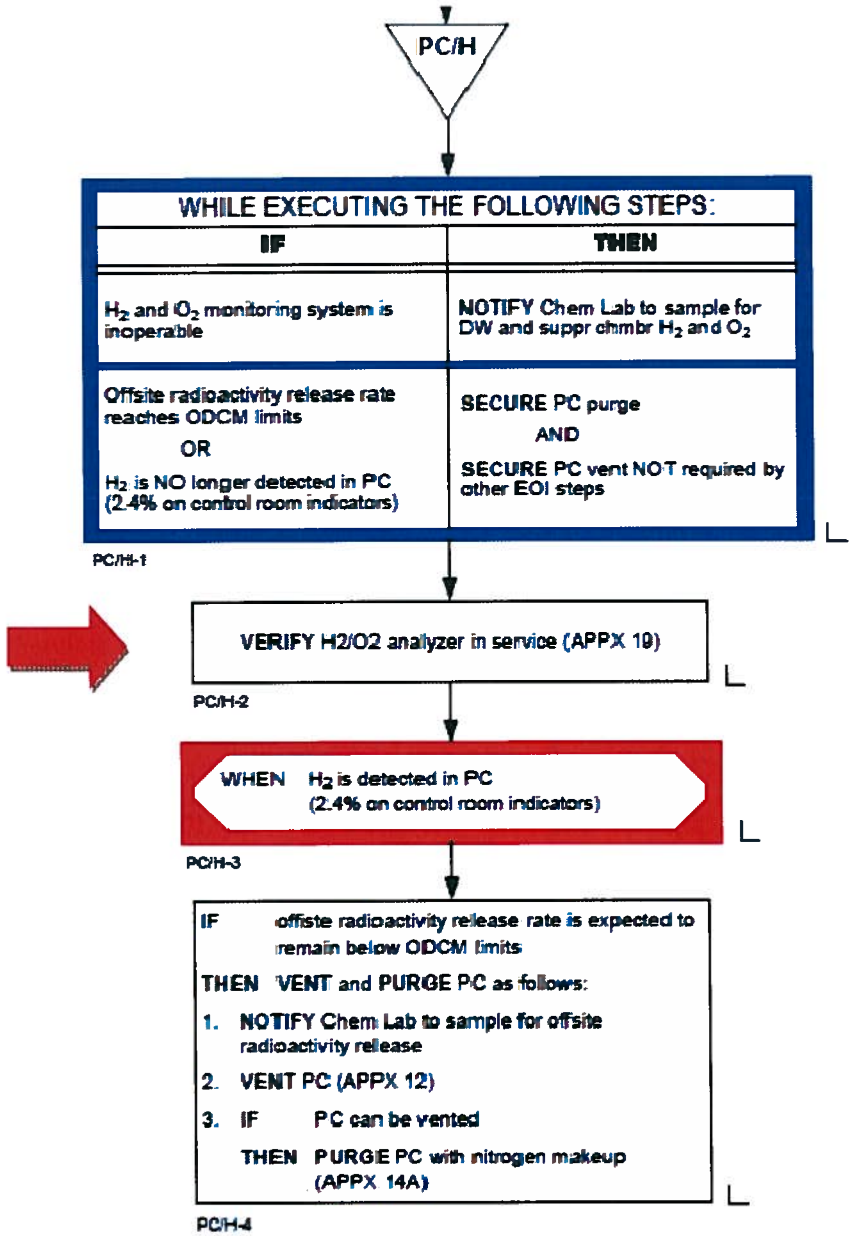
10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

7. **VERIFY** H2/O2 ANALYZER SAMPLE PUMP running using 1-XI-76-110 (Panel 1-9-55). ______

8. **VERIFY** LOW FLOW and TROUBLE indicating lights are extinguished on 1-MON-76-110, H2/O2 ANALYZER (Panel 1-9-55). ______

9. WHEN .... H2/O2 Analyzer has been aligned and sampling for 10 minutes or greater,

   THEN........... **OBTAIN** H2 and O2 readings from 1-XR-76-110 H2/O2 CONCENTRATION recorder (Panel 1-9-54) or from 1-MON-76-110, H2/O2 ANALYZER (Panel 1-9-55). ______



SRO Only Justification: The SRO will assess plant conditions and with a knowledge of 1-EOI Appendix 19, H_2O_2 Analyzer Operation, and 1-EOI-2, Primary Containment Control flowchart, determine that the analyzer reading is valid and the correct procedure to mitigate/ recover would be Nitrogen Makeup System, 1EOI-Appendix 14A.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

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

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

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

Yes → RO question

No ↓

Does the question require one or more of the following?

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

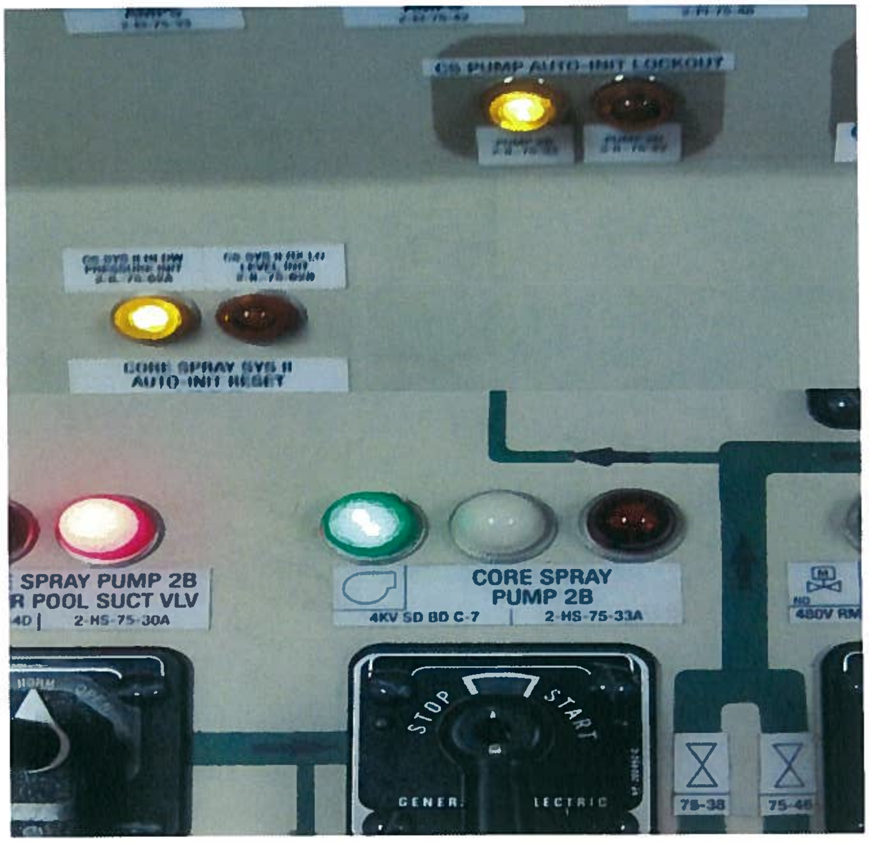
Yes → SRO-only question

No ↓

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

QUESTION 86

A startup was in progress on Unit 2 with a nuclear heatup in progress and Reactor Pressure at 550 psig when a LOCA occurred. The reactor scrammed automatically and the following indications now exist:



Which ONE of the following completes the statement?

Core Spray Pump 2B was ___(1)___ and the Unit Supervisor will direct ___(2)___ to be used for Core Spray Loop II injection.

A. (1) manually stopped
(2) 2-OI-75, CORE SPRAY SYSTEM

B. (1) manually stopped
(2) Appendix 6E, INJECTION SUBSYSTEM LINEUP CORE SPRAY SYSTEM II

C. (1) automatically tripped
(2) 2-OI-75, CORE SPRAY SYSTEM

D. (1) automatically tripped
(2) Appendix 6E, INJECTION SUBSYSTEM LINEUP CORE SPRAY SYSTEM II

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	209001 G2.1.31	
	Importance Rating		4.3

209001 Low Pressure Core Spray System
2.1.31. Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Explanation: **B** CORRECT – First Part: Correct. The pump has been manually stopped since the AMBER light on the vertical section of 9-3 is illuminated. Second Part: Correct. The amber light indicates that the pump has received an accident signal and the EOI are being implemented. The EOIs take precedence over the OIs.

A Incorrect – First Part: Correct. The pump has been manually stopped since the AMBER light on the vertical section of 9-3 is illuminated. Second Part: Incorrect. Plausible since the OIs have a section on auto initiation.

C Incorrect – First Part: Incorrect: The AMBER light on the vertical section of 9-3 being illuminated indicates that the pump handswitch has been placed in STOP with an accident signal present (i.e. manually overridden OFF). Plausible- There has been an auto start of this pump and it is not running now, therefore it is plausible that it tripped automatically. Second Part: Correct.

D Incorrect – First Part: Incorrect: The AMBER light on the vertical section of 9-3 being illuminated indicates that the pump handswitch has been placed in STOP with an accident signal present (i.e. manually overridden OFF). Plausible- There has been an auto start of this pump and it is not running now, therefore it is plausible that it tripped automatically. Second Part: Incorrect. Plausible since the OIs have a section on auto initiation.

Technical Reference(s): 2-EOI-1, Appendix 6E

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:
	Modified Bank:
	New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

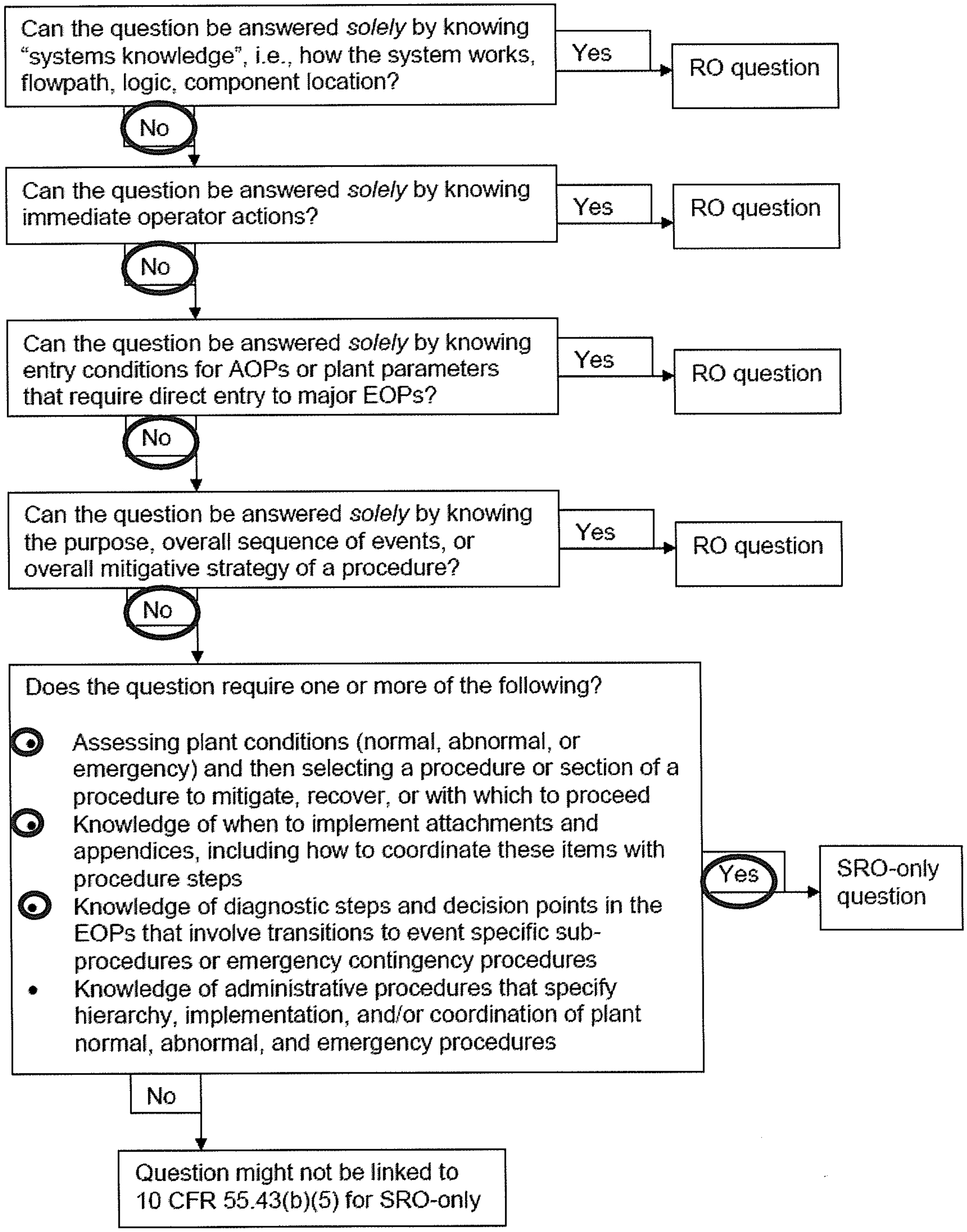
pumps would automatically start.

2) If a CS pump is stopped with an initiation signal present, it will not automatically start again until the CS initiation logic is reset. This condition is indicated by an amber light on Panel 9-3 (vertical section). This is accomplished by the stop handswitch energizing K21 which blocks auto start signals until the CS initiation signal is reset dropping out K25. Note that K21 is also deenergized by a loss and restoration of SD boards. (Loss of offsite power with initiation signal present) such that all pumps will restart when the DGs repower the boards.

TP-6. 8

SRO Only Justification: The SRO will need to use the pictured control room switches, controls, and indications to assess the plant conditions, and determine that they correctly reflect the desired plant lineup. From that information, the SRO will correctly select the procedure or section of a procedure to mitigate, recover, or with which to proceed, in this case Appendix 6E, INJECTION SUBSYSTEM LINEUP CORE SPRAY SYSTEM II.

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



QUESTION 87

Unit 1 is in MODE 3 with all rods fully inserted, following a scram from rated conditions. Reactor pressure is 940 psig and stable.

While completing post scram actions, annunciator 9-5B window #13, SLC TEMP ABNORMAL, alarms. The AUO reports that local tank temperature is 53° F and the breakers for the heaters are tripped.

SLC storage tank boron concentration is 8.05%.

Given that the temperature will continue to drop while troubleshooting the tripped breakers, which ONE of the following completes the statement?

There are ___(1)___ available before reactor coolant temperature must be less than 212 degrees, AND given these plant conditions, the bases for this action is that__(2)___ .

[REFERENCE PROVIDED]

A. (1) 32 hours
(2) the standby liquid solution may not meet its design criteria in response to an ATWS

B. (1) 44 hours
(2) the standby liquid solution may not meet its design criteria in response to an ATWS

C. (1) 32 hours
(2) the standby liquid solution may not meet its design criteria to maintain the proper pH in the suppression pool to control the accident source term following a LOCA

D. (1) 44 hours
(2) The standby liquid solution may not meet its design criteria to maintain the proper pH in the suppression pool to control the accident source term following a LOCA

ANSWER: **D**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	211000 A2.05	
	Importance Rating		3.4

211000 Standby Liquid Control System
A2.05 Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of SBLC tank heaters

Explanation: **D** CORRECT – The allowable time is 44 hours. Mode 4 must be achieved due to the LOCA concern.

A- Incorrect – The first part, 32 hours is plausible because candidate needs to know that even though the plant is currently in mode 3 due to the scram, the 12 hours that would have been used to achieve MODE 3 is still available to achieve mode 4 as discussed on page 1.3-4 of the tech spec. bases on completion times (8 hours of condition B plus only 24 hours of condition C). The second part, the plant is in MODE 3 and SLC is not required to perform its shutdown capability function during MODES 3, 4, or 5.

B- Incorrect – First part is correct. Second part, the plant is in MODE 3 and SLC is NOT required to perform its shutdown capability function during MODES 3, 4, or 5. This is plausible because the shutdown capability function is required in modes 1 and 2.

C- Incorrect - The first part, 32 hours is plausible because candidate needs to know that even though the plant is currently in mode 3 due to the scram, the 12 hours that would have been used to achieve MODE 3 is still available to achieve mode 4 as discussed on page 1.3-4 of the tech spec. (8 hours of condition B plus only 24 hours of condition C). Second part, the plant is in MODE 3 and SLC is not required to perform its shutdown capability function during MODES 3, 4, or 5. Both parts are plausible for the reasons stated above.

Technical Reference(s): Unit 1Tech Spec 3.1.7 and bases

Proposed references to be provided to applicants during examination: Unit 1 Tech Spec 3.1.7

Learning Objective (As available):

Question Source:	Bank: X Modified Bank: New:
Question History:	Previous NRC: Nile Mile Point U2 2010 NRC #86
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 5.43(b) 2 Facility operating limitations in the technical specifications and their bases.

SLC TEMP ABNORMAL 1-TA-63-3 13

(Page 1 of 1)

Sensor/Trip Point:

1-TS-063-0003	HI-105°F	LO-53.7°F suction pipe
1-TS-063-0004	HI-105°F	LO-53.7°F suction pipe

Sensor Location: 1-LPNL-925-0019
Elevation 639

Probable Cause:

A. Power supply to heaters is open (480V Rx vent Bd. 1A Bkr. 6D).
B. Cleared fuses at 480V Rx vent Bd. 1A.
C. Heater malfunction.
D. Sensor malfunction.
E. Ambient temperature below normal.
F. LC 109 BKR 5 off/tripped.

Automatic Action: None

Operator Action:

A. **DISPATCH** personnel to the standby liquid control tank to DETERMINE if temperature is low in the suction piping or the tank. ☐
B. **CHECK** Bkr 6D on 480V Rx Vent Bd 1A and **VERIFY** SLC tank heaters energized. ☐
C. **CHECK** Lighting Cabinet 109 BKR 5 in ON. ☐
D. **REFER TO** Technical Specification sect. 3.1.7. ☐

References: 1-45E620-6-2 1-47E610-63-1 1-45E779-3 1-729E854-1, -2

BASES (continued)

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure that the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System shutdown capability is not required to be OPERABLE when only a single control rod can be withdrawn.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure offsite doses remain within 10 CFR 50.67, "Accident Source Term," limits following a LOCA involving significant fission product releases. The SLC System is used to maintain suppression pool pH above 7 following a LOCA to ensure iodine is retained in the suppression pool water.

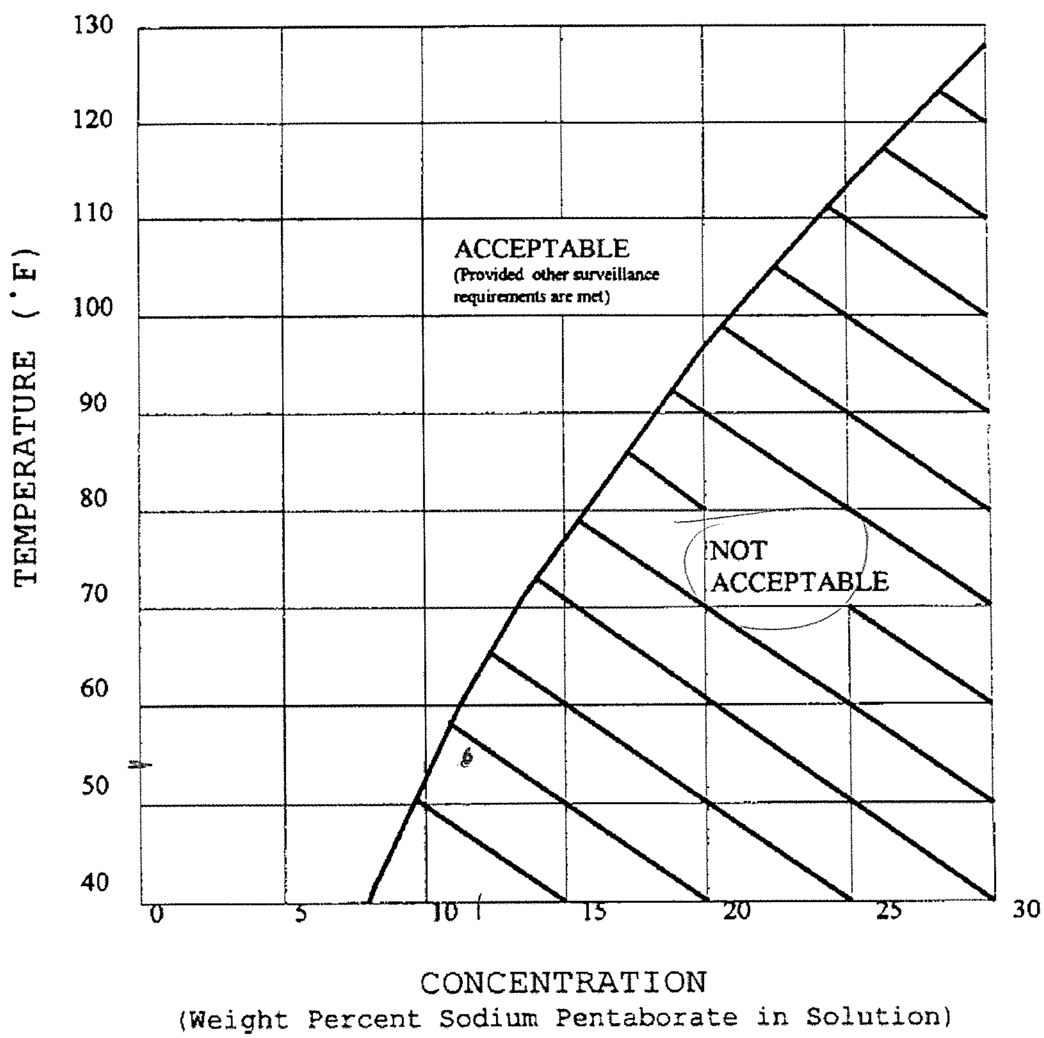


Figure 3.1.7-1
Sodium Pentaborate Solution Temperature Versus Concentration Requirements

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-1 (continued)

The Required Actions of Condition B are to be in MODE 3 within 12 hours AND in MODE 4 within 36 hours. A total of 12 hours is allowed for reaching MODE 3 and a total of 36 hours (not 48 hours) is allowed for reaching MODE 4 from the time that Condition B was entered. If MODE 3 is reached within 6 hours, the time allowed for reaching MODE 4 is the next 30 hours because the total time allowed for reaching MODE 4 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 4 is the next 36 hours.

EXAMPLE 1.3-2

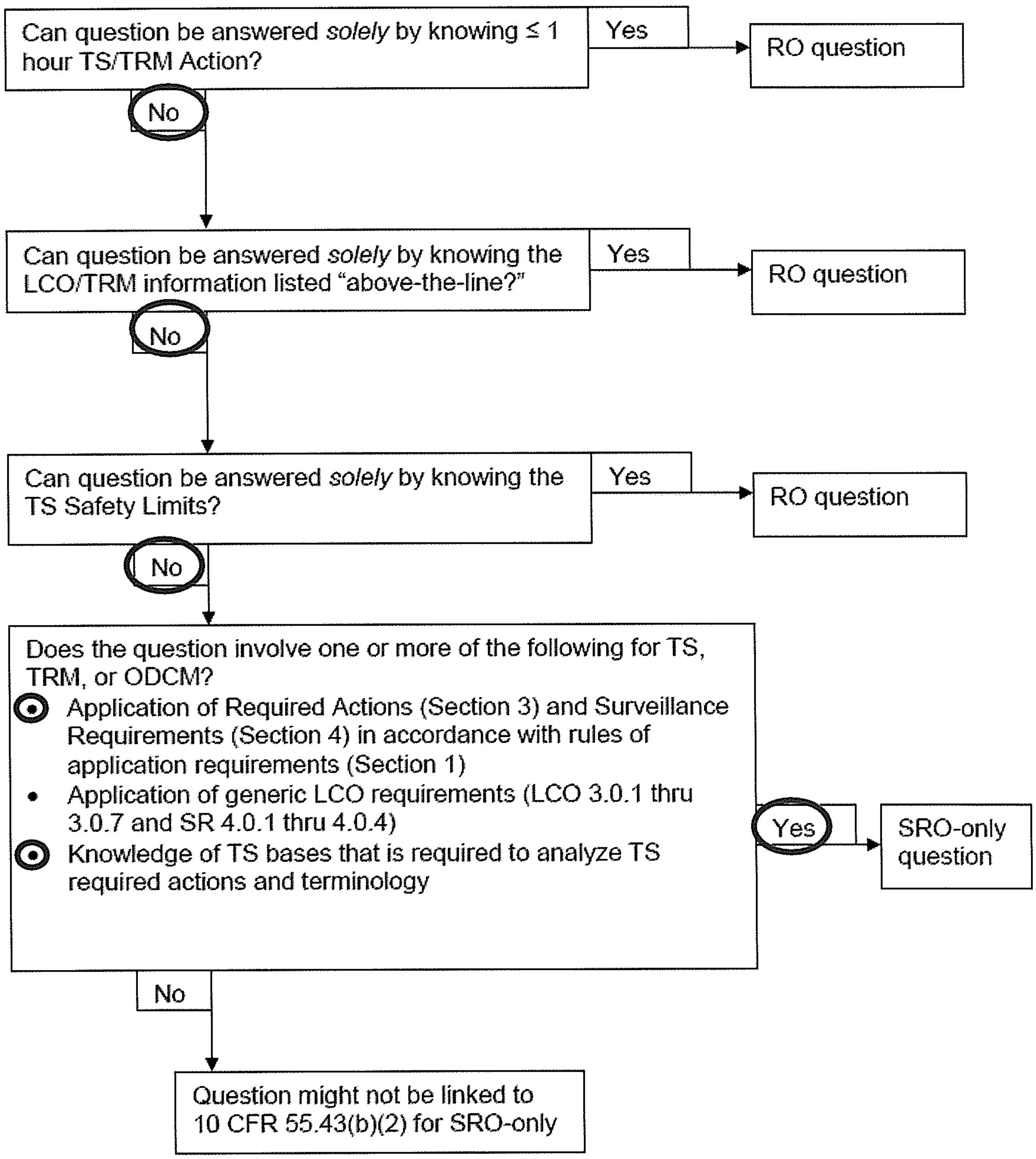
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

SRO Only Justification: The SRO will assess the annunciator 9-5B window #13, SLC TEMP ABNORMAL, and determine that both trains of Standby Liquid Control are Inoperable per Technical Specification 3.1.7.B. With knowledge of the application of Tech Spec Required Actions (Section 3), the SRO will determine that the Completion Time of 3.1.7.A is still available from the time the LCO was NOT met. This added to the 36 hour Completion Time in 3.1.7.C will give a total time allowed to reach MODE 4 of 44 hours. The SRO will need to know the Tech Spec 3.1.7 Bases relative to the current MODE 3 to determine that the reason for the TS Required Action is because standby liquid solution may not meet its design criteria to maintain the proper pH in the suppression pool following a LOCA.

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



Nile Mile 2 2010

Question: SRO #86

The reactor is shutdown following a scram from rated conditions. Reactor pressure is 940 psig and stable.

While completing post scram actions, annunciator 601711, SLCS TANK 1 TEMPERATURE HIGH/LOW, alarms. The field operator reports that local tank temperature is 69 degrees and that the breakers for both heaters are tripped.

Assuming that temperature continues to drop while troubleshooting the tripped breakers....

(1) How much time is available before reactor coolant temperature must be less than 200 degrees AND
(2) What is the reason for this action given these plant conditions?

A. (1) 32 hours
(2) The standby liquid solution may not meet its design criteria in responding to an ATWS.

B. (1) 44 hours
(2) The standby liquid solution may not meet its design criteria in responding to an ATWS.

C. (1) 32 hours
(2) The standby liquid solution may not meet its design criteria in maintaining the proper pH in the suppression pool following a LOCA.

D. (1) 44 hours
(2) The standby liquid solution may not meet its design criteria in maintaining the proper

pH in the suppression pool following a LOCA.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

QUESTION 88

Unit 1 is in MODE 2 withdrawing control rods for a startup.
The following conditions exist:

- All IRMs are on Range 1
- IRMs "G" and "H" are downscale
- All other IRMs are on scale and trending as expected
- SRM "D" is the highest reading SRM and has reached 1 x 10^5 cps

You are the Unit Supervisor, which ONE of the following actions is required per 1-GOI-100-1A, UNIT STARTUP?

A. Bypass SRM "D" and continue with the startup observing IRMs "G" and "H" for proper response.

B. Halt the startup and perform 1-SR-3.3.1.1, IRM Channel Calibration, on IRMs "G" and "H".

C. Declare IRMs "G" and "H" Inoperable and continue with the startup.

D. Bypass SRM "D" and perform 1-SR-3.3.1.1, IRM Channel Calibration, on IRMs "G" and "H".

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	215003 A2.01	
	Importance Rating		3.2

215003 Intermediate Range Monitor System A2.01 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to ~~correct, control, or~~ mitigate the consequences of those abnormal conditions or operations: Power supply degraded

Explanation: **C** CORRECT Per 1-GOI-100-1A, Unit Startup, and Tech Spec 3.3.1.1 Bases, the overlap between SRMs and IRMs exists when IRM downscale indications have cleared and IRM readings are on scale and trending higher prior to SRMs reaching 10^5 cps. IRM s G and H are downscale with SRM D is at $1X10^5$cps. IRMs G and H do not meet 1-SR-3.3.1.1.5, and therefore are Inoperable for Tech Spec 3.3.1.1, RPS Instrumentation, function 1a. However the LCO 3.3.1.1 is still met because there are still 3 required channels per trip system, therefore the startup CAN continue.

A- Incorrect – Bypassing SRM "D" is incorrect. There is nothing in the stem that would indicate that SRM "D" is not operating correctly. Plausible because the candidate may think that SRM "D" indicating 1×10^5 cps is the problem.

B- Incorrect. While IRMs G and H do not meet 1-SR-3.3.1.1.5, and therefore are Inoperable for Tech Spec 3.3.1.1, RPS Instrumentation, function 1a, the LCO 3.3.1.1 is still met because there are still 3 required channels per trip system, therefore there is no reason to stop and calibrate the IRMs. Doing so at this low power adds additional challenge and could result in a full reactor scram.

D- Incorrect - Bypassing SRM "D" is incorrect. There is nothing in the stem that would indicate that SRM "D" is not operating correctly. Plausible because the candidate may think that SRM "D" indicating 1×10^5 cps is the problem. While IRMs G and H do not meet 1-SR-3.3.1.1.5, and therefore are Inoperable for Tech Spec 3.3.1.1, RPS Instrumentation, function 1a, the LCO 3.3.1.1 is still met because there are still 3 required channels per trip system, therefore there is no reason to stop and calibrate the IRMs. Doing so at this low power adds additional challenge and could result in a full reactor scram.

Technical Reference(s): Unit 1 TS 3.3.1.1 and bases

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New:
Question History:	Previous NRC: Brunswick 2007 NRC SRO #88
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X

10 CFR Part 55 Content: 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

[16] **VERIFY** Reactor Engineer records applicable criticality data in 1-SR-3.1.1.1, Reactivity Margin Test.

_______ Initials _______ Date _______ Time

[17] **VERIFY** Reactor period greater than 60 seconds.

(R) _______ Initials _______ Date _______ Time

> **NOTE**
>
> 1) Completing paper closure of 1-SR-3.3.1.1.5, SRM and IRM Overlap Verification, is not required prior to performing Step 5.4[18]. However, all AC steps must be VERIFIED COMPLETED SATISFACTORILY prior to withdrawing SRMs.
>
> 2) Tech Spec Bases state that overlap between SRMs and IRMs exists when IRM downscale indications have cleared and IRM readings are on scale and trending higher prior to SRMs reaching 10^5 cps.

[18] **VERIFY** SRM/IRM overlap by obtaining data and completing 1-SR-3.3.1.1.5 SRM and IRM Overlap Verification.

(R) _______ Initials _______ Date _______ Time

Reactor Engineer

5.4 Withdrawal of Control Rods while in Mode 2 (continued)

NOTES

1) SRMs are fully withdrawn when IRMs are on Range 3 or above and indicating above their downscale trip point.
2) If a shutdown margin test has been performed using a different rod sequence, 1-SR-3.1.3.5(A), Control Rod Coupling Integrity Check, will provide required actions to insert all control rods, establish normal sequence and perform the subsequent start up with re-entry at Step 5.3[13].

[19] **WITHDRAW** SRMs as necessary to maintain them on scale between 10^2 cps and 10^5 cps.

_____ Initials _____ Date _____ Time

[20] **MAINTAIN** IRMs on scale between approximately 25 and 75 using IRM range switches.

_____ Initials _____ Date _____ Time

[21] **ENSURE** 1-SI-4.6.B.1-4, Reactor Coolant Chemistry, has been satisfactorily completed prior to pressurizing Reactor.

(R) _____ Initials _____ Date Chem Shift Supv _____ Time

[22] **WHEN** all operable IRMs are on Range 3 or above and all acceptance criteria is met for 1-SR-3.3.1.1.5, SRM and IRM Overlap Verification, **THEN**

WITHDRAW all operable SRMs.

(R) _____ Initials _____ Date _____ Time

Date ________

7.0 PROCEDURE STEPS (continued)

NOTES

The following table is configured in the same manner as the actual layout of IRM channels while facing Panel 1-9-12.

It is permissible to complete IRM channel status table entries in the following step at a later time if one (1) IRM channel in either or both RPS trip systems is slow clearing its downscale indication.

The following step may be completed when at least three (3) IRM channels per RPS trip system have cleared their downscale indications and are trending higher with all operable SRM channels indicating less than 10^5 cps.

[8] **OBSERVE** SRM/IRM channel indications during startup and **MARK** table below with a checkmark to indicate which IRM channels meet **ALL** three of the following:

- cleared downscale indication (1-9-5 & 1-9-12 panel IRM downscale lights are not lit),
- indicate ≥7.5%, and
- are trending higher

with all operable SRM channels indicating less than 10^5 cps. _____(AC)

RPS TRIP SYSTEM A

IRM	SAT	UNSAT		IRM	SAT	UNSAT
A				C		
E				G		

RPS TRIP SYSTEM B

IRM	SAT	UNSAT		IRM	SAT	UNSAT
B				D		
F				H		

BASES

SURVEILLANCE REQUIREMENTS (continued)

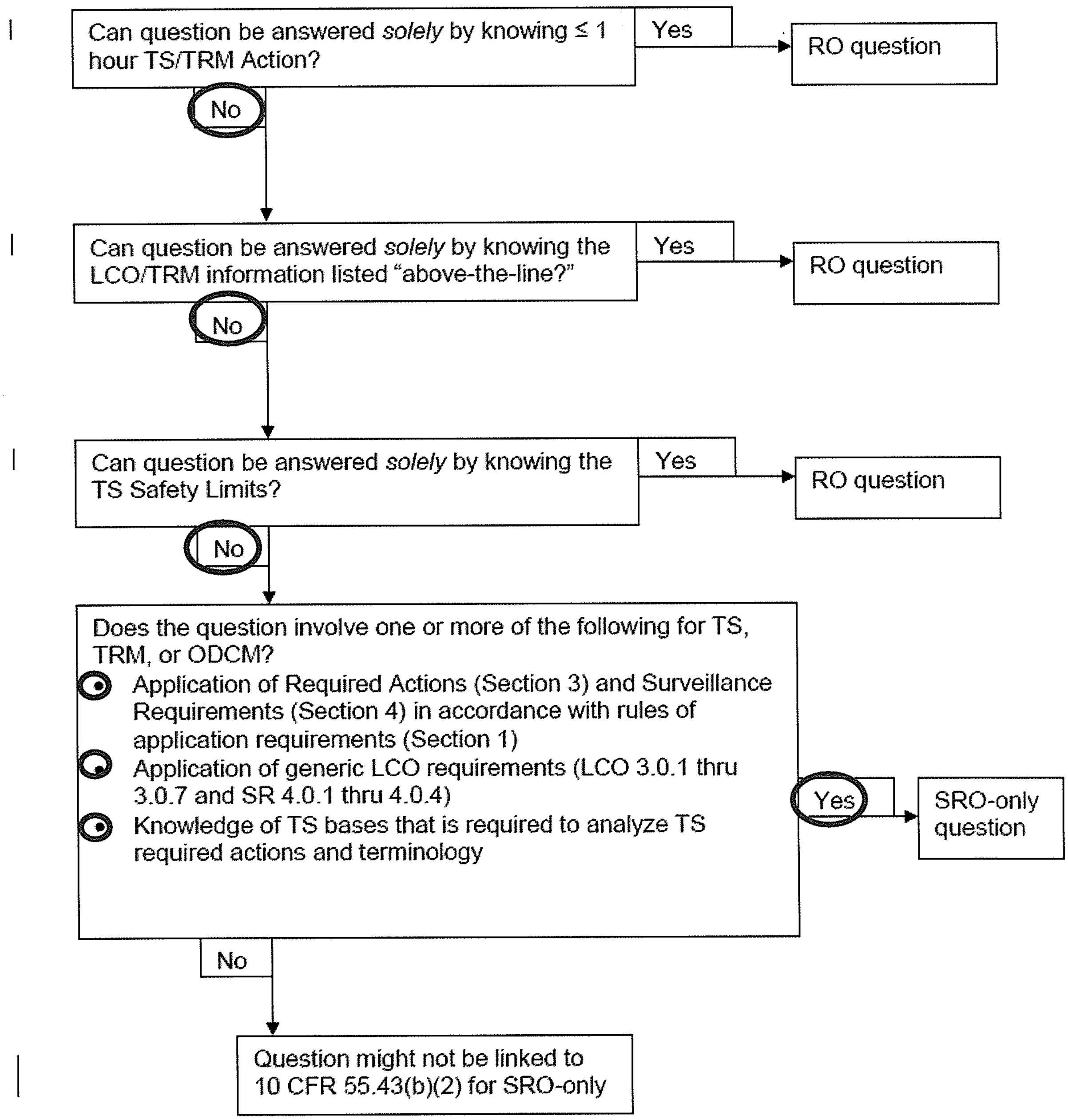
SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a neutron flux region without adequate indication. This is required prior to withdrawing SRMs from the fully inserted position since indication is being transitioned from the SRMs to the IRMs. Overlap between SRMs and IRMs exists when IRM downscale indications have cleared and IRM readings are onscale and trending higher prior to SRMs reaching 10^5 cps.

SRO Only Justification: The SRO will assess the plant conditions to determine that with IRMs G and H downscale while SRM D is at $1X10^5$cps, IRMs G and H do not meet 1-SR-3.3.1.1.5, and therefore are Inoperable for Tech Spec 3.3.1.1, RPS Instrumentation, function 1a. The SRO will also need to understand the application of generic LCO requirements and recognize that LCO 3.3.1.1 is still met because there are still 3 required channels per trip system, therefore the startup can continue.

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



3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each channel.

RPS Instrumentation
3.3.1.1

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

Brunswick 2007 #88

Proposed Question: Unit One (1) is in Mode 2 withdrawing control rods during a startup. All IRMs are operable and on Range 1.

IRMs "C" and "F" are downscale.

All other IRMs are responding as expected. ??

SRM "D" is rising and has reached 5 x 10^5 cps.

You are the SCO, which ONE of the following actions is required?

a. In accordance with GP-02 notify the reactor engineer and address T.S. 3.3.1.1.

b. Bypass SRM "D" and continue the startup observing IRMs "C" and "F" for proper response.

c. Halt the startup and perform 1OP-09, Section 8.6 SRM/IRM Response Check on IRMs "C" and "F".

d. Bypass SRM "D" continue the startup and perform 1OP-09, Section 8.6 SRM/IRM Response Check on IRMs "C" and "F".

A
C
E
G

B
D
F
H

QUESTION 89

Unit 3 is operating at 100% power when a loss of 250V RMOV Board 3B occurs.

Which ONE of the following identifies the required action statement(s), if any, in accordance with Technical Specification 3.5.1, ECCS - Operating?

[REFERENCE PROVIDED]

A. Action statements G.1 and G.2 are required to be entered

B. ONLY action statement E.1 is required to be entered

C. No action statement in LCO 3.5.1 is required to be entered for the supported systems

D. Action statement H.1 is required to be entered

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	218000 A2.05	
	Importance Rating		3.6

218000 Automatic Depressurization System
Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Loss of A.C. or D.C. power to ADS valves.

Explanation: **C** CORRECT –Per LCO 3.0.6, when a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered.

A Incorrect – Plausible because this would be the correct answer if LCO 3.0.6 didn't apply. Two ADS valves (1-18 and 1-19) are powered from 250V RMOV Bd 3B ONLY. Therefore Required actions G.1 and G.2 would be applicable.

B Incorrect – Plausible if the is unaware of LCO 3.0.6 and decides to enter LCO 3.5.1 he/she may believe that the power supplies are divided up so that no one power supply will cause a loss of power to <u>more than one</u> valve.

D Incorrect –Plausible if the candidate is unaware of LCO 3.0.6 and decides to enter LCO 3.5.1 he/she may know two ADS valves (1-18 and 1-19) are powered from 250V RMOV Bd 3B ONLY. Additionally, HPCI is a Division II system and the candidate may believe that 250V RMOV 3B powers Div II. This would lead them to two ADS valves and HPCI Inoperable.

Technical Reference(s): 3-AOI-1-1, Tech Spec 3.5.1

Proposed references to be provided to applicants during examination: Tech Spec 3.5.1, 3.0.6

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X
Question History:	Previous NRC: None	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis :	X

10 CFR Part 55 Content: 55.43 (b) (2) Facility operating limitations in the technical specifications and their bases.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an evaluation shall be performed in accordance with Specification 5.5.11, "Safety Function Determination Program (SFDP)." If a loss of safety function is

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.6 (continued)

determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure $\leq$ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. <u>OR</u> One low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Two or more ADS valves inoperable. <u>OR</u> Required Action and associated Completion Time of Condition C, D, E, or F not met.	G.1 Be in MODE 3. <u>AND</u> G.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
H. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. <u>OR</u> HPCI System and one or more ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately
E. One ADS valve inoperable.	E.1 Restore ADS valve to OPERABLE status.	14 days
F. One ADS valve inoperable. <u>AND</u> Condition A entered.	F.1 Restore ADS valve to OPERABLE status. <u>OR</u> F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours

(continued)

Attachment 1
(Page 1 of 3)

Unit 3 SRV Solenoid Power Breaker/Fuse Table, Panel 25 32

Unit 3 SRV Solenoid Power Breaker Table							
SERV		NORMAL PWR 250V RMOV	BRKR	ALTERNATE 250V RMOV	BRKR	ALTERNATE BATT. BRD	BRKR
1-4		3A	11B2				
1-5	*	3C	7A			2	710
1-18	*	3B	1C2	**			
1-19	*	3B	1B2	**			
1-22	*	3A	11C2	3B	1C1		
1-23		3C	1B1				
1-30		3A	11C1				
1-31		3B	8C1	**			
1-34	*	3C	10A			2	710
1-41	*	3A	9B1	3C	8A	2	710
1-42		3B	8B2				
1-179		3B	8C2	**			
1-180		3A	1D1				

* ADS Valves.

** All components have been removed from 3-PNL-25-32

SRO Only Justification: The SRO will assess plant conditions to determine that 2 ADS valves are Inoperable per Technical Specification 3.5.1.G. The SRO will need to understand the application of generic LCO requirements of LCO 3.0.6 for Support/ Supported systems. LCO 3.0.6 allows entering the Conditions and Required Actions of the SUPPORT system without entering the SUPPORTED system, in this case LCO 3.5.1.

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

Can question be answered *solely* by knowing ≤ 1 hour TS/TRM Action?

Yes → RO question

No

Can question be answered *solely* by knowing the LCO/TRM information listed "above-the-line?"

Yes → RO question

No

Can question be answered *solely* by knowing the TS Safety Limits?

Yes → RO question

No

Does the question involve one or more of the following for TS, TRM, or ODCM?

- Application of Required Actions (Section 3) and Surveillance Requirements (Section 4) in accordance with rules of application requirements (Section 1)
- Application of generic LCO requirements (LCO 3.0.1 thru 3.0.7 and SR 4.0.1 thru 4.0.4)
- Knowledge of TS bases that is required to analyze TS required actions and terminology

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(2) for SRO-only

QUESTION 90

During a Unit 3 outage, work was performed on Main Steam Relief Valve 3-PCV-1-19.

The work order for 3-PCV-1-19 (ADS valve) requires a Unit Operator to perform a partial surveillance, 3-SR-3.4.3.2, Main Steam Relief Valves Manual Cycle Test.

Which ONE of the following completes the statements?

This test method ___(1)___ an acceptable Post Maintenance Test (PMT) per NPG-SPP-06.3, Pre-/Post-Maintenance Testing.

During tests involving the manipulation of plant equipment, NPG-SPP-06.9.1, Conduct of Testing, requires Operations to be notified ___(2)____.

A. (1) is
(2) ANY time there is a change of test directors

B. (1) is NOT
(2) ANY time there is a change of test directors

C. (1) is
(2) when the test is suspended for more than one shift

D. (1) is NOT
(2) when the test is suspended for more than one shift

Answer: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		1
	K/A#	239002 G2.2.12	
	Importance Rating		4.1

239002 Relief/Safety Valves
2.2.12 Knowledge of surveillance procedures

Explanation: **C** CORRECT – First Part: Per NPG-SPP-06.3, Pre/Post-Maintenance Testing, the use of a partial surveillance IS an acceptable method of post maintenance testing. Second Part: Correct: NPG-SPP-06.9.1, Conduct of Testing, requires Operations to be notified, when the test is suspended for more than one shift.

A Incorrect –First Part: Correct. Second Part: Incorrect: Operations Notification is NOT required when changing test directors except for those tests being conducted by Operations personnel located in the Main Control Room.

B Incorrect- First Part: Incorrect. Per NPG-SPP-06.3, Pre/Post-Maintenance Testing, the use of a partial surveillance IS an acceptable method of post maintenance testing. Second Part: Incorrect: Operations Notification is NOT required when changing test directors except for those tests being conducted by Operations personnel located in the Main Control Room.

D Incorrect - First Part: Incorrect. Per NPG-SPP-06.3, Pre/Post-Maintenance Testing, the use of a partial surveillance IS an acceptable method of post maintenance testing. Correct: NPG-SPP-06.9.1, Conduct of Testing, requires Operations to be notified, when the test is suspended for more than one shift.

Technical Reference(s): NPG-SPP-06.3, NPG-SPP-06.9.1

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New	X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 2 Facility operating limitations in the technical specifications and their bases.

NPG-SPP-6.3 Pre/Post Maintenance Testing

3.2.3 PMT Instructions

NOTE
Surveillance instructions (SI), as specified in this procedure, included other instructions which implement site TS requirements such as surveillance requirements (SR), offsite dose instructions (ODIs), etc.

A. General Requirements

1. An SI may be used as PMT. While an SI may be required for TS operability, supplemental PMT may be required in order to test all components or features either directly or potentially affected by the activity.

2. PMT instructions in a WO shall be specified as follows:

   a. Pre-approved plant instructions.

   b. Stand-alone portions of pre-approved plant instructions.

   c. Specific steps in the body of a WO which satisfy PMT requirements.

   d. Specific steps for inspections, visual examinations, data comparisons, or other straight-forward checks.

   e. A formal instruction prepared specifically for the WO PMT.

3.2.5 Operations Notification

Operations shall be notified, during test performance involving the manipulation of plant equipment, of the following items:

A. When starting/stopping the test.

B. When changing test directors except for those tests being conducted by Operations personnel located in the Main Control Room.

C. When suspending the test for more than one shift.

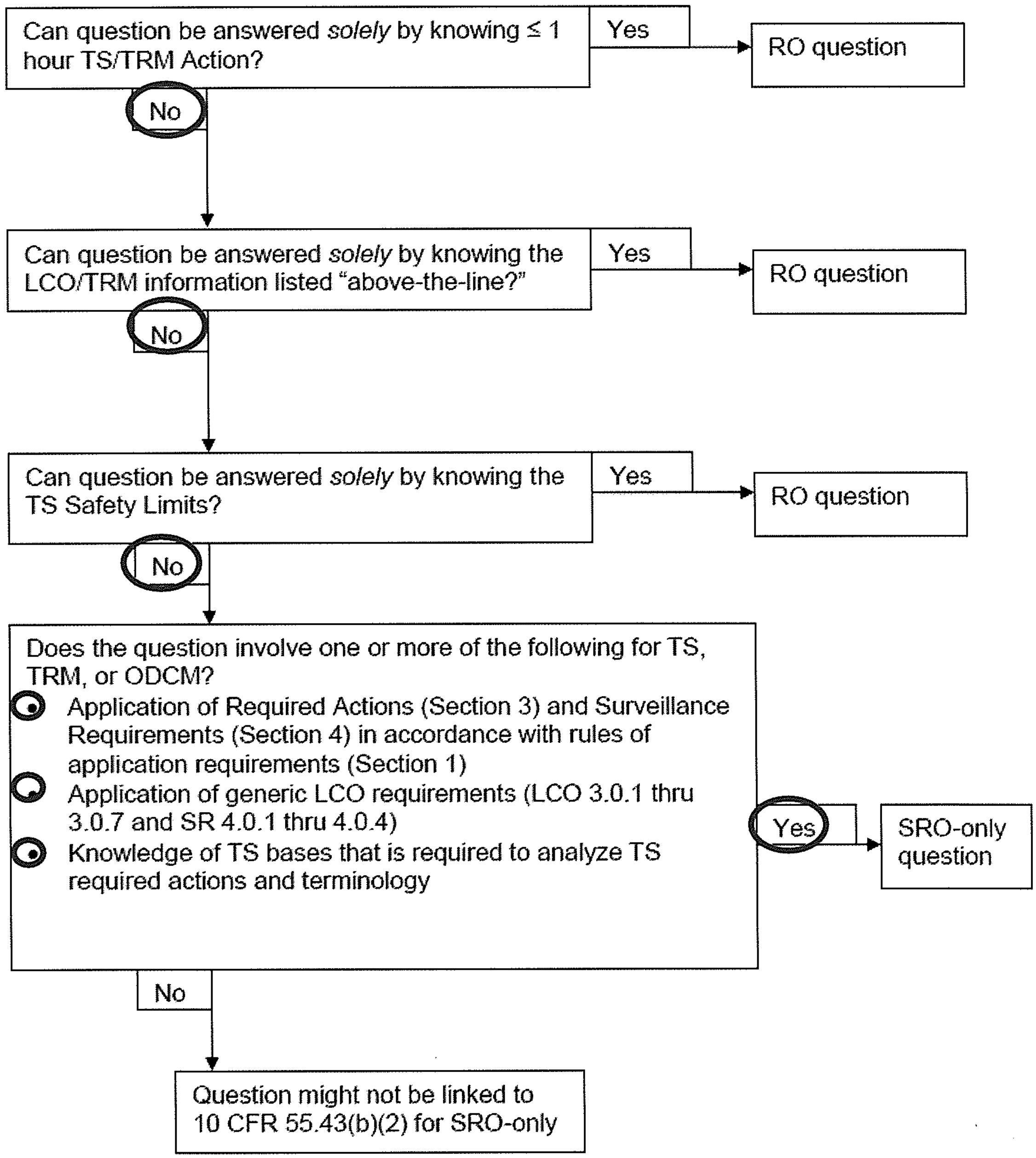
D. When continuing a test past an Operations shift change - timely notification of the oncoming Operations personnel is required except for those tests being conducted by Operations personnel located in the Main Control Room.

E. When Technical Specification (TS) deficiencies and/or difficulties occur.

F. When changing the expected duration of temporary conditions.

SRO Only Justification: With knowledge of the Pre/Post-Maintenance Testing and the surveillance process, the SRO will recognize a partial surveillance IS an acceptable method of post maintenance testing. From the surveillance test results the SRO will determine that the valve is NOT Operable for Technical Specification 3.5.1, ECCS - Operating. 3.5.1 which requires the valve to cycle on a signal (relief function) either from the switch or the logic. However, the SRO will determine that the valve REMAINS Operable for Technical Specification 3.4.3, Safety Relief valves (S/RVs).

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



QUESTION 91

An ATWS has occurred on Unit 3. The Unit Supervisor is operating in accordance with 3-C-5, LEVEL AND POWER CONTROL.

The following conditions currently exist:

- RPV water level band is -50 to -100 inches and steady using Feedwater per 3-EOI Appendix -5A, INJECTION SYSTEMS LINEUP-CONDENSATE/FEEDWATER
- RPV Pressure is 900 psig and slowly lowering
- No Boron has been injected
- The Unit Operator ATC is inserting control rods per 3-EOI Appendix-1D, INSERT CONTROL RODS USING REACTOR MANUAL CONTROL SYSTEM
- 17 Control Rods are at position 02, all other rods are fully inserted.

Which ONE of the following describes the actions the Unit Supervisor is required to take, and the basis for that action?

a. Exit 3-EOI-C-5, LEVEL AND POWER CONTROL, enter 3-EOI-1 RPV CONTROL, at step RC/L-1, and direct RPV Water Level restored to +2 to +51 inches.
The reactor will remain subcritical under all conditions.

b. Remain in 3-EOI-C-5, LEVEL AND POWER CONTROL until all rods are inserted beyond position 02, and maintain current RPV water level band.
The reactor will **NOT** remain subcritical under all conditions.

c. Exit 3-EOI-C-5, LEVEL AND POWER CONTROL, enter 3-EOI-1 RPV CONTROL, at step RC/L-1, and maintain current RPV water level band.
The reactor will **NOT** remain subcritical under all conditions.

d. Remain in 3-EOI-C-5, LEVEL AND POWER CONTROL until all rods are inserted beyond position 02, and direct RPV Water Level restored to +2 to +51 inches.
The reactor will remain subcritical under all conditions.

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	201003 G2.4.18	
	Importance Rating		4.0

2001003 Control Rod and Drive Mechanism. Generic 2.4.18 Knowledge of the specific bases for EOPs.

Explanation: **A** CORRECT – The reactor will remain subcritical under all conditions with control rods in the configuration stated in the stem. The correct action for the Unit Supervisor in this case would be to exit 3-EOI-C-5, LEVEL AND POWER CONTROL, enter 3-EOI-1 RPV CONTROL, at step RC/L-1. This is stated in the retainment override C5-1. Once in 3-EOI-1 RPV CONTROL, the RC/L leg will direct RPV Water Level restored to +2 to +51 inches.

B- Because reactor will remain subcritical under all conditions, it would be incorrect for the Unit Supervisor to remain in 3-EOI-C-5, LEVEL AND POWER CONTROL. However, this is plausible because it would be the correct action for Unit 1. Unit 1 requires all control rods to be inserted to or beyond position 02 to be subcritical under all conditions.

C- The action is correct however the bases is not correct. The reactor will remain subcritical under all conditions with control rods in the configuration stated in the stem. This is plausible because the control rod configuration required to remain shutdown under all conditions is different for different units.

D- Because reactor will remain subcritical under all conditions, it would be incorrect for the Unit Supervisor to remain in 3-EOI-C-5, LEVEL AND POWER CONTROL. However, this is plausible because it would be the correct action for Unit 1. Unit 1 requires all control rods to be inserted to or beyond position 02 to be subcritical under all conditions. The water level will not be restored to +2 to +51 inches in 3-EOI-C-5, LEVEL AND POWER CONTROL.

Technical Reference(s): 3-EOI-C-5, EOIPM 0-V-K

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: New: X	
Question History:	Previous NRC: NO	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis : X	
10 CFR Part 55 Content:	55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.	

WHILE EXECUTING THIS PROCEDURE:	
IF	**THEN**
RPV water lvl CANNOT be determined	EXIT this procedure and ENTER C4, RPV Flooding
1 The reactor will remain subcritical without boron under all conditions	EXIT this procedure and ENTER EOI-1, RPV Control at Step RC/L-1
PC water lvl CANNOT be maintained below 105 ft OR Suppr chmbr press CANNOT be maintained below 55 psig	STOP inj into the RPV from sources external to the PC NOT required for adequate core cooling or shutting down the reactor
DW Control Air becomes unavailable	CROSSTIE CAD to DW Control Air (APPX 8G)

C5-1

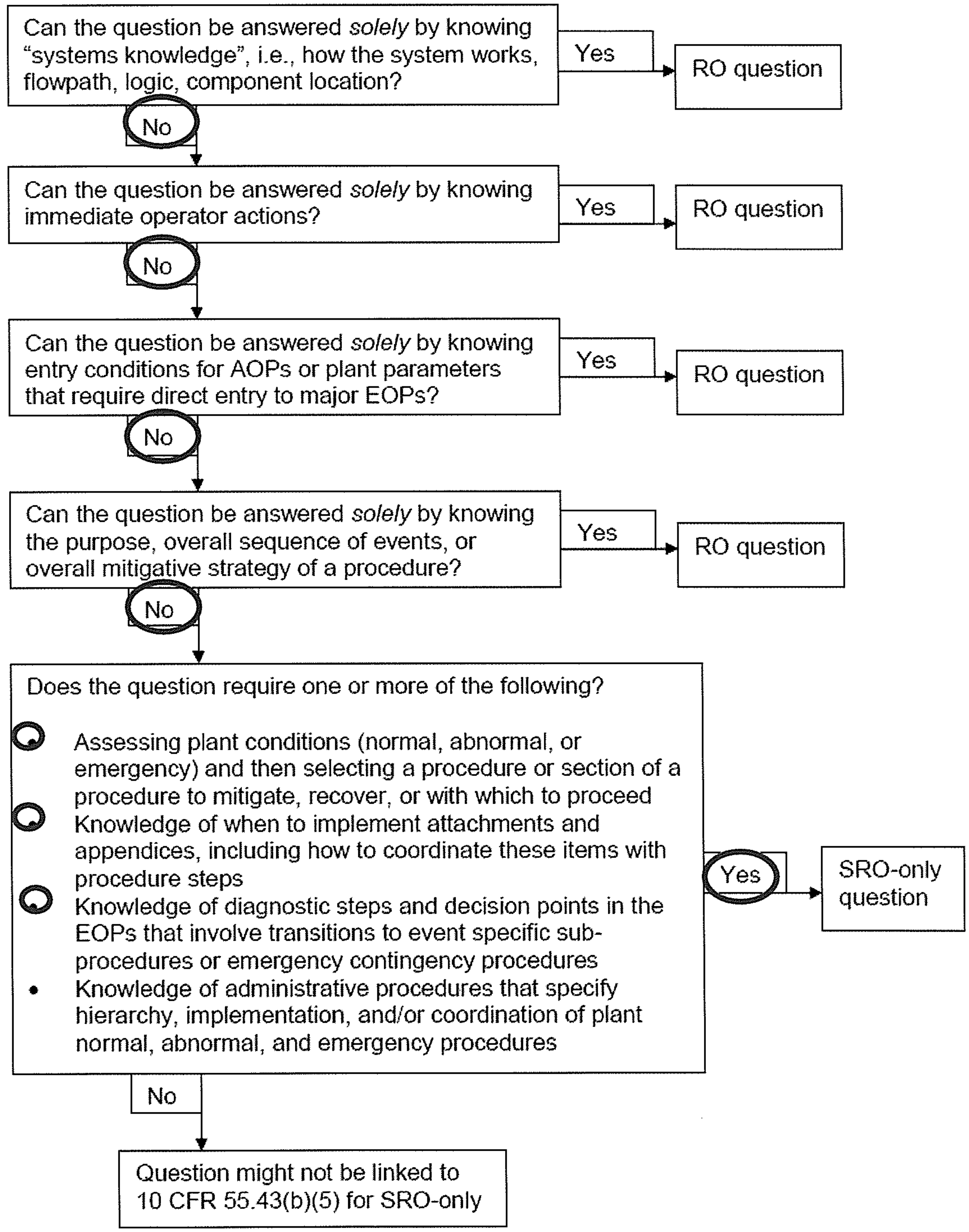
NOTE	
1	The reactor will remain subcritical without boron under all conditions when: • Any 19 control rods are at position 02 with all other control rods fully inserted OR • All control rods except one are inserted to or beyond position 00 OR • Determined by Reactor Engineering

EOIPM 0-V-K Flowchart C-5, LEVEL AND POWER CONTROL Bases

…retainment overide statement directs the operator to transfer RPV water level control actions if it has been determined that the reactor will remain subcritical under all conditions. Engineering calculations have determined that when all control rods are inserted to or beyond position (Maximum Subcritical Banked Withdrawal Position), the reactor will remain subcritical under all conditions.

SRO Only Justification: The SRO will assess the plant conditions and with knowledge of the determine the Notes and Tables in 3-EOI-C-5, LEVEL AND POWER CONTROL, recocognize the requirement to transition from 3-EOI-C-5 to 3-EOI-1 RPV CONTROL, at step RC/L-1, and direct RPV Water Level restored to +2 to +51 inches since 17 Control Rods at position 02, all other rods fully inserted meets the definition of the reactor shutdown under all conditions without boron.

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



QUESTION 92

Unit 3 is operating at 100% power when the following occurs:

- Main steam line flow increases rapidly
- All Main Steam Line Isolation Valves close, except MSIV LINE A INBOARD, 3-FCV-1-14A, which indicates intermediate.
- Main Steam Line Leak Detection High (Panel 9-3D-Window 24) is in alarm and TIS-1-60A reached 240°F and is slowly lowering

After conditions have stabilized, which ONE of the following completes the statements?

The Unit Supervisor shall direct ___(1)____.

The Emergency Plan classification for this event is____(2)_____.

[REFERENCE PROVIDED]

A. (1) Main Steam Line A isolated within 8 hours by verifying CLOSED and de-activating Main Steam Line A Outboard Valve, 3-FCV-1-15
(2) 4.2-U

B. (1) Main Steam Line A isolated within 8 hours by verifying CLOSED and de-activating Main Steam Line A Outboard Valve, 3-FCV-1-15
(2) 3.1-S

C. (1) Main Steam Line A isolated within 4 hours by verifying CLOSED and de-activating Main Steam Line A Outboard Valve, 3-FCV-1-15
(2) 4.2-U

D. (1) Main Steam Line A isolated within 4 hours by verifying CLOSED and de-activating Main Steam Line A Outboard Valve, 3-FCV-1-15
(2) 3.1-S

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	239001 A2.11	
	Importance Rating		4.3

239001 Main and Reheat Steam System
Ability to (a) predict the impacts of the following on the MAIN AND REHEAT STEAM SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.11 Steam line break

Explanation: A CORRECT: MSIV LINE A INBOARD 3-FCV-1-14 is Inoperable for Technical Specification 3.6.1.3, PCIVs. LCO Condition A and Required Action A1 apply. This will direct closing and verifying deactivated the Main Steam Line A Outboard Valve, 3-FCV-1-15. The Emergency classification for this event is 4.2-S.

B Incorrect –First Part: Correct. Second Part: Incorrect. While there is a primary system discharging to Secondary Containment, the area temperature given is below the Maximum Safe Operating Temperature of 315° F. Plausible if the candidate mistakes the Max Safe Temperature value. 240°F is the Maximum Safe Operating Temperature for other areas of Secondary Containment.

C Incorrect – First Part: Incorrect. Plausible since the 4 hour LCO Required Completion time is for other Primary Containment Penetrations (not Main Steam). Second Part: Correct.

D Incorrect – First Part: Incorrect. Plausible since the 4 hour LCO Required Completion time is for other Primary Containment Penetrations (not Main Steam). Second Part: Incorrect. While there is a primary system discharging to Secondary Containment, the area temperature given is below the Maximum Safe Operating Temperature of 315° F. Plausible if the candidate mistakes the Max Safe Temperature value. 240°F is the Maximum Safe Operating Temperature for other areas of Secondary Containment.

Technical Reference(s): 3-EOI-1, EPIP-1,Unit 3 TS 3.6.1.3

Proposed references to be provided to applicants during examination: EPIP-1,Unit 3 TS 3.6.1.3

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

--NOTES--

1. Penetration flow paths except for 18 and 20 inch purge valve penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria in MODES 1, 2, and 3.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. --------------NOTE------------ Only applicable to penetration flow paths with two PCIVs. ---------------------------------- One or more penetration flow paths with one PCIV inoperable except due to MSIV leakage not within limits.	A.1	Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.	4 hours except for main steam line <u>AND</u> 8 hours for main steam line
	<u>AND</u>		

(continued)

MAIN STEAM LINE BREAK	LIQUID EFFLUENT	
Description	Description	
4.2-U	4.3-U	
Main Steam Line break outside Primary Containment with isolation. OPERATING CONDITION: Mode 1 or 2 or 3	Liquid release rate exceeds 20 times ECL as determined by chemistry sample AND Release duration exceeds or will exceed 60 minutes OPERATING CONDITION: ALL	UNUSUAL EVENT
	4.3-A	
	Liquid release rate exceeds 2000 times ECL as determined by chemistry sample AND Release duration exceeds or will exceed 15 minutes. OPERATING CONDITION: ALL	ALERT
4.2-S		
Unisolable Main Steam Line break outside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3		SITE EMERGENCY
		GENERAL EMERGENCY

SECONDARY CONTAINMENT TEMPERATURE

Description

UNUSUAL EVENT

ALERT

SITE EMERGENCY

3.1-S			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1.

OPERATING CONDITION:
Mode 1 or 2 or 3

GENERAL EMERGENCY

3.1-G			TABLE	US

An unisolable Primary System leak is discharging into Secondary Containment

AND

Any area temperature exceeds the Maximum Safe Operating Temperature limit listed in Table 3.1

AND

Any indication of potential or significant fuel cladding failure exists. Refer to Table 3.1-G/3.2-G with RCS Barrier intact inside Primary Containment

OPERATING CONDITION
Mode 1 or 2 or 3

NOTES

CURVES/TABLES:

TABLE 3.1
MAXIMUM SAFE OPERATING AREA TEMPERATURE LIMITS

AREA	APPLICABLE PANEL 9-21 TEMPERATURE ELEMENTS (UNLESS OTHERWISE NOTED)	MAX SAFE OPERATING VALUE °F		
		UNIT 1	UNIT 2	UNIT 3
RHR A/C Pump Room	74-95A	215	150	155
RHR B/D Pump Room	74-95B	150	210	215
HPCI Turbine Area	73-55A	275	270	270
CS A/C Pump and RCIC Turbine Area	71-41A	190	190	190
RCIC Steam Supply Area	71-41B, 41C, 41D	195	200	250
HPCI Steam Supply Area	73-55B, 55C, 55D	245	240	240
RHR A/C Pump Supply Area	74-95H	245	240	240
RHR B/D Pump Supply Area	74-95G	190	240	240
Main Steam Line Leak Detection High	(XA-55-3D-24) Panel 9-3 TIS-1-60A	315	315	315
RHR Valve Room	74-95E	175	170	175
RWCU Isol Logic Channel A/B Temp High	(XA-55-5B-32/33) Panel 9-5 69-835A, B, C, D Aux Inst Room	175	170	175
RWCU Outbd Isol Vlv Area	69-29F	220	220	220
RWCU Hx Area	69-29G	220	220	220
RWCU Hx Exh Duct	69-29H	220	220	220
RWCU Recirc Pump A Area	69-29D	215	215	215
RWCU Recirc Pump B Area	69-29E	215	215	215
RHR A/C Hx Room	74-95C	210	195	200
RHR B/D Hx Room	74-95D	210	195	200
FPC Hx Area	74-95F	160	155	155

MAIN STEAM LINE LEAK DETECTION TEMP HIGH 1-TA-1-60 (24)	Sensor/Trip Point:

(Page 1 of 2)

Sensor/Trip Point:

1-TE-1-60A (1-TIS-1-60A)	170°F
1-TE-1-60B through D (1-TR-69-29)	160°F

Sensor Location:

1-TE-1-60A	Main Steam Tunnel
1-TE-1-60B	Main Steam Line Steam Vault Area Elev 565
1-TE-1-60C	Main Steam Line Bypass Vlv Area Elev 586
1-TE-1-60D	Main Steam Line Control Vlv Area Elev 617

Probable Cause:

A. Main Steam, RWCU, Feedwater, RCIC, or HPCI discharge (only with HPCI in service and elevated Suppression Pool water Temp.) line break.
B. TB or RB Coolers out of service
C. Sensor malfunction
D. Steam Vault Exhaust Booster Fan out of service.

Automatic Action: Impending MSIV Isolation at 189°F area temp.

Operator Action:

NOTE

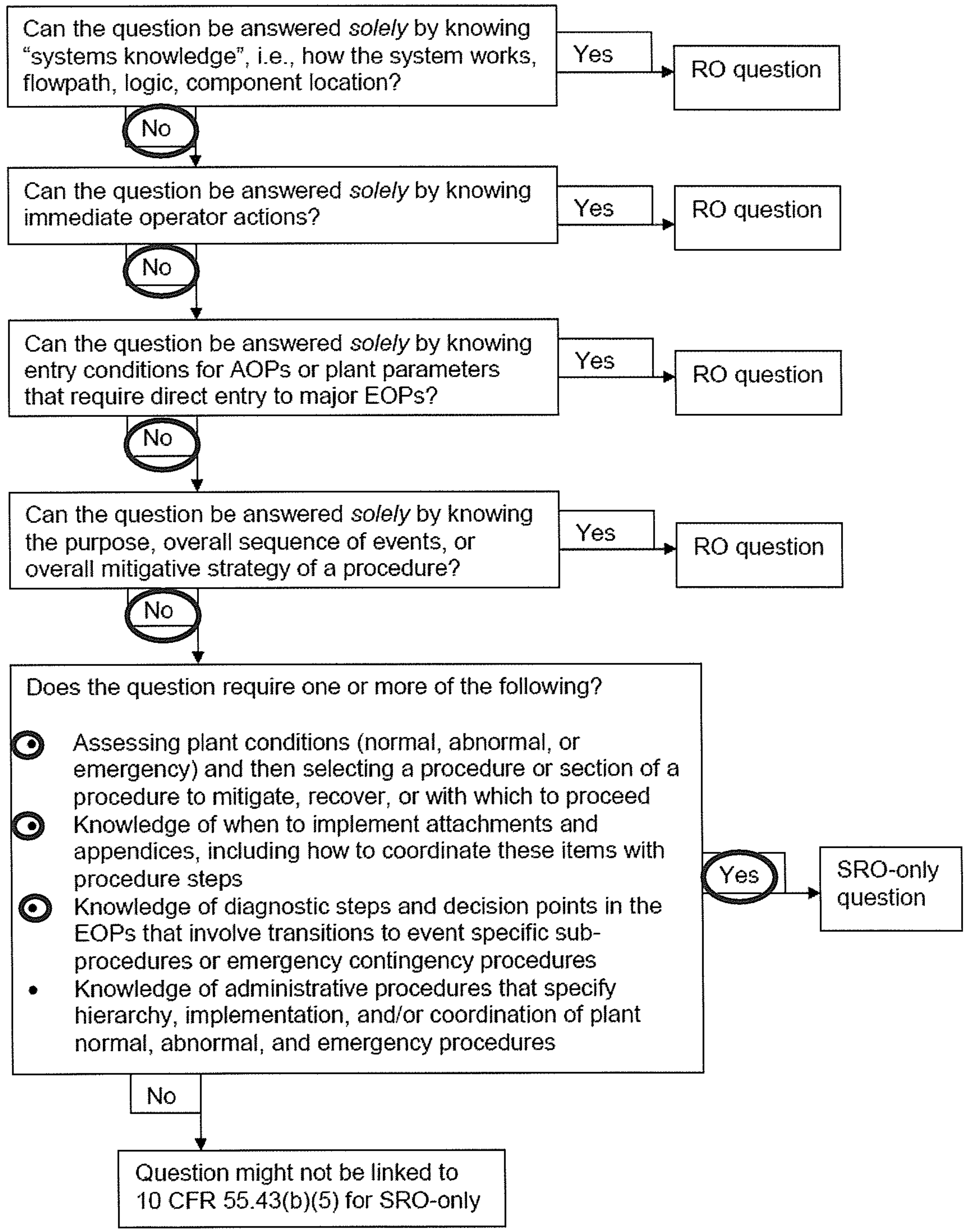
The following steps may be performed in any order or concurrently, as necessary.

A. **CHECK** the following temperature indicators:
- Main Steam temperature elements on LEAK DETECTION SYSTEM TEMPERATURE Recorder, 1-TR-69-29 (Points 13-15) on Panel 1-9-21, ☐
- MN STEAM TUNNEL TEMP, 1-TIS-1-60A on 1-9-3, ☐
- RWCU Piping in the Main Steam Tunnel temperature indicators, 1-TIS-69-834A-D, Aux Inst Room 1-PNLA-009-0083(84)(85)(86) or "HPTURB" mimic on ICS. ☐

B. **CHECK** the following flow indications:
- MAIN STEAM LINE FLOW A(B, C, D), 1-FI-46-1(2, 3, 4) on Panel 1-9-5. ☐
- RFW FLOW LINE A(B) 1-FI-3-78A(B) on Panel 1-9-5. ☐
- RFP 1A(B, C) flow indicators, 1-FI-3-20(13, 6) on Panel 1-9-6. ☐

SRO Only Justification: The SRO will assess plant conditions and recognize that MSIV LINE A INBOARD 3-FCV-1-14 is Inoperable for Technical Specification 3.6.1.3, PCIVs. He will need to enter LCO Condition A and Required Action A1 to close and verify deactivated the Main Steam Line A Outboard Valve, 3-FCV-1-15. Additionally he will have to use the indications given in the stem to classify the event per EPIP-1, and will chose 4.2-S.

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



QUESTION 93

Given the following plant conditions:

- BFN is in the process of discharging the Waste Sample Tank to the river in accordance with an approved Discharge Permit.

- The discharge has been in progress for 90 minutes when Security reports that water is bubbling up from the ground in the vicinity of the Standby Gas Treatment Building.

- 0-RR-90-130, Radwaste Effluent Radiation Monitor, is currently reading the same as the initial background radiation level prior to commencing the discharge.

Which ONE of the following completes the statements below regarding the appropriate action and the basis for this action?

Terminate the discharge and ____(1)_____.

Enter procedure EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE, in order to determine _____(2)_____.

A. (1) have Radiation Protection survey the area of the leak
(2) the source and isotopic analysis of the leak

B. (1) have Radiation Protection survey the area of the leak
(2) if Effluent Concentration Limits have been exceeded

C. (1) have Chemistry sample the spilled water
(2) the source and isotopic analysis of the leak

D. (1) have Chemistry sample the spilled water
(2) if Effluent Concentration Limits have been exceeded

ANSWER: **D**

Radwaste

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		2
	Group #		2
	K/A#	268000 A2.01	
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the RADWASTE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System rupture

Explanation: D CORRECT: In order to answer this question correctly, the candidate must determine the following: Without references, determine the appropriate notification requirements, procedure entry and basis for a rupture in the radwaste system.

A- A Radiation Protection survey is not required in accordance with EPIP-13, DOSE ASSESSMENT, until directed by EPIP-2, UNUSUAL EVENT; EPIP-3, ALERT; EPIP-4, SITE AREA EMERGENCY; or EPIP-5, GENERAL EMERGENCY.

B- A chemistry sample is appropriate but entrance into EPIP-13, DOSE ASSESSMENT, is not required until an Emergency Classification has been entered. This may or may not occur depending on the results of the chemistry sample.

C- A Radiation Protection is not required in accordance with EPIP-13 until directed by either EPIP-2, UNUSUAL EVENT; EPIP-3, ALERT; EPIP-4, SITE AREA EMERGENCY; or EPIP-5, GENERAL EMERGENCY. In addition, determination of ECLs (Effluent Concentration Levels) is by chemistry sample, not radiation surveys.

Technical Reference(s): EPIP-1, EPIP-2, EPIP-5

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: BFN 0610 #91

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

10 CFR Part 55 Content: 55.43(b) (4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

MAIN STEAM LINE BREAK	LIQUID EFFLUENT	
Description	Description	
4.2-U	4.3-U	
Main Steam Line break outside Primary Containment with isolation. OPERATING CONDITION: Mode 1 or 2 or 3	Liquid release rate exceeds 20 times ECL as determined by chemistry sample AND Release duration exceeds or will exceed 60 minutes. OPERATING CONDITION: ALL	UNUSUAL EVENT
	4.3-A	
	Liquid release rate exceeds 2000 times ECL as determined by chemistry sample AND Release duration exceeds or will exceed 15 minutes. OPERATING CONDITION: ALL	ALERT
4.2-S		
Unisolable Main Steam Line break outside Primary Containment. OPERATING CONDITION: Mode 1 or 2 or 3		SITE EMERGENCY
		GENERAL EMERGENCY

LIQUID EFFLUENT 4.3-U

UNUSUAL EVENT

EAL: Liquid release rate exceeds 20 times ECL as determined by chemistry sample

AND

Release duration exceeds or will exceed 60 minutes.

OPERATING CONDITION: ALL

BASIS: Liquid release rates are determined using Surveillance Instructions which utilize liquid samples rather than instrument readings for activity determination. Effluent Concentration Limits (ECL) are those annual concentrations given in 10CFR20 Appendix B, Table 2, Column 2. 10 times ECL is equivalent to the instantaneous ODCM limit. Unplanned radioactivity releases that exceed 20 times ECL (2 times ODCM limit) and continue for 60 minutes or longer represent an uncontrolled situation and potential degradation in the level of safety of the plant. The release should not be averaged over 60 minutes. For example, a release of 40 times ECL for 30 minutes does not meet the requirements of this event classification. The 60 minute time period allows sufficient time to isolate any release after exceeding ECL. Greater than 60 minutes represents inability to isolate or control the release. The Site Emergency Director should declare the event as soon as it is determined that the release duration has or will likely exceed 60 minutes. The Chemistry Department determines the magnitude of the release by sample procedure for any release as required by initiating procedures (i.e., SI, ARP, AOI, EOI). The sample results are reported to the Site Emergency Director as a fraction or multiple of ECL.

Escalation to Alert is based on release in excess of 2000 times ECL for greater than 15 minutes.

REFERENCES:
Reg Guide 1.101 Rev. 3, (NUMARC-AU1 example-2)
EDMS L63 010206 800
10CFR20

NOTES:

CURVES/TABLES:

SRO Only Justification: Without references, the SRO will determine that the discharge is the cause of the water is bubbling up from the ground in the vicinity of the Standby Gas Treatment Building. And from that determine that EPIP-1, EMERGENCY CLASSIFICATION PROCEDURE entry will be needed to determine if Effluent Concentration Limits have been exceeded.

SRO Only Guidance

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.

91. Given the following plant conditions:

- BFN is in the process of discharging the Waste Sample Tank to the river in accordance with an approved Discharge Permit.
- The discharge has been in progress for 90 minutes when Security reports that water is bubbling up from the ground in the vacinity of the SGT Building.
- 0-RR-90-130 (Radwaste Effluent Radiation Monitor) is currently reading the same as the initial background radiation level prior to commencing the discharge.

Which ONE of the following describes the appropriate action and the basis for this action?

Terminate the discharge and ________(1)________. Enter procedure EPIP-1 in order to determine ________(2)________.

A. (1) have Radcon survey the area of the leak.

   (2) an Offsite Dose Assessment.

B. (1) have Radcon survey the area of the leak.

   (2) area dose rates and posting requirements.

C. (1) have Chemistry sample the spilled water.

   (2) the source and isotopic analysis of the leak.

D. (1) have Chemistry sample the spilled water.

   (2) if the Effluent Concentration Limits have been exceeded.

QUESTION 94

In accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling , which ONE of the following describes where the **OFFICIAL** copy of the Fuel Assembly Transfer Forms (FATF) shall be maintained during fuel handling?

A. The Fuel Handling Supervisor's desk

B. With the duty Reactor Engineer

C. On the refuel platform

D. In the Control Room

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.1.35	
	Importance Rating		3.9

G2.1.35 Knowledge of the fuel-handling responsibilities of SROs.

Explanation: **A** CORRECT: In accordance with 0-GOI-100-3C, Fuel Movement Operations During Refueling , the copies of the Fuel Assembly Transfer Forms (FATF) shall be maintained at the Fuel Handling Supervisor's desk, on the refuel platform, in the Control Room, and at the refuel floor tag board. The OFFICIAL copy is maintained at the Fuel Handling Supervisor's desk.

B- Incorrect. A copy is not required by 0-GOI-100-3C to be kept with the duty Reactor Engineer

C- Incorrect. A copy is not required by 0-GOI-100-3C to be kept with the duty Reactor Engineer. Fuel Handling Supervisor's desk is missing.

D- Incorrect. A copy is not required by 0-GOI-100-3C to be kept with the Work Control SRO. Fuel Handling Supervisor's desk is missing.

Technical Reference(s): 0-GOI-100-3C

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank:	
	Modified Bank:	
	New:	X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 6 Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

5.1 Operations During Refueling (continued)

NOTE

The copies of the FATFs shall be maintained at their assigned locations.

[2.1] Copies of the FATFs are distributed as follows:

- The official FATF is on the Fuel Handling Supervisor's desk.
- The first copy is on the refuel platform.
- The second copy is in the control room.
- The third copy is at the refuel floor tag board (or tag board equivalent).

SRO Only Justification: The SRO will need to know the procedures (in this case 0-GOI-100-3C) involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. In this case specifically related to the details of where the Fuel Assembly Transfer Forms (FATF) are to be maintained during fuel handling.

Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

F. Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity. [10 CFR 55.43(b)(6)]

Some examples of SRO exam items for this topic include:

- Evaluating core conditions and emergency classifications based on core conditions.
- Administrative requirements associated with low power physics testing processes.
- Administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities.
- Administrative controls associated with the installation of neutron sources.
- Knowledge of TS bases for reactivity controls.

QUESTION 95

Unit 1 is operating at 100% power when a loss of extraction steam to the 1A1 Feedwater Heater is lost. NO operator actions are taken.

Which ONE of the following describes the plant response and the actions required by the Unit Supervisor?

Reactor power will rise and generator output will ___(1)___ slightly.

The Shift Manager must make a report to the NRC within ___(2)___.

A. (1) rise
   (2) 8 hours

B. (1) rise
   (2) 24 hours

C. (1) lower
   (2) 8 hours

D. (1) lower
   (2) 24 hours

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.1.43	
	Importance Rating		4.3

G2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Explanation: **B** CORRECT -Since it is the #1 heater generator output will go up slightly per 1-AOI-6-1A, High Pressure Feedwater Heater String/Extraction Steam Isolation. This will cause the plant to exceed licensed power limits requiring a 24 hour report.

A Incorrect –First Part: Correct. Second Part: Incorrect. While 8 hour reports are required for some items, exceeding a licensed condition requires an 24 hour report.

C Incorrect – First Part: Incorrect. Since it is the #1 heater generator output will go up slightly per 1-AOI-6-1A, High Pressure Feedwater Heater String/Extraction Steam Isolation. Plausible since generator output will lower if any other heater is lost. Second Part: Some discoveries/events are reportable even if they do not actually cause a problem (RPS initiation).

D Incorrect – First Part: Incorrect. Since it is the #1 heater generator output will go up slightly per 1-AOI-6-1A, High Pressure Feedwater Heater String/Extraction Steam Isolation. Plausible since generator output will lower if any other heater is lost. Second Part: Incorrect. See C

Technical Reference(s): 1-AOI-6-1A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source:	Bank: Modified Bank: X New
Question History:	Previous NRC: Vermont Yankee 2010 #94
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis: X

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

B. The following symptoms may occur for a closure of an extraction steam isolation valve:

1. Heater drain cooler flow rises for the next higher heater.

2. Heater drain cooler flow lowers on the affected heater.

3. Feedwater temperature lowers.

4. Reactor power rises.

5. Slight lowering in generator output if affected heater is NOT number one feedwater heater.

2.0 SYMPTOMS (continued)

6. Slight rise in generator output if affected heater is number one feedwater heater.

Appendix A
(Page 7 of 14)

Reporting of Events or Conditions Affecting Licensed Nuclear Power Plants

3.1 **Immediate Notification - NRC (continued)**

(i) The results of ensuing evaluations or assessments of plant conditions,

(ii) The effectiveness of response or protective measures taken, and

(iii) Information related to plant behavior that is not understood.

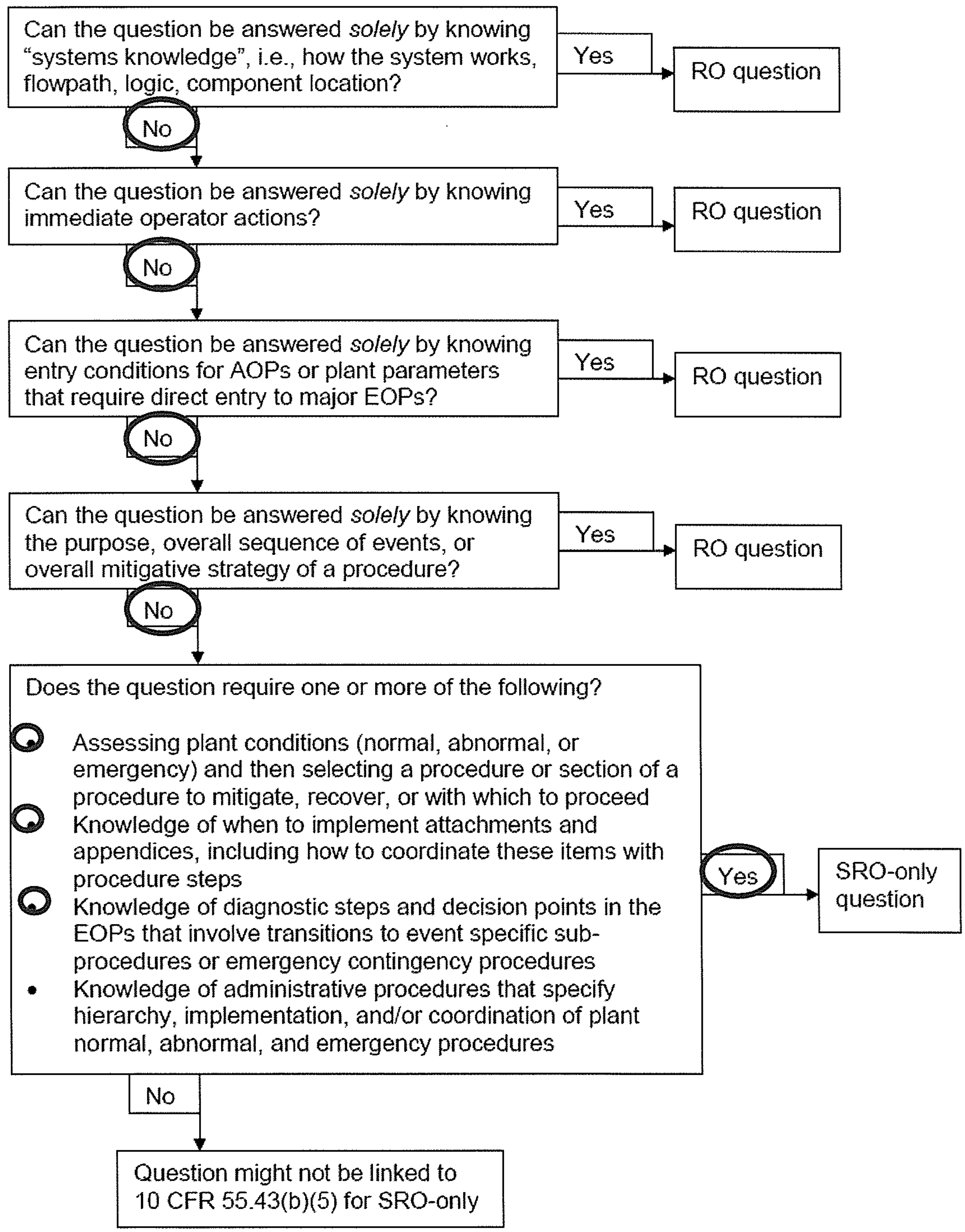
(2) Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.

3.2 **Twenty-Four Hour Notification - NRC**

Any violation of the requirement contained in specific operating license conditions, shall be reported to NRC in accordance with the license condition.

SRO Only Justification: The SRO will assess the plant conditions to determine that power will rise with a loss of 1A1 Feedwater Heater per 1-AOI-6-1A, and that after the 5% power reduction AOI immediate action, the correct subsequent action the SRO should direct is to refer to 1-OI-6 for turbine/heater load restrictions.

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



Vermont Yankee 2010 NRC #94

ES-401	Sample Written Examination Question Worksheet	Form ES-401-5

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #	2.1.43	
	Importance Rating		4.3

(K&A Statement) 2.1.43 Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.

Proposed Question: SRO 94

With a plant startup in progress and operating at 60% RTP, the extraction steam supply from the High Pressure Turbine is lost to the applicable Heater on the "A" string.

Mechanical Maintenance has determined the supply will not be restored for another ten hours.

(1) How is the Feedwater Heater string bypassed
(2) What is the operational restriction with a Feedwater Heater String bypassed?

A. (1) isolate the heater string IAW OP 2172, "Feedwater System"
(2) IAW OT 3110, "Positive Reactivity Insertion", reduce reactor power to <23% RTP

B. (1) isolate the heater string IAW OP 2172, "Feedwater System"
(2) IAW OP 0105, "Reactor Operations", do not exceed 75% with A Feedwater Heater string isolated.

C. (1) isolate the heater string IAW RP 2170, "Condensate System"
(2) IAW OT 3110, "Positive Reactivity Insertion", IAW OT 3110, "Positive Reactivity Insertion", reduce reactor power to <23% RTP

D. (1) isolate the heater string IAW RP 2170, "Condensate System"
(2) IAW OP 0105, "Reactor Operations", do not exceed 75% RTP with A Feedwater Heater string isolated.

Proposed Answer: A

QUESTION 96

Unit 2 is in MODE 4. The forced outage schedule has the unit making a mode change to MODE 2 later this shift. A problem with the RHR 2A pump breaker has just been identified and the repair is estimated to go beyond this shift. The Shift manager has requested a risk assessment be performed per Technical Specification 3.0.4.b to make the mode change to MODE 2.

Which ONE of the following completes the statement?

Per NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, a MODE Restraint Assessment for Tech Spec 3.0.4(b) can ONLY be used if _______________________.

A. there is reasonable likelihood that the RHR 2A will be made Operable within the applicable completion time once the MODE 2 is entered

B. a single TS/TR/Maintenance Rule a(4) system/component is impacted

C. Operations has identified that RHR 2A is a HIGHER RISK System/Component

D. no unusual conditions are present such as impending weather conditions or grid disturbances

Answer: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.2.18	
	Importance Rating		3.9

G2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.

Explanation: **A** CORRECT: Per NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, a MODE Restraint Assessment for Tech Spec 3.0.4(b) is used there is reasonable likelihood that the Inoperable equipment will be made Operable within the applicable completion time once the MODE is entered.

B Incorrect –Plausible because this is one of the four required criteria listed in NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, when a risk assessment is NOT required to be performed.

C Incorrect – Plausible because this is an SSC category that is specifically prohibited by NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, from having a risk assessment performed.

D Incorrect – Plausible because this is one of the four required criteria listed in NPG-SPP-09.11.2, Risk Assessment Methods for Technical Specifications, when a risk assessment is NOT required to be performed.

Technical Reference(s): NPG-SPP-09.11.2

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

3.2.1 SR/TSR 3.0.3/4.0.3 - Missed SR Assessment (continued)

C. Qualitative Risk Assessment

For SR/TSRs related to component or parameters not specifically modeled in the plant PRA or of low risk significance, a qualitative risk assessment may be performed.

3.2.2 LCO/TR 3.0.4(b) - MODE Restraint Assessments

A. General Requirements

1. TS LCO/TR 3.0.4(b) allows entry into a MODE or specified condition in the applicability with inoperable systems or components; provided a risk assessment is performed and any necessary risk management actions are identified. The risk impact of the MODE change must be assessed and considered, and risk management actions defined as appropriate using the plant programs established to implement Section (a)(4) of the Maintenance Rule (10CFR50.65). These programs are defined in NPG-SPP-03.4, "Maintenance Rule Performance, Indicator Monitoring, Trending, and Reporting 10CFR50.65"; NPG-SPP-07.1, "On-line Work Management"; NPG-SPP-07.2.11, "Shutdown Risk Management"; and NPG-SPP-09.11.1, "Equipment Out of Service (EOOS) Management".

2. TS LCO/TR 3.0.4 is to be used to go to higher modes of power operation. TS LCO/TR 3.0.4 shall not prevent changes in MODES or other specified conditions in Applicability that are required to comply with actions or part of a shutdown of a unit.

3. This provision should only be used when there is reasonable likelihood that the inoperable equipment will be made Operable within the applicable completion time once the MODE is entered. This provision is intended to be used when unanticipated circumstances occur which would otherwise delay unit startup. It is not intended for routine, intentional use.

4. If a surveillance has not been performed within its specified frequency SR/TSR 3.0.3/4.0.3 provides an allowance to delay declaring the associated LCO/TR not met. The delay allows time for the surveillance or a risk assessment to be performed. If the LCO/TR is declared not met then entry into a MODE or specified condition in the Applicability shall be made in accordance with LCO/TR 3.0.4. When an LCO is not met due to Surveillances not having been met, entry into a MODE or other condition in the Applicability shall only be made in accordance with LCO/TR 3.0.4."

5. The scope of risk assessments currently performed for 10CFR50.65 a(4) include equipment as defined in NPG-SPP-03.4, "Maintenance Rule Performance, Indicator Monitoring, Trending, and Reporting 10CFR50.65".

6. The risk assessment performed by Corporate PRA will be in accordance with NEDP-26. All inoperable Technical Specification equipment as well as the risk significant equipment included in the Maintenance Rule (a)(4) scope is included. The risk assessments performed for TS LCO/TR 3.0.4(b) will use the existing Maintenance Rule (a)(4) scope and NPG-SPP-09.11.1 as a base and explicitly consider, on a case-by-case basis, any additional scope requirements due to existing Technical Specifications Inoperable equipment.

3.2.2 LCO/TR 3.0.4(b) - MODE Restraint Assessments (continued)

BFN Only

2. Generic risk assessments have been performed by the BWROG to justify the Mode change for conditions when only one Technical Specification/Technical Requirements Manual system/component is Inoperable. The generic analysis identified systems for which no MODE change is allowed if the LCO is not met (i.e. LCO 3.0.4(b) does not apply) which are listed below:

System	MODE to be entered (LCO 3.0.4(b) does not apply)
Diesel Generators	1, 2, 3
HPCI	1, 2
RCIC	1, 2

3.2.2 LCO/TR 3.0.4(b) - MODE Restraint Assessments (continued)

C. Risk Assessment - General Requirements

1. To determine if the mode change will be allowed refer to Figure 1 for a flow chart giving the general approach to TS LCO/TR 3.0.4(b) risk evaluation.

NOTE

All modes changes made using the TS LCO/TR 3.0.4(b) provision must be approved by the Plant Manager or Designee and documented on Attachment 1.

2. Mode changes are allowed without a risk assessment provided that all four of the following items are true:

   a. Only a single TS/TR/Maintenance Rule a(4) system/component impacted.

   b. The system/component is not a high risk system.

   c. There is a high probability that the work involved can be completed within the expected completion time.

   d. No unusual conditions are present such as impending weather conditions or grid disturbances.

SRO Only Justification: The SRO must have knowledge of NPG-SPP-9.11.2 Risk Assessment Methods for Technical Specifications.

Figure 1
(Page 1 of 1)

Flow Chart for Risk Assessment

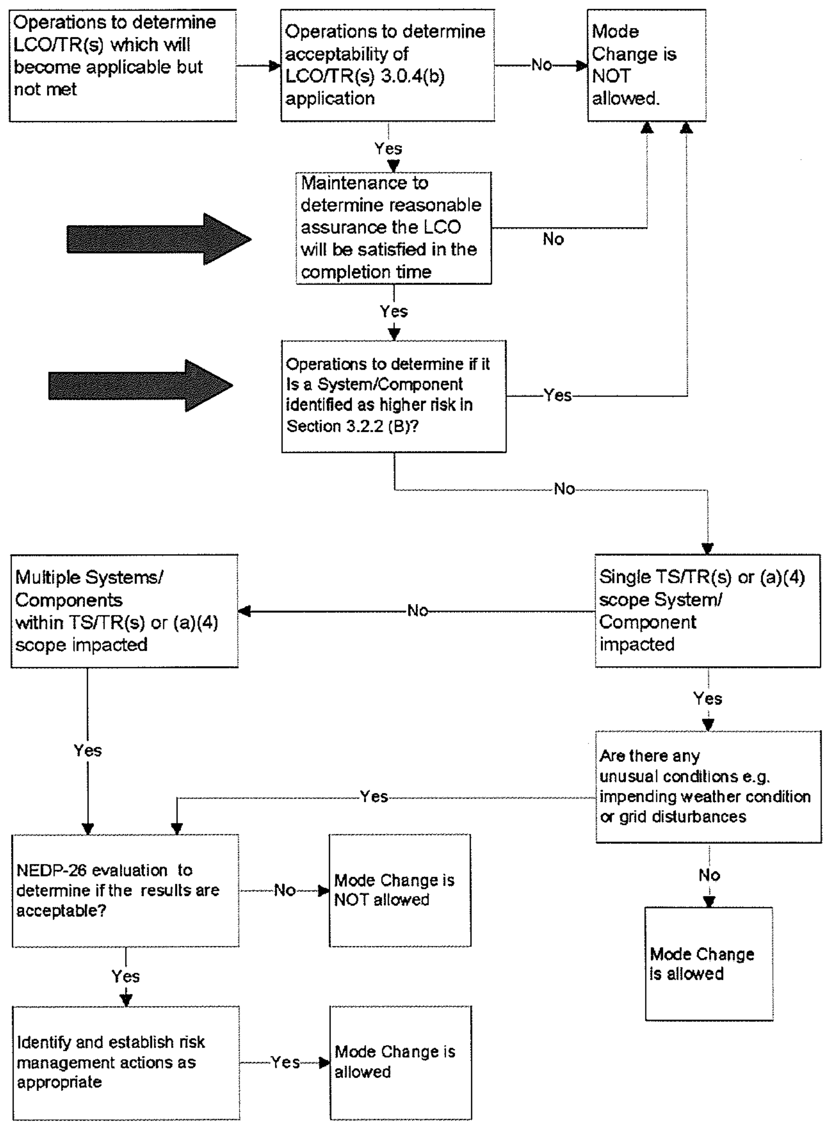
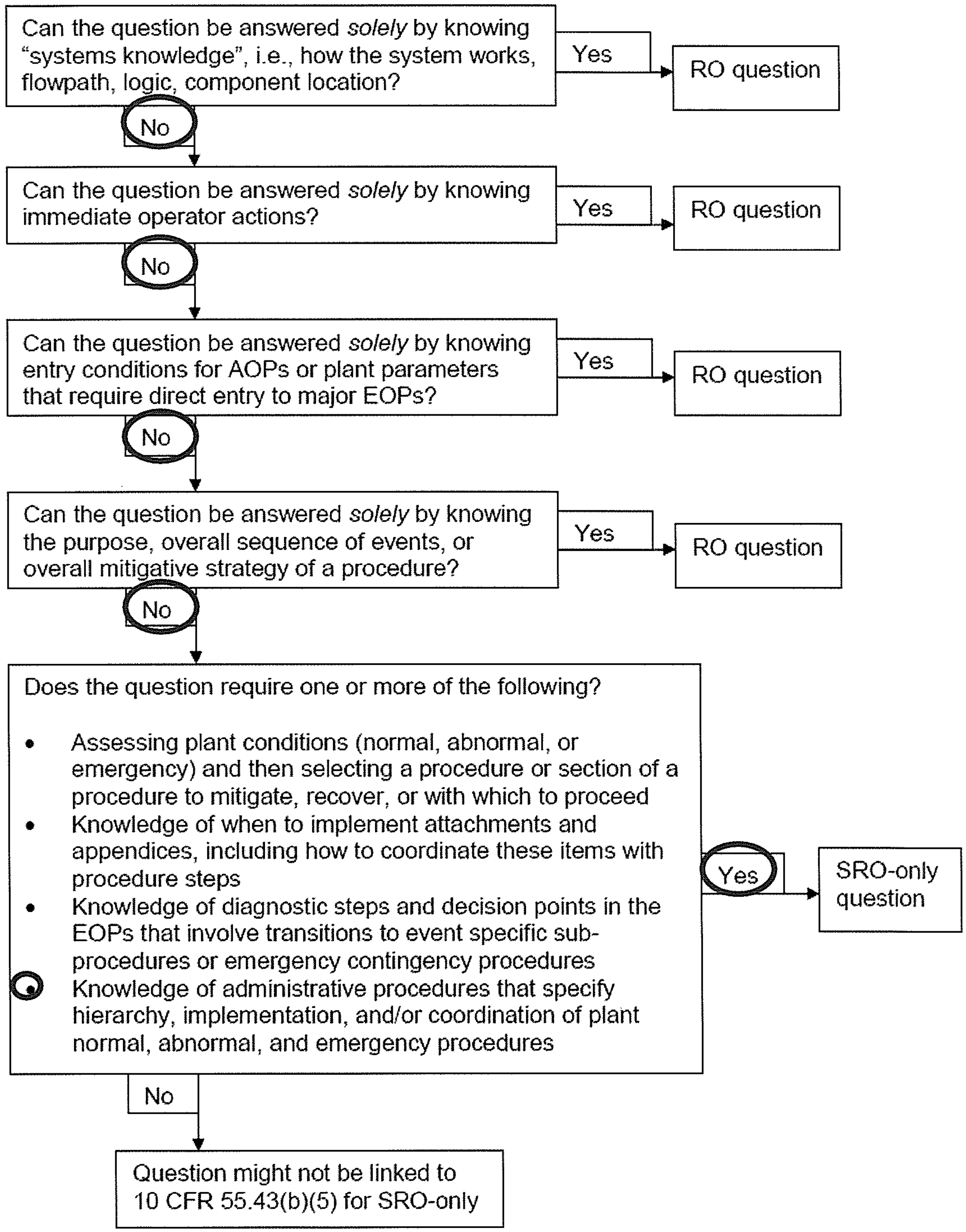


Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)
(Assessment and selection of procedures)



QUESTION 97

In accordance with EOI Flowchart C-1, ALTERNATE LEVEL CONTROL, which ONE of the following completes the statement?

The Minimum Zero Injection RPV Water Level, MZIRWL, value is ___(1)___ inches for ___(2)___.

A. (1) (-)200
(2) all three Units

B. (1) (-)195
(2) Units 1 and 2 ONLY

C. (1) (-)195
(2) Units 2 and 3 ONLY

D. (1) (-)200
(2) Unit 2 ONLY

ANSWER: **C**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.2.3	
	Importance Rating		3.9

G2.2.3 Knowledge of the design, procedural, and operational differences between units.

Explanation: **C** CORRECT: -195 inches is the Minimum Zero Injection Reactor Water Level for Units 2 and 3.

A- Incorrect. First Part: Correct. Second Part: Incorrect, -200 inches is the Minimum Zero Injection Reactor Water Level for Unit 1, NOT all three Units. Plausible in that the candidate may not know there is a difference between units.

B- Incorrect. Plausible since -195 inches is the Minimum Zero Injection Reactor Water for Units 2 and 3, NOT Units 1 and 2.

D- Incorrect. Plausible since -200 inches is the Minimum Zero Injection Reactor Water for Units 1, NOT Unit 2.

Technical Reference(s): 1 (2,3)-EOI Flowchart C-4

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: No

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

18. Step C1-18

This decision step is reached from step C1-17 or from step C1-21 (third override).

RPV water level at this point is somewhere between -180" and MZIRWL".

UNIT DIFFERENCE

MZIRWL is:

Unit 1 -200"

Unit 2 -195"

Unit 3 -195"

a. If this decision step was reached from the steam cooling override (C1-21), and water level cannot be restored and maintained above -180" then emergency depressurization is required at this point because:

1) The calculation for the MZIRWL assumes that there is no subcooling at the core inlet. Since any injection would invalidate this assumption, adequate core cooling cannot be assured if RPV water level is below -180" and water is being injected into the RPV.

MZIRWL, Minimum Zero Injection Reactor Water Level is defined in OPL171.201, Introduction to EOI's

UNIT 1

STEAM COOLING IS REQUIRED
(EOI-1, RC/P-7)

C1-20

WHILE EXECUTING STEPS C1-22 THROUGH C1-23:

IF	THEN
DW CONTROL AIR BECOMES UNAVAILABLE	**CROSSTIE** CAD TO DW CONTROL AIR (APPX 8G)
RPV PRESS IS RISING AND CANNOT BE STABILIZED OR MSRVs ARE BEING USED TO STABILIZE RPV PRESS AND THE CONTINUOUS PNEUMATIC SUPPLY IS LOST OR EMERGENCY RPV DEPRESSURIZATION IS REQUIRED	**CONTINUE** AT STEP C1-12 A
ANY INJECTION SOURCE IS ALIGNED WITH AT LEAST ONE PUMP RUNNING	**CONTINUE** AT STEP C1-18 B

C1-21

CAUTION

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140° F

STABILIZE RPV PRESS NEAR THE EXISTING VALUE WITH THE FOLLOWING SYSTEMS:

DEPRESSURIZATION SYSTEM	APPX
MSRVs ONLY WHEN SUPPR PL LVL IS ABOVE 5.5 FT	11A
HPCI, WITH SUCTION FROM CST IF POSSIBLE	11C
RCIC, WITH SUCTION FROM CST IF POSSIBLE	11B

C1-22

WHEN RPV WATER LVL DROPS TO -200 IN.

THEN **CONTINUE**

C1-23

UNIT 2

STEAM COOLING IS REQUIRED
(EOI-1, RC/P-7)

C1-20

WHILE EXECUTING STEPS C1-22 THROUGH C1-23:

IF	THEN
DW CONTROL AIR BECOMES UNAVAILABLE	**CROSSTIE** CAD TO DW CONTROL AIR (APPX 8G)
RPV PRESS IS RISING AND CANNOT BE STABILIZED OR MSRVs ARE BEING USED TO STABILIZE RPV PRESS AND THE CONTINUOUS PNEUMATIC SUPPLY IS LOST OR EMERGENCY RPV DEPRESSURIZATION IS REQUIRED	**CONTINUE** AT STEP C1-12 A
ANY INJECTION SOURCE IS ALIGNED WITH AT LEAST ONE PUMP RUNNING	**CONTINUE** AT STEP C1-18 B

C1-21

CAUTION

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140°F

STABILIZE RPV PRESS NEAR THE EXISTING VALUE WITH THE FOLLOWING SYSTEMS:

DEPRESSURIZATION SYSTEM	APPX
MSRVs ONLY WHEN SUPPR PL LVL IS ABOVE 5.5 FT	11A
HPCI, WITH SUCTION FROM CST IF POSSIBLE	11C
RCIC, WITH SUCTION FROM CST IF POSSIBLE	11B

C1-22

WHEN RPV WATER LVL DROPS TO -195 IN.

THEN **CONTINUE**

C1-23

UNIT 3

SRO Only Justification:

STEAM COOLING IS REQUIRED
(EOI-1, RC/P-7)

C1-20

WHILE EXECUTING STEPS C1-22 THROUGH C1-23:

IF	THEN
DW CONTROL AIR BECOMES UNAVAILABLE	**CROSSTIE** CAD TO DW CONTROL AIR (APPX 8G)
RPV PRESS IS RISING AND CANNOT BE STABILIZED OR MSRVs ARE BEING USED TO STABILIZE RPV PRESS AND THE CONTINUOUS PNEUMATIC SUPPLY IS LOST OR EMERGENCY RPV DEPRESSURIZATION IS REQUIRED	**CONTINUE** AT STEP C1-12 A
ANY INJECTION SOURCE IS ALIGNED WITH AT LEAST ONE PUMP RUNNING	**CONTINUE** AT STEP C1-18 B

C1-21

CAUTION

#3 ELEVATED SUPPR CHMBR PRESS MAY TRIP RCIC

#6 HPCI OR RCIC SUCTION TEMP ABOVE 140 °F

STABILIZE RPV PRESS NEAR THE EXISTING VALUE WITH THE FOLLOWING SYSTEMS:

DEPRESSURIZATION SYSTEM	APPX
MSRVs ONLY WHEN SUPPR PL LVL IS ABOVE 5.5 FT	11A
HPCI, WITH SUCTION FROM CST IF POSSIBLE	11C
RCIC, WITH SUCTION FROM CST IF POSSIBLE	11B

C1-22

WHEN RPV WATER LVL DROPS TO -195 IN.

THEN **CONTINUE**

C1-23

SRO Only Justification: The SRO is required to have knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event specific subprocedures or emergency contingency procedures. The Minimum Zero Injection Reactor Water Level, MZIRWL value is a major transition point in the application of EOI Flowchart C-1, ALTERNATE LEVEL CONTROL. If this RPV water level is reached during steam cooling, the EOI's require transitioning to EOI-2, Emergency depressurization.

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

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

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

Yes → RO question

No

Does the question require one or more of the following?

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

Yes → SRO-only question

No

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

QUESTION 98

An individual who has already received an emergency TEDE dose of 23 rem has volunteered to accept another emergency exposure.

Which ONE of the following completes the statements?

He/she __(1)__ participate in this emergency exposure.

The __(2)__is (are) required to authorize an emergency exposure.

A. (1) CANNOT
   (2) Site Emergency Director ONLY

B. (1) CANNOT
   (2) Site Emergency Director AND Radiation Protection Manager

C. (1) CAN
   (2) Site Emergency Director ONLY

D. (1) CAN
   (2) Site Emergency Director AND Radiation Protection Manager

ANSWER: **A**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.3.4	
	Importance Rating		3.7

G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Explanation: **A** CORRECT: Part 1 is correct. This is considered a once-in-a-lifetime exposure, therefore he/she cannot participate. Part 2 is correct, Per EPIP-15 Appendix-B, the SED is the only required signature.

B- First Part: Correct. This is considered a once-in-a-lifetime exposure, therefore he/she cannot participate. Second Part: Incorrect, Per EPIP-15 Appendix-B, the SED is the <u>only</u> required signature.

C- First Part: Incorrect. This is considered a once-in-a-lifetime exposure, therefore he/she <u>cannot</u> participate. Second Part: Correct.

D- First Part: Incorrect. This is considered a once-in-a-lifetime exposure, therefore he/she <u>cannot</u> participate since > 25 rem was received in the previous exposure. Second Part: Incorrect, Per EPIP-15 Appendix-B, the SED is the <u>only</u> required signature.

Technical Reference(s): EPIP-15, Emergency Exposure

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank: X
Modified Bank:
New:

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43(b) 5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

3.3 **Guidance for All Emergency Dose Limits**

3.3.1 The exposure of personnel during emergency operations shall be maintained As Low As Reasonably Achievable (ALARA).

3.3.2 Internal exposure should be minimized by the use of respiratory protection equipment. Respirator Protection Factors are provided in Standard Programs and Processes (SPP) 5.10. If a projected dose to the thyroid is expected to exceed 10 rem during emergency conditions, Potassium Iodide (KI) should be issued. EPIP-14 contains information regarding issuance and precautions for the use of KI.

3.3.3 Personnel undertaking an emergency operation in which the dose will exceed 10 CFR 20.1201 entitled "Occupational Dose for Adults" limits shall do so on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects as depicted on Appendix A. Acknowledgment of this decision shall be documented on Appendix B of this procedure by the individual involved in this activity.

3.3.4 Other factors being equal, older volunteers should be selected first.

3.3.5 Other factors being equal, selection of female volunteers capable of reproduction should be avoided.

3.3.6 Exposures under these conditions shall be limited to a once in a lifetime. Personnel who have received previous accident or emergency exposures in excess of 25 rem TEDE shall not participate in further emergency exposure assignments.

APPENDIX B

ACKNOWLEDGMENT AND AUTHORIZATION TO EXCEED OCCUPATIONAL DOSE LIMITS

READ THE FOLLOWING STATEMENT BEFORE SIGNING THIS FORM:

I acknowledge by signature on this form that I am volunteering for exposures in excess of 10 CFR 20.1201 limits and that I have been made aware through training or a briefing of the risks involved. Briefing material was presented from Appendix A of this procedure.

The persons listed below have acknowledged and volunteered to receive dose limits in excess of 10CFR20.1201 limits. Authorization is required by the Site Emergency Director to administer any emergency exposure limit. Authorization is acknowledged by Site Emergency Director signature on the bottom of this form.

Name (Please print Last, First, MI)	Employee Identification Number (EIN)	Signature	Dose Limit (Rem)

Brief Description of Task: ______________________________

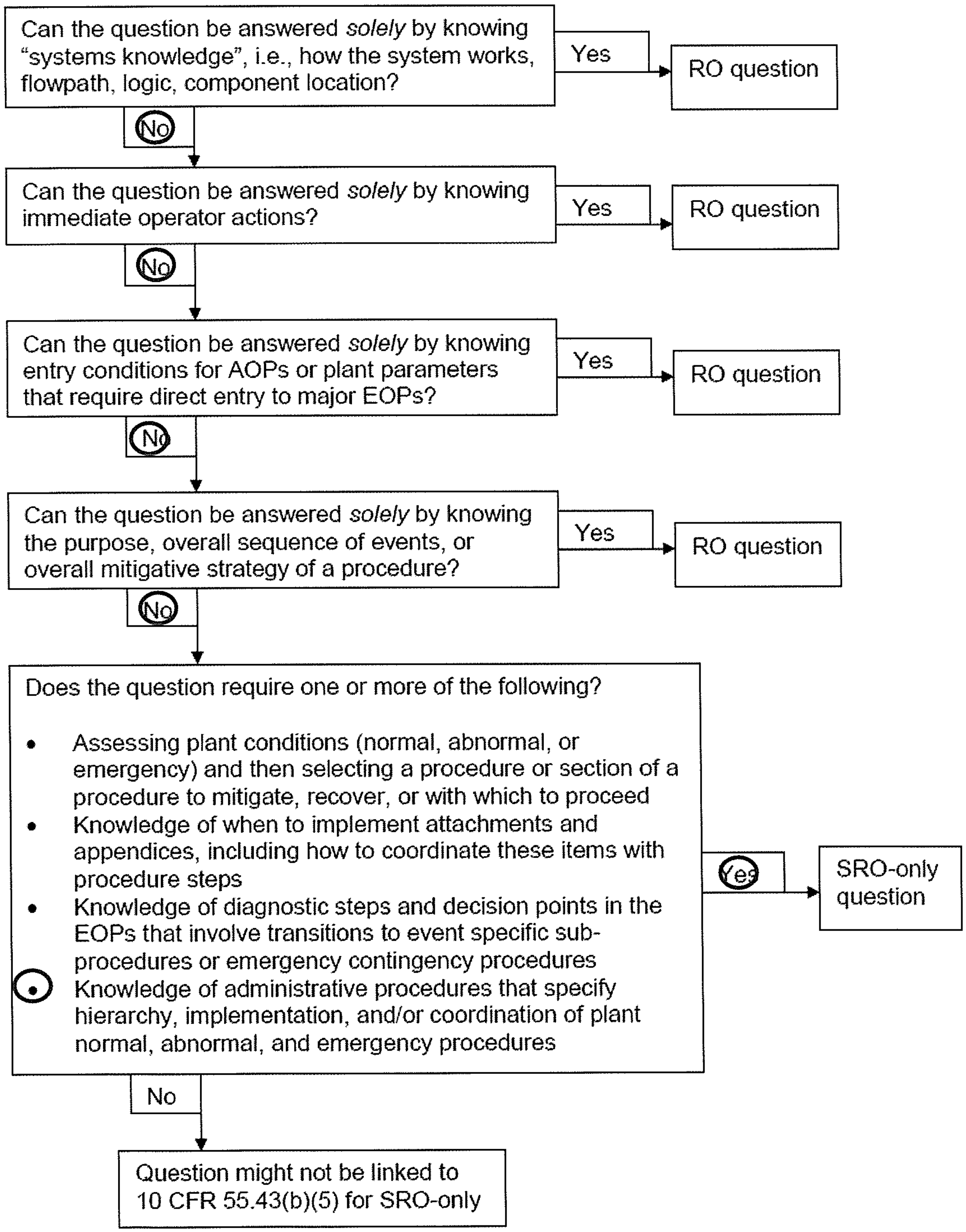
____

Authorized by : ______________________ / ______________________

Site Emergency Director   Time/Date

SRO Only Justification: The SRO will need to know that EPIP-15, Emergency Exposures details the specifics regarding selection of a candidate for Emergency expose including the fact that the candidate must be a volunteer, and can only perform one emergency exposure per life-time. Additionally the SRO would need to know that the form authorizing exceeding occupational dose limits is approved by the Site Emergency Director.

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



QUESTION 99

Unit 1 is currently in Mode 1. Fuel movement is in progress in Unit 1 Spent Fuel Pool. During the movement of irradiated fuel, a fuel bundle is severely damaged, resulting in the following conditions.

- Field Assessment Team surveys at the Site Boundary Indicate Iodine-131 levels have exceeded the General Emergency limit.
- Radiation levels in numerous areas of the Reactor Building are above Max Safe.

Which ONE of the following completes the statements?

In accordance with the EOIs the Unit Supervisor must __(1)__.

In accordance with the EOI Program Manual the basis for this action is to __(2)__.

Note: 1-EOI-1, RPV Control
1-EOI-4, Radioactive Release Control
1-GOI-100-12A, Unit Shutdown from Power Operation to Cold Shutdown and Reductions in Power During Power Operations

A. (1) enter 1-GOI-100-12A, from 1-EOI-3 and proceed to Cold Shutdown
(2) mitigate a direct and immediate threat, relative to plant equipment and personnel, both on and off site

B. (1) enter 1-GOI-100-12A, from 1-EOI-3 and proceed to Cold Shutdown
(2) limit the release of radioactivity discharging into areas outside the primary and secondary containments

C. (1) enter 1-EOI-1 from 1-EOI-4, Scram the Reactor and Emergency Depressurize
(2) mitigate a direct and immediate threat, relative to plant equipment and personnel, both on and off site

D. (1) enter 1-EOI-1 from 1-EOI-4, Scram the Reactor and Emergency Depressurize
(2) limit the release of radioactivity discharging into areas outside the primary and secondary containments

ANSWER: A

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.4.18	
	Importance Rating		4.0

G2.4.18 Knowledge of the specific bases for EOPs.

Explanation: Answer **A** – CORRECT: The US must determine that Emergency Depressurization will not reduce radiation levels in secondary containment and with two radiation levels in the Reactor building above max safe, Cold Shutdown is required. The basis for this action is to mitigate a direct and immediate threat, relative to plant equipment and personnel, both on and off site.

B – Incorrect – First part is correct. Second part is plausible in that this is the correct basis for Emergency Depressurization.

C – Incorrect – First part incorrect. The candidate may determine this is the correct action based on meeting the entry requirements for EOI-4 and meeting the Reactor Scram and Emergency Depressurization requirement if a primary system was discharging. Second part is the correct basis for proceeding to cold shutdown .

D– Incorrect – First part incorrect. The candidate may determine this is the correct action based on meeting the entry requirements for EOI-4 and meeting the Reactor Scram and Emergency Depressurization requirement if a primary system was discharging. Second part is plausible in that this is the correct basis for Emergency Depressurization.

Technical Reference(s): 1-EOI-3, Secondary Containment Control 0-EOI-4, Radioactive Release Control, EOI Program Manual

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

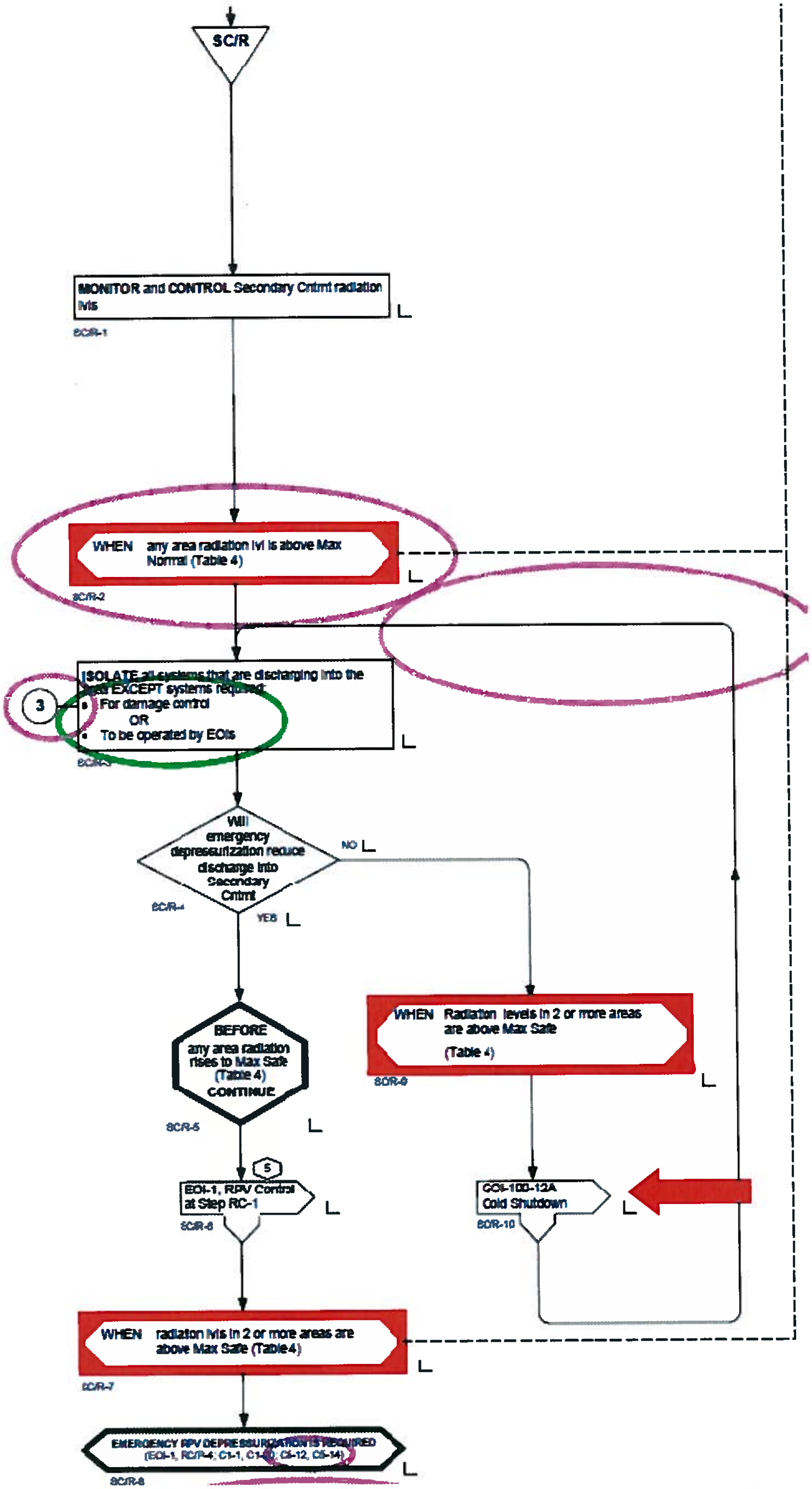
Question Source:	Bank: X Modified Bank: New
Question History:	Previous NRC: No
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis X

10 CFR Part 55 Content: 55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Bases for 1-EOI-3 steps SC/R-9 and 10

This decision step has the operator evaluate status of radiation levels in all secondary containment areas listed in Table 4, to determine if a normal reactor shutdown is required.

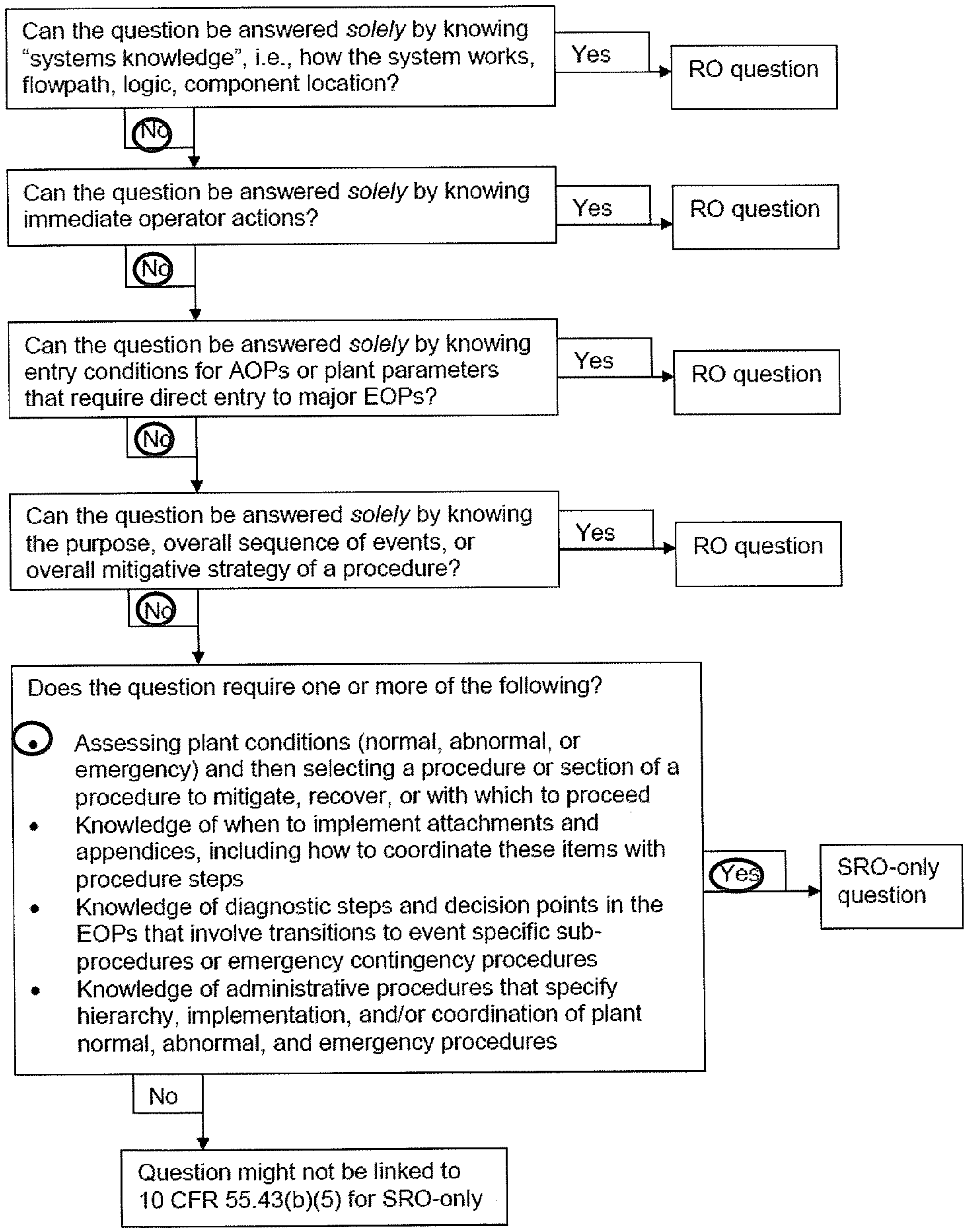
To reach this step, it was previously determined that no primary systems were discharging into secondary containment. When this condition exists, and when radiation levels in two or more secondary containment areas exceed their respective maximum safe operating values, it is prudent to commence an orderly reactor shutdown because a direct and immediate threat exists relative to plant equipment and to personnel, both on and off site. The operator continues in this procedure at Step SC/R-10 where actions to perform a normal plant shutdown are directed.



SC/R
MONITOR and CONTROL Secondary Cntmt radiation lvls
SC/R-1
WHEN any area radiation lvl is above Max Normal (Table 4)
SC/R-2
ISOLATE all systems that are discharging into the area EXCEPT systems required:
• For damage control
OR
• To be operated by EOIs
3
Will emergency depressurization reduce discharge into Secondary Cntmt
SC/R-4
NO
YES
BEFORE any area radiation rises to Max Safe (Table 4) CONTINUE
SC/R-5
WHEN Radiation levels in 2 or more areas are above Max Safe (Table 4)
SC/R-9
5
EOI-1, RPV Control at Step RC-1
SC/R-6
GOI-100-12A Cold Shutdown
SC/R-10
WHEN radiation lvls in 2 or more areas are above Max Safe (Table 4)
SC/R-7
EMERGENCY RPV DEPRESSURIZATION IS REQUIRED
(EOI-1, RC/P-4; C1-1, C1-10; C5-12, C5-14)
SC/R-8

SRO Only Justification: The SRO will assess plant conditions and recognize that no primary systems were discharging into secondary containment and emergency depressurization will not reduce discharge into the secondary containment. Additionally, the SRO will need to know that EOI-3 Secondary Containment Control is the procedure with which to proceed and it will direct a Normal Shutdown per GOI-100-12A based on these conditions. The SRO will also need to know the EOI Program Manual the basis for this action.

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



QUESTION 100

Which ONE of the following completes the statement regarding the proper EOI Flow Chart direction for entry into a Contingency Procedure and the impact of the entry on the use of the original (directing) procedure?

Entry into a Contingency Procedure by the EOI Flowcharts will ___(1)___ and, upon entry to the Contingency Procedure, the Unit Supervisor ___(2)___ exit the original (sending) procedure.

A. (1) be directed in an override OR an exit arrow to the Contingency
   (2) will ALWAYS

B. (1) be directed in an override OR an exit arrow to the Contingency
   (2) may or may not

C. (1) be directed in an override with an exit arrow to the Contingency in the override
   (2) will ALWAYS

D. (1) be directed in an override with an exit arrow to the Contingency in the override
   (2) may or may not

Answer: **B**

Examination Outline Cross-Reference	Level:	RO	SRO
	Tier #		3
	Group #		
	K/A#	G2.4.19	
	Importance Rating		4.1

Knowledge of EOP layout, symbols, and icons.

Explanation: **B** CORRECT: First Part: For example in EOI-1, entry into C-4 is directed as an override RC/L-3 and entry into C-1 in RC/L-13 with a match arrow. Second Part: Entry into C-2 from C-1 directs the operator to exit RC/P but not to exit C-1.

A Incorrect –Plausible as part (1) is correct but as noted above the originating procedure may or may not be exited.

C Incorrect – Plausible Match lines are used but not in conjunction with the overrides.

D Incorrect – Plausible Match lines are used but not in conjunction with the overrides.

Technical Reference(s): EOIPM 0-VIII-A

Proposed references to be provided to applicants during examination: None

Learning Objective (As available):

Question Source: Bank:
Modified Bank:
New: X

Question History: Previous NRC: None

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

10 CFR Part 55 Content: 55.43 b (5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

When the operator reaches an Exit Arrow the flowpath is exited at that point and operation continues in the procedure designated by the Exit Arrow. When the target procedure is entered the operator locates the Entry Arrow displaying the procedure number and step number, or matchmark letter, associated with the Exit Arrow.

WHILE EXECUTING THE FOLLOWING STEPS:

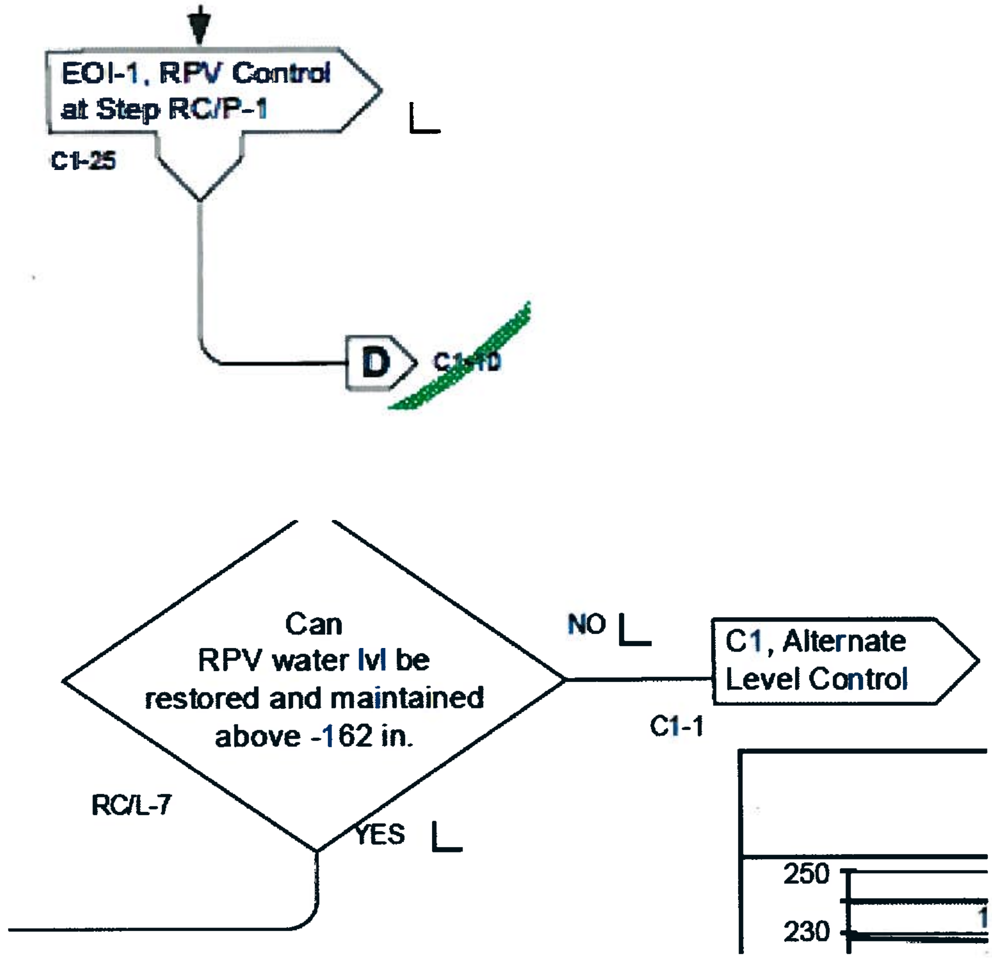
IF	THEN
1 It has NOT been determined that the reactor will remain subcritical without boron under all conditions	**EXIT** RC/L and **ENTER** C5, Level/Power Control
RPV water lvl CANNOT be determined	**EXIT** RC/L and **ENTER** C4, RPV Flooding
PC water lvl CANNOT be maintained below 105 ft OR Suppr chmbr press CANNOT be maintained below 55 psig	**STOP** inj into the RPV from sources external to the PC NOT required for adequate core cooling

RC/L-3

WHILE EXECUTING THIS PROCEDURE:

IF	THEN
Emergency RPV depressurization is required	EXIT RC/P and Steam Cooling ENTER C2, Emergency RPV Depressurization
1 It has NOT been determined that the reactor will remain subcritical without boron under all conditions	EXIT this procedure and ENTER C5, Level/Power Control
RPV water lvl CANNOT be determined	EXIT this procedure and ENTER C4, RPV Flooding
PC water lvl CANNOT be maintained below 105 ft OR Suppr chmbr press CANNOT be maintained below 55 psig	STOP inj into the RPV from sources external to the PC NOT required for adequate core cooling

C1-1


EOI-1, RPV Control
at Step RC/P-1
C1-25
D
C1-10
Can
RPV water lvl be
restored and maintained
above -162 in.
NO
C1, Alternate
Level Control
C1-1
RC/L-7
YES
250
230

SRO Only Justification: The SRO will need to know the proper usage of flow chart marking in the EOI's. This is the responsibility of the SRO to drive the flowcharts and direct operators according the flowchart usage rules.

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