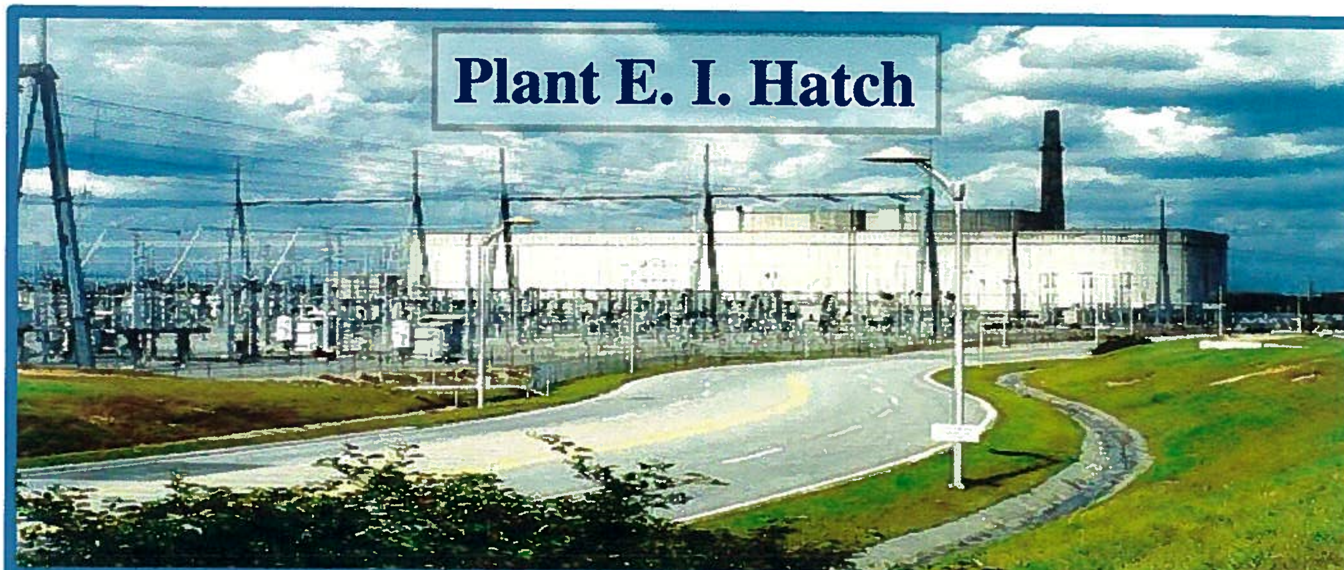
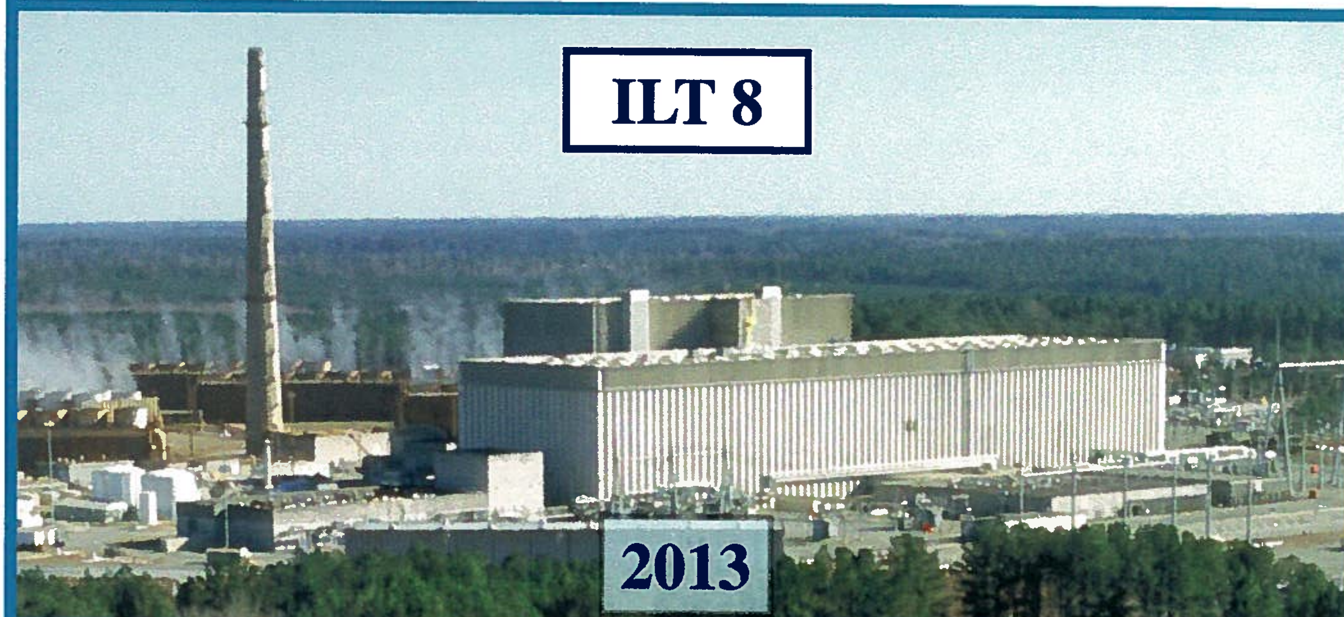
Plant E. I. Hatch



NRC Exam Material

SRO Written Exam
+ References
SRO Q# 1-25 (Exam Q# 76-100)

ILT 8



2013

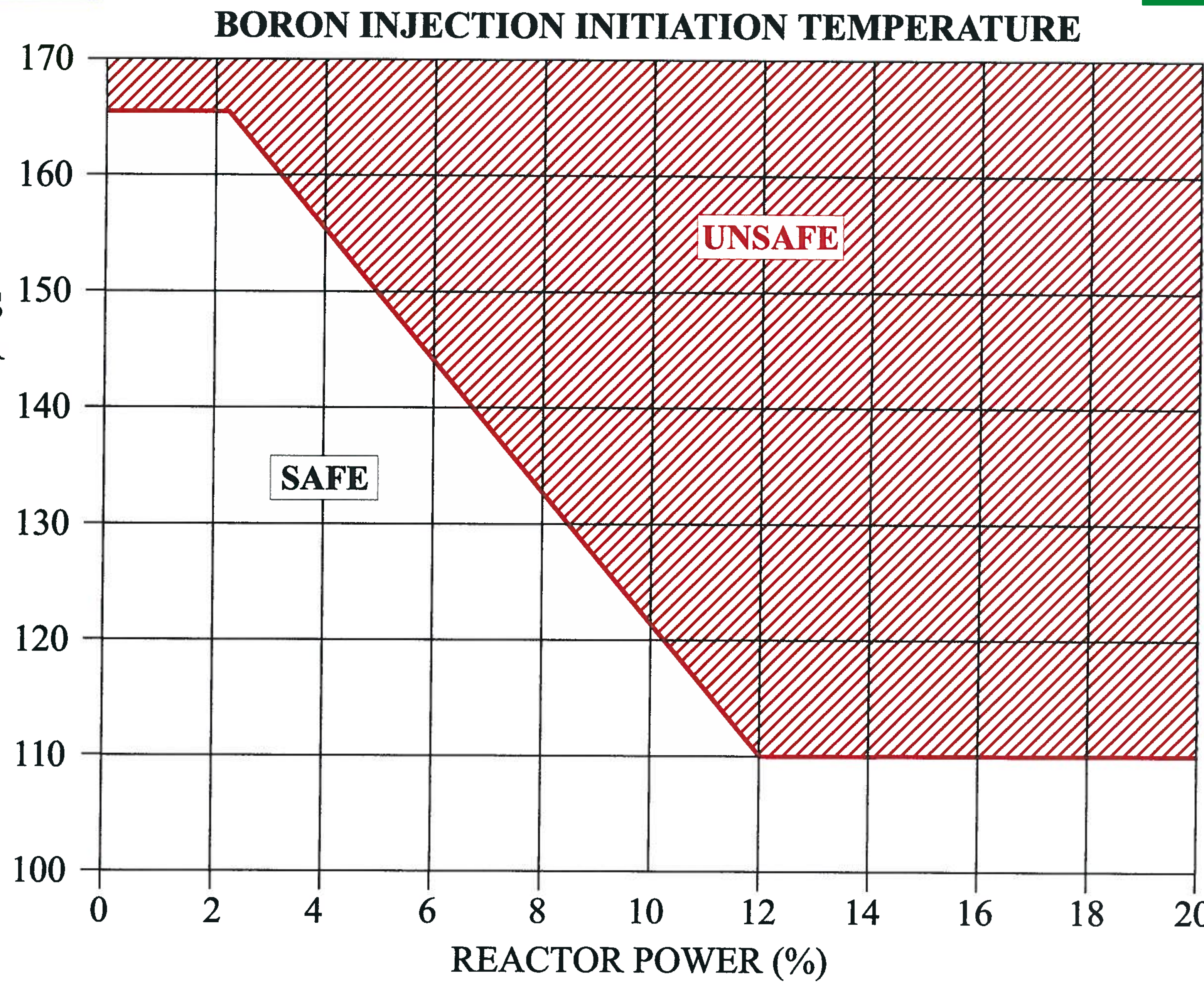
NRC SRO REFERENCES

SRO EXAM

1. *Unit 2 EOP Graph 5, BIIT Curve*
2. *Unit 2 EOP Graph 12A, RHR NPSH Limit, (Torus Water Level At or Above 146") & Unit 2 EOP Graph 12B, RHR NPSH Limit, (Torus Water Level Below 146")*
3. *Unit 1 TS 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation & Unit 1 TS 3.5.1 Emergency Core Cooling System (ECCS)*
4. *Table 6 of 34AB-T22-003-1, Secondary Containment Control & TS 3.7.4 Main Control Room Environmental Control (MCREC) System*
5. *NMP-EP-110-GL02, Figure 1 – Fission Product Barrier Matrix*
6. *Unit 1 EOP Graph 8 Drywell Spray Initiation Curve*
7. *Unit 1 EOP Graph 5 Boron Injection Initiation Temperature Curve*
8. *NMP-EP-110-GL02, "Emergency Classification & Initial Actions", Attachment 2 "Hot" Initiating Condition Matrix Evaluation Chart, AC Power Section*

GRAPH 5

UNIT 2

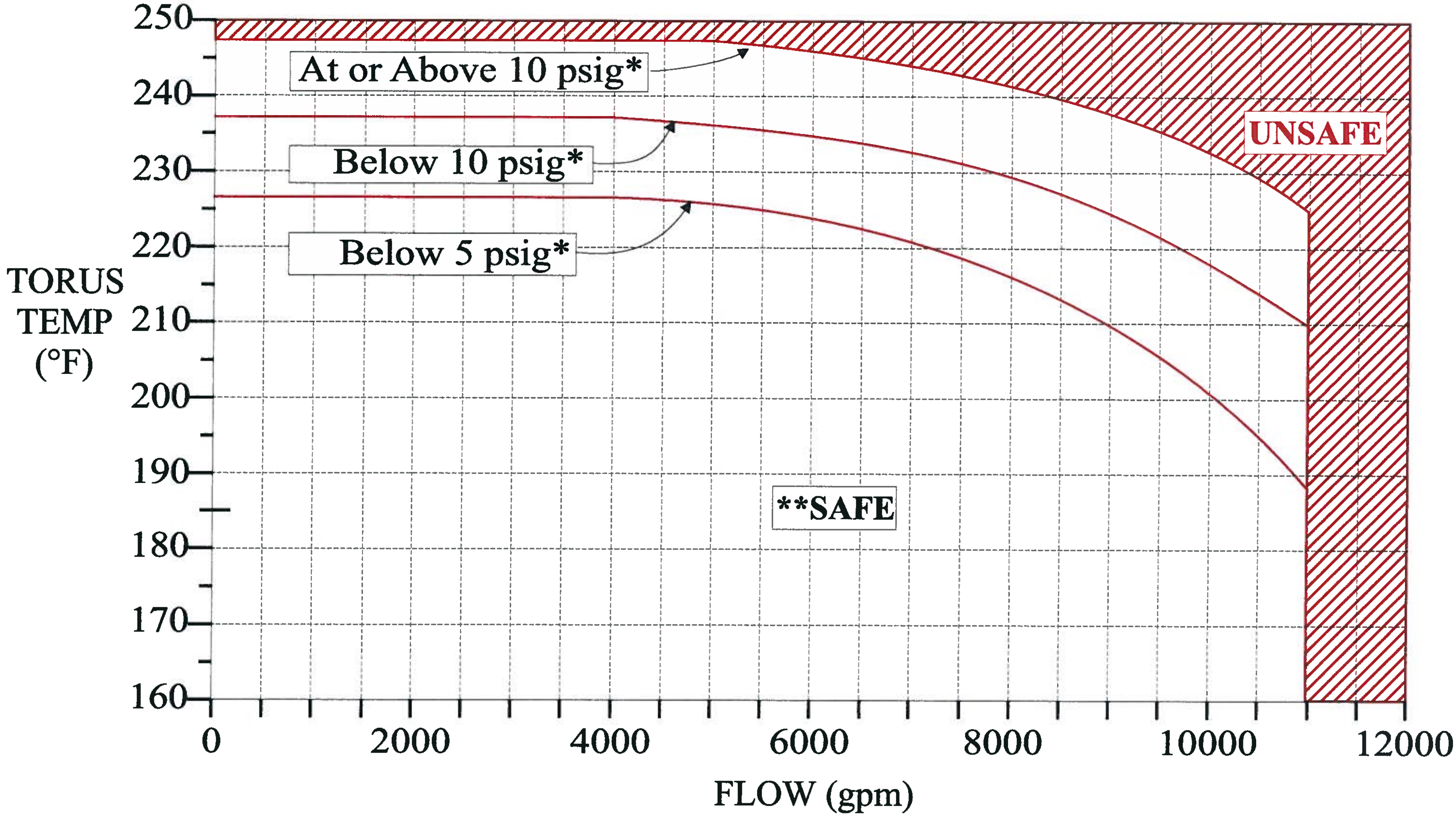


NOTE: May use SPDS Emergency Displays in place of this Graph.

GRAPH 12A

UNIT 2

RHR Pump NPSH Limit
(Suppression Pool Water Level At or Above 146")

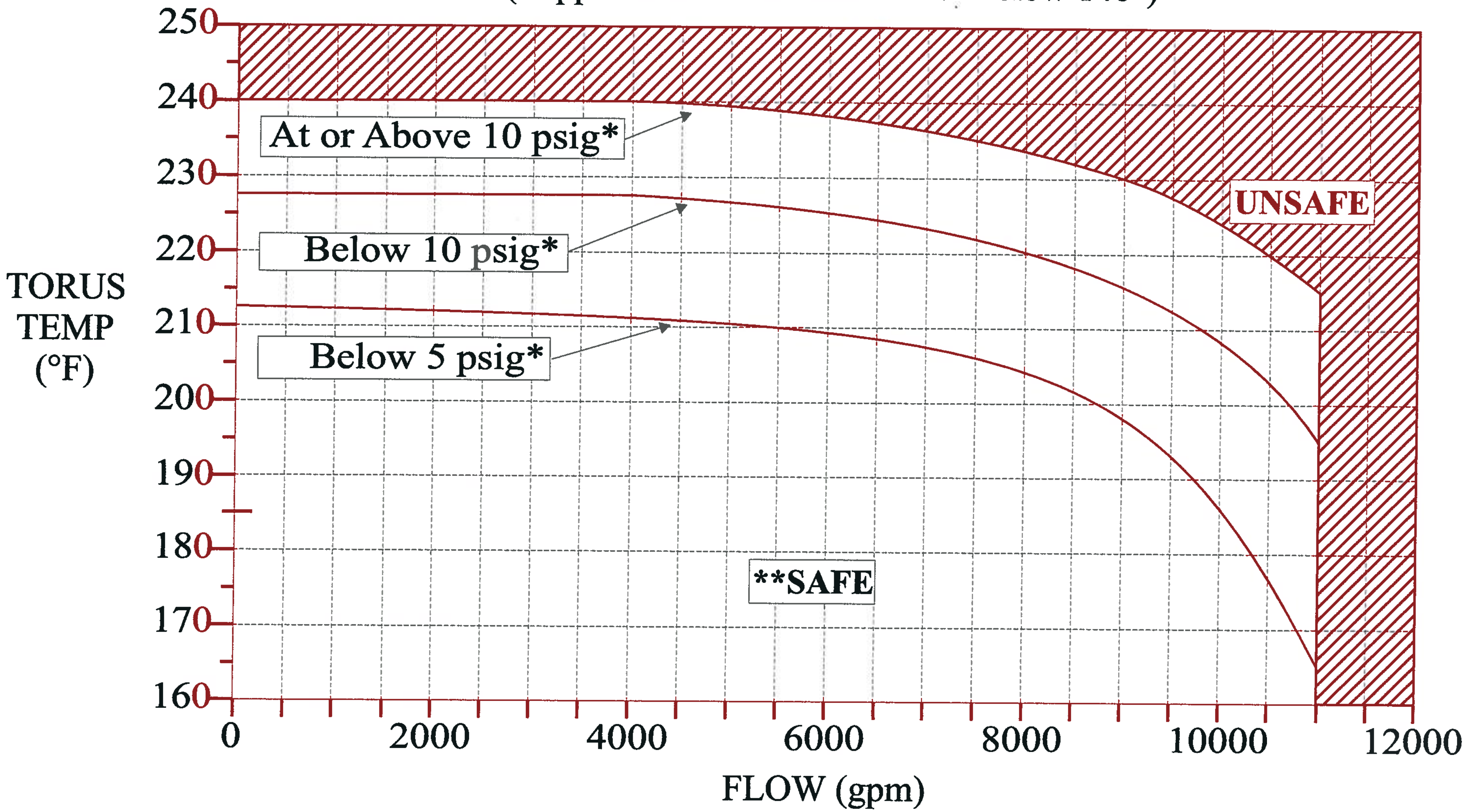


NOTE: May use SPDS Emergency Displays in place of this Graph.
- * Suppression Chamber Pressure.
- ** Safe operating region is below the applicable pressure line.

GRAPH 12B

UNIT 2

RHR Pump NPSH Limit
(Suppression Pool Water Level Below 146")



NOTE: May use SPDS Emergency Displays in place of this Graph.
* Suppression Chamber Pressure.
** Safe operating region is below the applicable pressure line.

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1 -----NOTES----- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. ----- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	-----------NOTE----------- Only applicable for Functions 3.a and 3.b. ------------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	-----------NOTES---------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 2.c, 2.d, and 2.f. ------------------------------	
		Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	------------NOTE------------ Only applicable if HPCI pump suction is not aligned to the suppression pool. -------------------------------- Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	D.2.1	Place channel in trip.	24 hours
	<u>OR</u>		
	D.2.2	Align the HPCI pump suction to the suppression pool.	24 hours
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	-----------NOTES----------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.d and 2.g. -------------------------------- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1 Declare Automatic Depressurization System (ADS) valves inoperable. AND F.2 Place channel in trip.	1 hour from discovery of loss of ADS initiation capability in both trip systems 96 hours from discovery of inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable AND 8 days
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare ADS valves inoperable. AND G.2 Restore channel to OPERABLE status.	1 hour from discovery of loss of ADS initiation capability in both trip systems 96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1
1. Core Spray System			
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2, 3, 4[(a)], 5[(a)]	4[(b)]	B
b. Drywell Pressure - High	1, 2, 3	4[(b)]	B
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	4	C
	4[(a)], 5[(a)]	4	B
d. Core Spray Pump Discharge Flow - Low (Bypass)	1, 2, 3, 4[(a)], 5[(a)]	1 per subsystem	E

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 of seven 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 ≤ 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.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. <u>AND</u> C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. HPCI System inoperable. <u>AND</u> One low pressure ECCS injection/spray subsystem is inoperable.	D.1 Restore HPCI System to OPERABLE status. <u>OR</u> D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. Two or more ADS valves inoperable. <u>OR</u> Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours
F. Two or more low pressure ECCS injection/spray subsystems inoperable. <u>OR</u> HPCI System and two or more ADS valves inoperable.	F.1 Enter LCO 3.0.3.	Immediately

AREA RADIATION MONITORS on 1H11-P600, 1D21-P600		
<u>REFUEL FLOOR AREA</u>		
1 Reactor head laydown area (1D21-K601A)	50	1000
2 Refueling Floor Stairway (1D21-K601B)	50	1000
3 Refueling Floor (1D21-K601D)	50	1000
4 Drywell Shield Plug (1D21-K601E)	50	1000
5 Spent Fuel Pool & New Fuel Storage (1D21-K601M)	50	1000
<u>203' ELEVATION AREA</u>		
6 RB 203' Working Area (1D21-K601X)	50	1000
<u>185' ELEVATION AREA</u>		
7 Spent Fuel Pool Demin. Equip (1D21-K601C)	150	1000
8 Fuel Pool Demin. Panel (1D21-K617)	50	100
<u>158' ELEVATION AREA</u>		
9 RB 158' Working Area (1D21-K601K)	50	1000
10 Rx Wtr Sample Rack Area 158' (1D21-K601L)	50	1000
<u>130' ELEVATION NORTH AREA</u>		
11 TIP Area (1D21-K601F)	50	1000
12 North CRD HCU (1D21-K601P)	50	1000
13 TIP Probe Drives Area (1D21-K601U)	100	1000

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Environmental Control (MCREC) System

LCO 3.7.4 Two MCREC subsystems shall be OPERABLE.

-----NOTE-----
The main control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MCREC subsystem inoperable.	A.1 Restore MCREC subsystem to OPERABLE status.	7 days
B. Two MCREC subsystems inoperable due to inoperable control room boundary in MODE 1, 2, or 3.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------------NOTE------------------ LCO 3.0.3 is not applicable. ---	
	D.1 Place OPERABLE MCREC subsystem in pressurization mode.	Immediately
	OR	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately
E. Two MCREC subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two MCREC subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	----------------NOTE------------------ LCO 3.0.3 is not applicable. --	
	F.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	F.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	F.3 Initiate action to suspend OPDRVs.	Immediately

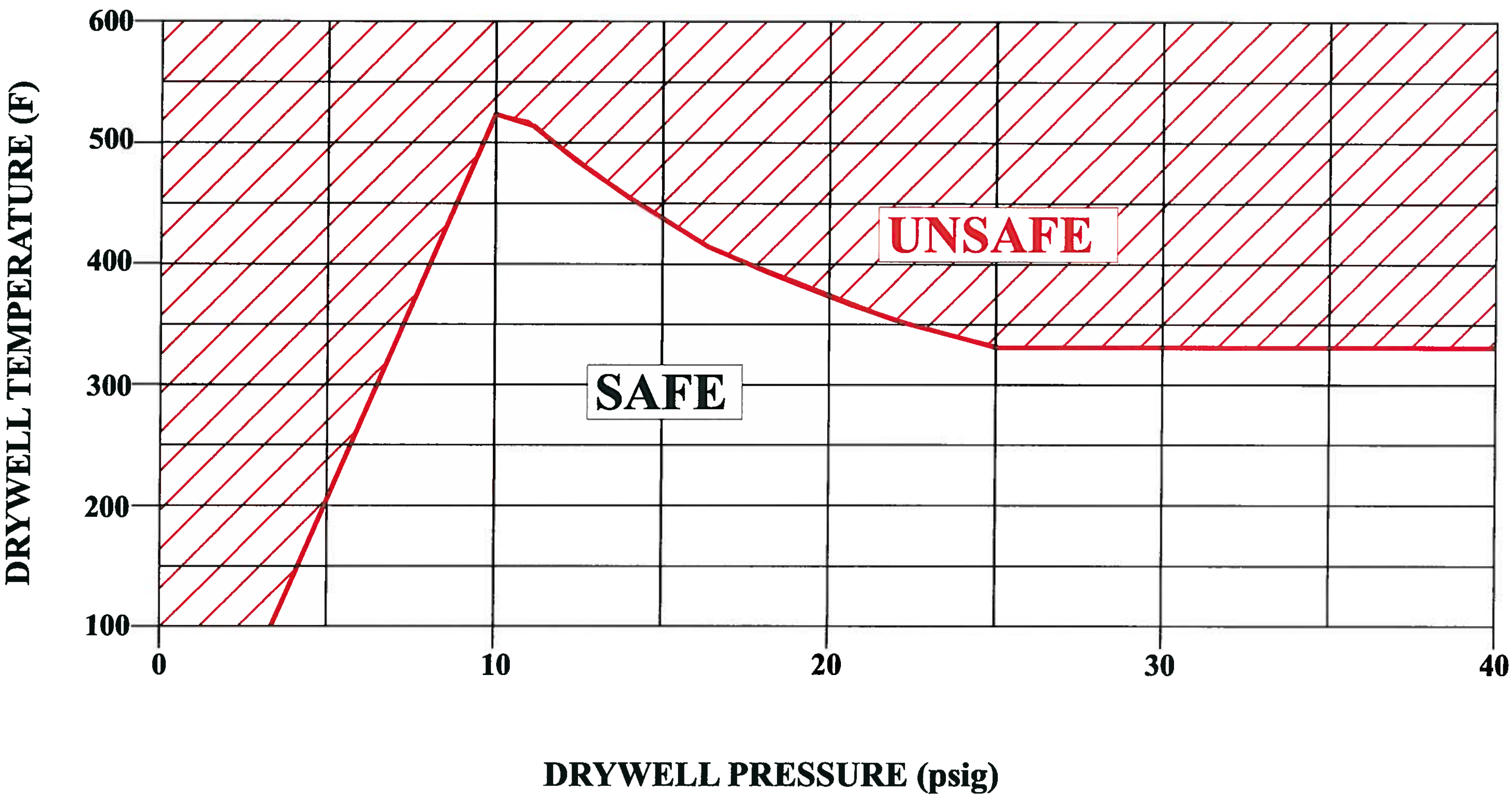
RCS Barrier (Pg. 39)

Loss	Potential Loss
1. Drywell Pressure (Pg. 39) Pressure greater than 1.85 PSIG	
2. Reactor Vessel Water Level (Pg. 39) Level less than -155 inches	
3. RCS Leak Rate (Pg. 39) Unisolable Main Steamline break as indicated by the failure of both MSIVs in any one line to close **AND** A. High MSL Flow **OR** B. High Steam Tunnel Temperature annunciators **OR** C. Turbine Building MSL leak annunciator **OR** D. Direct report of steam release	**3. RCS Leak Rate** (Pg. 39) RCS leakage GREATER THAN 50 gpm inside the drywell **OR** Unisolable primary system leakage outside drywell as indicated by Secondary Containment operating temperatures or radiation levels above Max. Normal Operating Values (SC - Secondary Containment Control Flowchart – Table 4 & Table 6)
4. Drywell Radiation Monitoring (Pg. 39) DWRRM greater than 138 R/hr	
5. Other Indications (Pg. 40) Drywell Post LOCA Monitor 4.71E+04 cpm	
6. Emergency Director Judgment (Pg. 40) Judgment by the ED that the RCS Barrier is lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier	**6. Emergency Director Judgment** (Pg. 40) Judgment by the ED that the RCS Barrier is potentially lost. Consider conditions not addressed and inability to determine the status of the RCS Barrier.

GRAPH 8

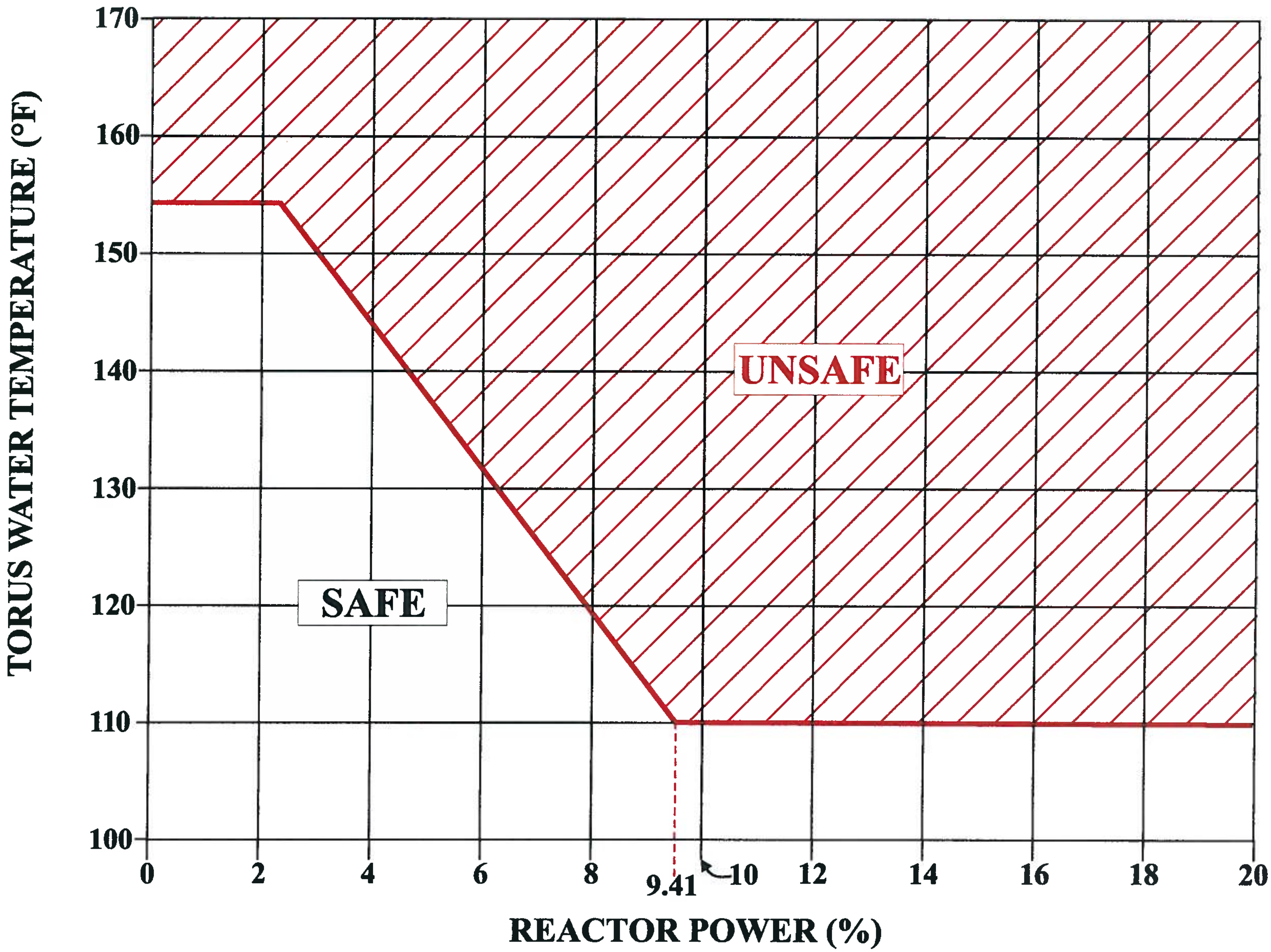
UNIT 1

DRYWELL SPRAY INITIATION LIMIT



NOTE: May use SPDS Emergency Displays in place of this Graph.

BORON INJECTION INITIATION TEMPERTURE



NOTE: May use SPDS Emergency displays in place of this Graph.

AC/DC POWER

SG1 - Prolonged Loss of All Offsite Power **AND** Prolonged Loss of All Onsite AC Power to Essential Busses (Pg. 44)

1. Loss of all AC power indicated by:
   a. Loss of power to or from Startup Auxiliary Transformers (SAT) 1/2C and 1/2D resulting in loss of all off-site electrical power to 4160 VAC Emergency Buses 1/2E, 1/2F, and 1/2G for greater than 15 minutes

   AND

   b. Failure of emergency diesel generators to supply power to emergency busses.

AND EITHER

2. Restoration of at least one 4160 VAC Emergency Bus, 1/2E, 1/2F, or 1/2G, within 4 hours of time of loss is **NOT** likely.

   OR

3. Fuel Clad Barrier Evaluation indicates continuing degradation (Loss or Potential Loss) of core cooling

<u>SS1</u> - Loss of All Offsite Power **<u>AND</u>** Loss of All Onsite AC Power to Essential Busses (Pg. 48)

1. Loss of all AC power indicated by:
   a. Loss of power to or from Startup Auxiliary Transformers (SAT) 1/2C and 1/2D resulting in loss of all off-site electrical power to 4160 VAC Emergency Buses 1/2E, 1/2F, and 1/2G for greater than 15 minutes

   <u>AND</u>

   b. Failure of diesel generators to supply power to emergency busses.

   <u>AND</u>

   c. Restoration of at least one 4160 VAC Emergency bus, 1/2E, 1/2F, or 1/2G, has **<u>NOT</u>** occurred within 15 minutes of time of loss of all AC power

<u>SS3</u> - Loss of All Vital DC Power (Pg. 51)

1. Loss of Vital DC power to 125/250 VDC Bus 1/2R22-S016 and 1/2R22-S017 indicated by bus voltage indications less than 105/210 VDC for greater than 15 minutes.

SA5 - AC power capability to Essential Busses reduced to a single power source for greater than 15 minutes such that any additional single failure would result in STATION BLACKOUT. (Pg. 57)

1. a. AC power capability to 4160 VAC Emergency Buses 1/2E, 1/2F, and 1/2G reduced to a single power source for greater than 15 minutes

   AND

   b. ANY additional single failure will result in station blackout.

76. 201006A2.06 001

Unit 2 is at 70% RTP.

- o Rod Worth Minimizer (RWM) is NOT in Sequence Control Mode
- o A ** group control rod at position 36 is selected

- o A malfunction in APRM "D" occurs
- o APRM "D" is currently reading 4%

- o NO operator action has been taken

With the above conditions, the mode of operation for RWM will be _________ Low Power Setpoint, (LPSP).

IAW with 31GO-OPS-006-0, "Conditions, Required Actions and Completion Times", for APRM "D", the Shift Supervisor and Shift Manager will __________ in the Required Action Sheet boxes below.

SS SIGN / TSA ACTIVE	SM SIGN

A. less than;
sign

B.✔ less than;
initial (signature is NOT allowed)

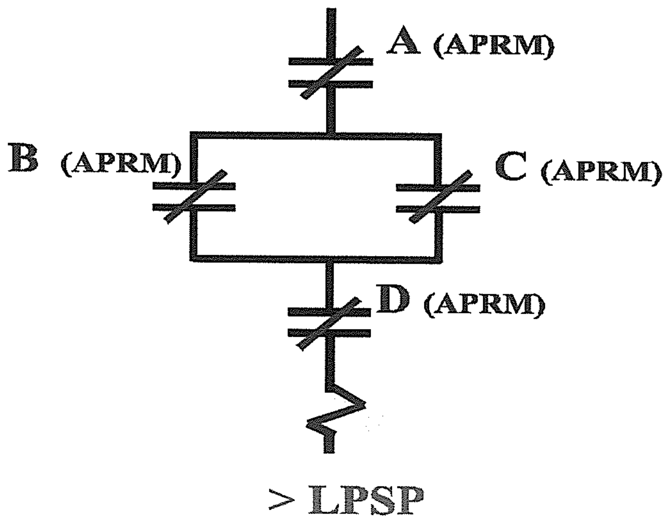
C. greater than;
sign

D. greater than;
initial (signature is NOT allowed)

Description:

The RWM enforces adherence to the Control Rod pull sequence during Startup, and Shutdown when reactor power is less than the Low Power Set Point (LPSP), 21% (20.6% U1) rated reactor power based on APRMs.

RWM uses either "A" or "**D**" or "B" and "C" for the value for LPSP and LPAP. For example on increasing power, RWM uses "A" and "**D**" AND "B" or "C" to accept greater than LPSP and LPAP. On decreasing power, "A" or "**D**" alone OR both "B" and "C" to accept less than LPSP and LPAP.



CONTACTS CLOSE WHEN ASSOCIATED APRM IS ABOVE THE LPSP

RELAY ENERGIZED - - - - - ABOVE LPSP

RELAY DE-ENERGIZED - - BELOW LPSP

With APRM "D" indicating 4%, RWM will determine the power level mode of operation to be <LPSP.

APRM "D" is inop at 70% power. IAW TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation, 3 of 4 APRMs shall be operable. This makes APRM "D" a "Tracking" RAS.

31GO-OPS-006-0, "Conditions, Required Actions and Completion Times" Section 7.1, "Initiation Of A Required Action Sheet " step 7.1.1.5 directs you to Section 7.3 for initiating a Required Action Sheet when a SSC is inoperable in a condition when it is not required to be operable. SS & SM signatures makes a RAS active.

Section 7.3 (Tracking RAS) directs the SS & SM to **initial** the appropriate boxes of the form OPS-1349. This will make the RAS a Tracking RAS.

The SRO must have detailed administrative procedure knowledge of 31GO-OPS-006-0, to answer this question. Completion of RAS administrative forms are above the RO knowledge level.

The "A" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses or does not know how to complete OPS-1349 and would be correct if 2 APRMs were inop. Also plausible since the blocks themselves indicate a signature is required.

The "C" distractor is plausible if the applicant confuses the APRM power arrangement inputs to RWM and would be correct if the "B" or "C" APRM was the one that failed. The second part is plausible if the applicant confuses or does not know how to complete OPS-1349 and would be correct if 2 APRMs were inop. Also plausible since the blocks themselves indicate a signature is required.

The "D" distractor is plausible if the applicant confuses the APRM power arrangement inputs to RWM and would be correct if the "B" or "C" APRM was the one that failed. The second part is correct.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

<u>K/A:</u>

201006 Rod Worth Minimizer System (RWM) (Plant Specific)

A2. Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWH) (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.06 Loss of reactor water level control input: P-Spec (Not-BWR6) 2.9 3.3

AFTER DISCUSSION WITH CHIEF EDWIN LEA ON 04/09/2013, WE WILL ATTEMPT TO WRITE A QUESTION ON THE INTENT OF THE K/A AND WILL USE APRMs TO MEET THE INTENT OF RWM INPUTS SINCE HATCH RWM DOES NOT USE RWLC INPUTS ANYMORE. DOING THIS WILL MEET THE INTENT OF THE K/A.

SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

LT-LP-30006, "LCO/RAS TRACKING" EO 300.027.A.03
C11-RWM-LP-05403, Rod Worth Minimizer, EO 001.010.A.18

References used to develop this question:

31GO-OPS-006-0, Conditions, Required Actions and Completion Times, **Ver. 8.0**
U2 TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation, **Amendment 154**

Item 1: SRO ONLY Guideline
Item 2: 31GO-OPS-006-0, pages 9, 10 & 13, **Ver. 7.0**
Item 3: U2 TS 3.3.1.1, pages 406-408 & 412, **Amendment 154**

Modified from HLT-5 NRC Exam Q#97

ORIGINAL QUESTION (HLT-5 NRC Exam Q#97)

Unit 2 is operating at 100% power when it is discovered that the "A" ADS valve, 2B21-F013A, is inop for its ADS function.

IAW with 31GO-OPS-006-0, "Conditions, Required Actions and Completion Times", which ONE of the following completes following statement.

A (An) ______ Required Action Sheet is required.

The Shift Supervisor and Shift Manager must ______ in the appropriate boxes below.

SS SIGN / TSA ACTIVE	SOS SIGN

A.✓ Tracking;
ONLY initial (signature is NOT allowed)

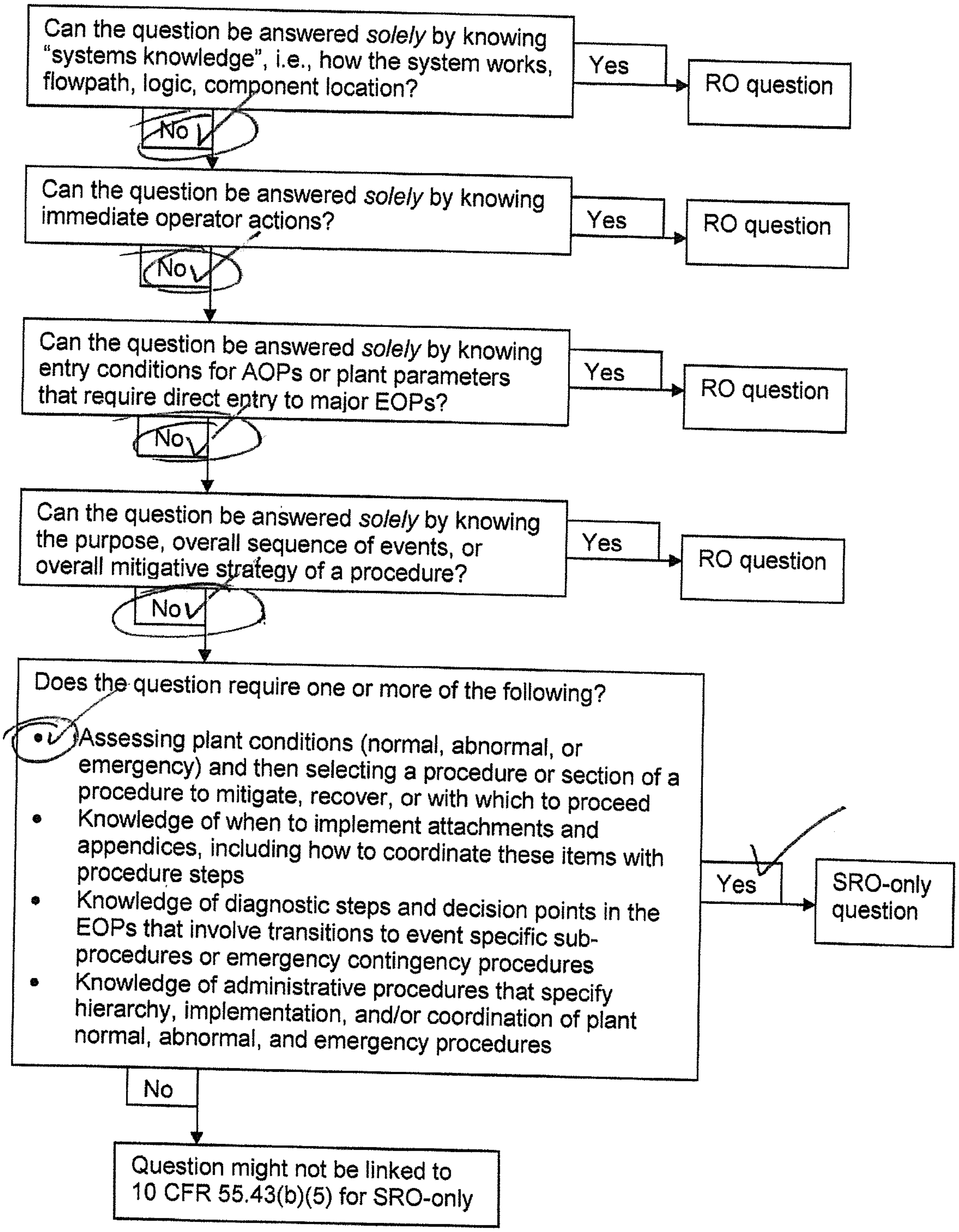
B. Active;
sign

C. Active;
ONLY initial (signature is NOT allowed)

D. Tracking;
sign

Q#76 K/A 201006A2.06

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



7.1.1.8 IF applicable (e.g., TS LCO 3.1.7, TS LCO 3.8.1), ENTER the MODIFIED COMPLETION TIME/DATE in Section 2.

7.1.1.8.1 This time is the maximum allowable time that is allowed for failing to meet a REFERENCE DOCUMENT CONDITION / REQUIRED ACTION (e.g., TS LCO 3.1.7, REQUIRED ACTION A.1 has a MODIFIED COMPLETION TIME/DATE of 10 days for SLC).

7.1.1.8.2 This time is the same for subsequent INOPERABLE equipment which occurs while related TS LCO INOPERABLE equipment exists (e.g., SBLC pump A is INOPERABLE on day 1 and, then, on day 2 the SLC Tank fails to meet a requirement. The MODIFIED COMPLETION TIME for the tank and any subsequent entries would be the same as for SBLC pump A OR 10 days from the day 1 entry into the TS LCO REQUIRED ACTIONS.).

7.1.1.9 Indicate the applicability of the Safety Function Determination Program in the SFDP ENTERED block of Section 2.

7.1.1.10 Check appropriate box in the INOP STATUS INDIC LIT block of Section 2 for the inoperable equipment status indicators.

7.1.1.11 Post signs on equipment required to be protected per NMP-OS-010 and check the appropriate box in Section 2.

7.1.1.12 ENTER CONDITIONS, plant modes AND/OR applicability of the equipment OR CONDITION in the APPLICABILITY block of Section 2.

7.1.1.13 RECORD the REFERENCE DOCUMENT in the block of the same name of Section 2. (e.g., Unit 1 TS 3.6.1).

7.1.1.14 RECORD the REFERENCE DOCUMENT revision or amendment number in the REVISION/AMENDMENT block of Section 2.

7.1.1.15 ENTER the REQUIRED ACTION in the REQ. ACTION IF COMP TIME IS EXCEEDED block of Section 2, using the following:

7.1.1.15.1 ENTER the REQUIRED ACTION Number as shown in the REFERENCE DOCUMENT block of Section 2 (e.g., TS LCO 3.6.1.1, REQUIRED ACTION D.1). THEN, fill in the REFERENCE DOCUMENT, REQUIRED ACTION, REQ. COMP TIME OR FREQ., AND SEQ NO. blocks of Section 4.

AND/OR

7.1.1.15.2 ENTER the ACTION required (e.g., Be in Mode 3 in 12 hours and in Mode 4 in 36 hours.)

7.1.1.16 SS signs SS SIGN/TSA ACTIVE block of Section 2 of the RAS, signifying the sheet is active.

7.1.1.17 SM reviews and signs SM SIGN block of Section 2, indicating concurrence with the RAS.

7.1.1.18 As equipment is returned to OPERABLE status, ENTER the time/date in the RETURN TO OPER TIME/DATE block AND initial in the INIT block of Section 1.

7.1.2 PERFORM the following in the ≤ 1 HOUR ACTIONS section (section 3) of the RAS:

7.1.2.1 ENTER the REFERENCE DOCUMENT and section number in the appropriate block (e.g., TS LCO 3.4.1).

7.1.2.2 ENTER the REQUIRED ACTION* as specified above (e.g., REQUIRED ACTION C.1: ENTER 3.0.3) using separate lines for each ACTION containing multiple parts. IF the REFERENCE DOCUMENT allows a choice of REQUIRED ACTIONS, list only the one chosen.

7.1.2.2.1 List the Administrative Control Document (APC, Tagout, Rep Task, or other) in the table at the bottom of form OPS-1349.

7.1.2.2.2 Reference the RAS on the Administrative Control Document, (APC, Tagout, Rep Task, or other), and initial in the table at the bottom of form OPS-1349.

7.1.2.2.3 IF using the ESOMs program for administrative control, ensure that the SS or TDO is signed on as a Document Holder, OR has locked the tagout denoting that the tagged equipment is covered by a REQUIRED ACTION, and initial in the table at the bottom of form OPS-1349.

7.1.2.2.4 IF tracking items per NMP-AD-012, ensure that compensatory actions are established to ensure that the requirements of the IDO are met.

7.1.2.3 ENTER the REQ COMP TIME block.

7.1.2.4 ENTER the Time and Date the ACTION is completed in the PERFORMED TIME/DATE block and initial in the INIT block WHEN the ACTION is completed.

7.1.3 PERFORM the following in the > 1 HOUR ACTIONS section (section 4):

7.1.3.1 ENTER the REFERENCE DOCUMENT and section number in the REFERENCE DOCUMENT block (e.g., TS 3.1.7, REQUIRED ACTION A.1).

7.1.3.2 ENTER the REQUIRED ACTION specified above in the REQUIRED ACTION block using the following guidelines:

7.1.3.2.1 WHEN more than one ACTION is required, use separate lines for each ACTION.

7.3 INITIATING REQUIRED ACTION SHEET ON SSCs NOT REQUIRED TO BE OPERABLE DUE TO EXISTING PLANT CONDITIONS

INFORMATION

7.3.1 WHEN the equipment OR CONDITION does NOT apply due to current plant / equipment conditions (Conditional Tracking Actions), perform the following:

7.3.1.1 Check appropriate box in INOP STATUS INDIC LIT block in Section 2 of OPS-1349 for the inoperable equipment status indicators.

7.3.1.2 Post signs on equipment required to be protected per NMP-OS-010 and check the appropriate box in Section 2.

7.3.1.3 ENTER CONDITIONS, plant modes AND/OR applicability of the equipment OR CONDITION in the APPLICABILITY block in Section 2 of OPS-1349.

7.3.1.4 RECORD the REFERENCE DOCUMENT in the REFERENCE DOCUMENT block in Section 2 of OPS-1349. (e.g., Unit 1 TS LCO 3.6.1, REQUIRED ACTION A.1)

7.3.1.5 RECORD the REFERENCE DOCUMENT revision or amendment number in REVISION/AMENDMENT block in Section 2 of OPS-1349.

7.3.1.6 SS INITIALS the SS SIGN/TSA ACTIVE block.
The SS SIGNS this block WHEN the RAS becomes active.

7.3.1.7 SM INITIALS the SM SIGN block.
The SM SIGNS this block WHEN the RAS becomes active.

7.3.1.8 Complete the remainder of the REQUIRED ACTION TRACKING SHEET (per 7.1.1) WHEN the equipment becomes required to be OPERABLE by plant / equipment conditions.
The Initiation Time and Date is WHEN plant / equipment conditions are such that the equipment is required to be OPERABLE.

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.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 ------------NOTE----------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, and 2.f. ------------------------------- Place associated trip system in trip.	12 hours
B. --------------NOTE-------------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, and 2.f. ------------------------------------ One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip.	6 hours
	<u>OR</u>	
	B.2 Place one trip system in trip.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 27.6% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations.	12 hours
	AND	
	I.2 Restore required channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Be in MODE 2.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program

(continued)

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 Monitor					
a. Neutron Flux - High	2	2(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	2(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	2(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	2(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux - High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 0.57W + 56.8% RTP and ≤ 115.5% RTP(b)
c. Neutron Flux - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1, 2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.57W + 56.8% - 0.57 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

77. 209001G2.4.49 001

Unit 1 was operating at 100% RTP when a transient resulted in the following:

Date 4/10/2013 Time 12:00
- o Reactor scram, all rods fully insert
- o Drywell pressure: 0.7 psig
- o Reactor water level: -102 inches, decreasing
- o Reactor pressure: 415 psig
- o Both Core Spray (CS) systems are in standby due to a failure of ALL Core Spray Instrumentation Initation logic

Based on the current above plant conditions:

As a MINIMUM, the operator will start both CS pumps ______________ .

IAW TECH SPEC, the EARLIEST listed time that REQUIRES Unit 1 to be in mode 4 is 4/12/2013 at __________ .

Reference Provided

A. , but will NOT open their respective discharge valves;

02:00

B. , but will NOT open their respective discharge valves;

03:00

C.✓ and WILL open their respective discharge valves;

02:00

D. and WILL open their respective discharge valves;

03:00

Description:
31GO-OPS-021
5.2.9
The Nuclear Plant Operators (NPOs) have the responsibility to manually align, start, or initiate any automatically actuated system, equipment, signal, or function that has indication of a start failure or incomplete initiation so that it will perform its intended function unless operation would create a condition that would not mitigate a transient.
7.7.3
Transient Acts are those actions that can be performed by Plant Operators during a transient

the procedure, as soon as practical, and review it to ensure all necessary steps were performed.
Transient acts include
Manual operation of RWL control / injection systems

Core Spray pumps should have automatically started at -101 inches, but failed to do so. 1E21- F005 should automatically open when reactor pressure lowers to < 449 psig. 1E21-F004 is normally open in a standby lineup, so the action of manually opening the valve is not required. Reactor Pressure is < 449 psig allowing 1E21- F005 to be being manually opened. 1E21-F004 valve is already opened. Starting the pump(s) and opening 1E21-F005 is the MINIMUM action required.

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

Core Spray System	modes	channels required	required action
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2, 3,	4	B

Condition "B": Declare supported feature(s) inoperable. 1 hour from discovery of loss of initiation capability for feature(s) in both divisions. Includes 1A,1B,1C EDGs and PSW TB isolation valves 1P41-F310 A/B/C/D.

3.5.1 ECCS - Operating

Condition	required action	completion time
IF. Two or more low pressure ECCS injection/spray subsystems inoperable.	Enter LCO 3.0.3	Immediately

LCO 3.0.3: When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be *initiated within 1 hour* to place the unit, as applicable, in:

a. MODE 2 within 7 hours;
b. MODE 3 within 13 hours; and
c. MODE 4 within 37 hours.

When the SRO determines there is a loss of 2 low pressure ECCS system and is directed into LCO 3.0.3 the SRO must apply the requirements of LCO 3.0.3.

The "A" distractor is plausible if the Unit 1 TS limit of 390 psig was the set point for the CS pump discharge valve auto open permissive circuit. The second half is correct.

The "B" distractor is plausible if the Unit 1 TS limit of 390 psig was the set point for the CS pump discharge valve auto open permissive circuit. The second half is plausible if during the application of LCO 3.0.3 the "*Action shall be initiated within 1 hour*" requirement is added to the MODE 4 within 37 hours.

The "D" distractor is plausible because the first half is correct. The second half is plausible if added to the MODE 4 within 37 hours.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation without SURVEILLANCE REQUIREMENTS

and

3.5.1 Emergency Core Cooling System (ECCS) without SURVEILLANCE REQUIREMENTS

K/A:

209001 Low Pressure Core Spray System

2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
(CFR: 41.10 / 43.2 / 45.6) 4.6 4.4

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the TS and their bases.

> **Application of required actions (section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (section 1).**
>
> **Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.3. thru 3.0.7; SR 4.0.1 thru 4.0.4) .**

LESSON PLAN/OBJECTIVE:
E21-CS-LP-00801, **Ver. 5.0**/EO 300.010.A.25

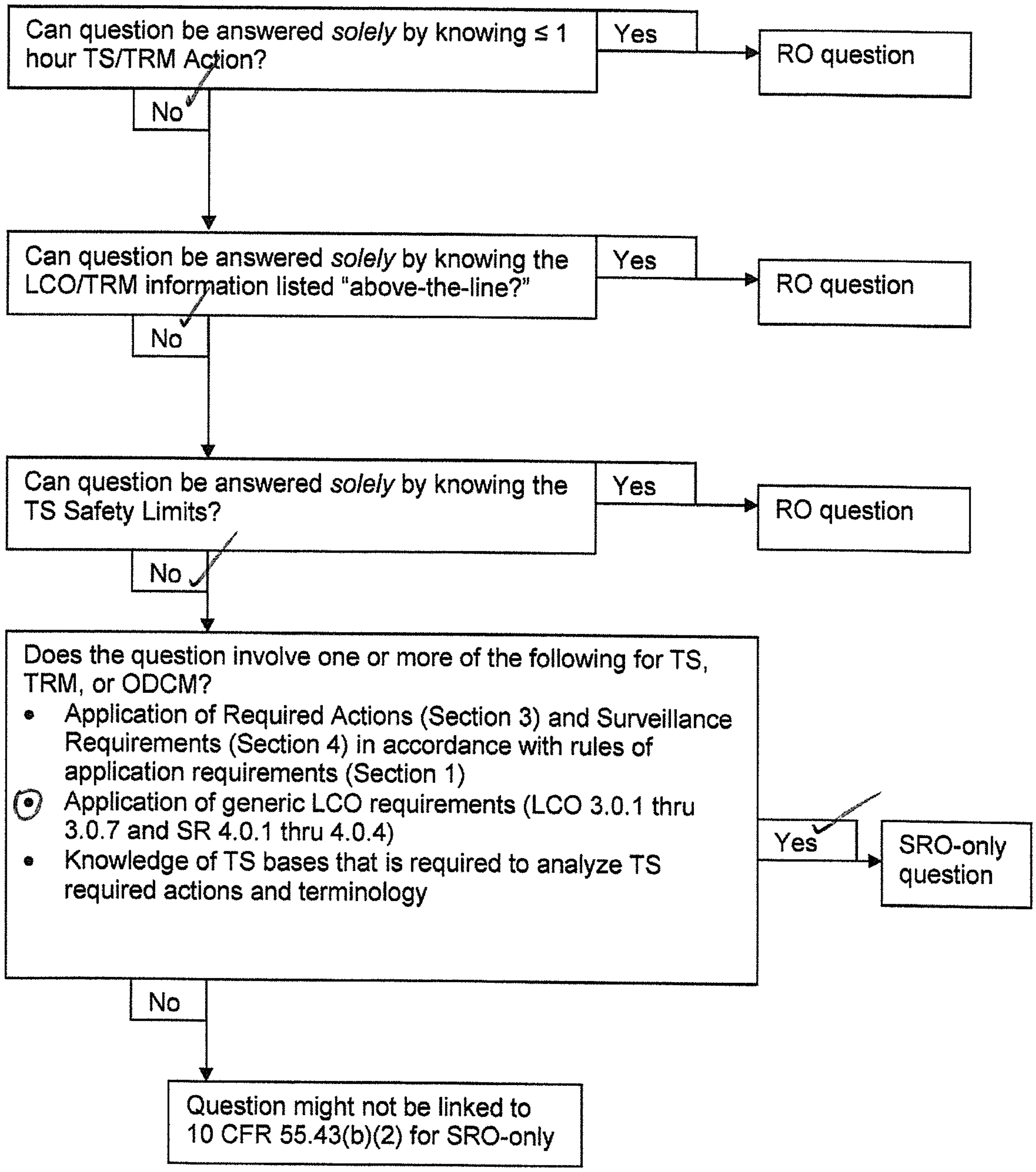
References used to develop this question:

31GO-OPS-021-0, Manipulation Of Controls And Equipment, **Ver. 4.1**
U1 TS 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation
U1 TS 3.5.1 ECCS - Operating

Item 1: SRO ONLY Guideline
Item 2: 31GO-OPS-021-0, pages 5, 13 & 14, **Ver. 4.1**
Item 3: U1 TS 3.3.5.1 & TS 3.5.1, pages 104-107, 109, 167-169 **Amend. 204**

Q#77 K/A 209001 G2.4.49

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



5.2.7 If a component is found mispositioned, or has been unintentionally mispositioned (i.e. bumped, manipulation of the wrong component), contact the appropriate unit Shift Supervisor immediately. Do NOT re-position the component to the correct/intended position unless there is a compelling safety reason (nuclear or personnel) to do so.
If the mispositioned component is presenting a safety concern, person(s) involved are allowed to reposition only the affected component to eliminate the unsafe condition prior to contacting the appropriate unit Shift Supervisor.

5.2.8 Spare electrical breakers shall be kept in the racked in position and turned OFF.

5.2.9 The Nuclear Plant Operators (NPOs) have the responsibility to manually align, start, or initiate any automatically actuated system, equipment, signal, or function that has indication of a start failure or incomplete initiation so that it will perform its intended function unless operation would create a condition that would not mitigate a transient.

6.0 PREREQUISITES

N/A - Not applicable to this procedure

7.7 SIMPLE QUICK ACTS / TRANSIENT ACTS

INFORMATION

7.7.1 Plant systems are normally operated by system operating procedures, surveillance procedures, abnormal operating procedures, and emergency operating procedures. Under certain situations, Plant Operators may perform actions in the Main Control Room without reference to a procedure.
These "skill of the craft" actions will be defined as Simple Quick Acts or Transient Acts and listed in Attachment 1.

7.7.2 Simple Quick Acts are those actions that may be performed by trained, qualified individuals without a procedure provided that the task is simple, short, and routine.

7.7.3 Transient Acts are those actions that can be performed by Plant Operators during a transient without immediate procedure reference. The individual taking the action will obtain a copy of the procedure, as soon as practical, and review it to ensure all necessary steps were performed.

7.7.4 Although no procedure guidance is required for a Simple Quick Act, proper conservative decision making and consideration of the consequences of the manipulations shall be applied.

1.0 TRANSIENT ACTS

- Manual Scram Initiation
- Rapid Recirc flow reductions
- Complete a failed Recirc runback
- Tripping the Main Turbine
- Tripping RFPTs
- Tripping pumps/motors as directed by the SS
- Manually inhibiting ADS
- Manually operating SRVs
- Making adjustments to process controllers to maintain a process parameter.
- Manual operation of RWL control / injection systems
- Reset of Group II logic (Isolation must be at least visually verified)

2.0 SIMPLE QUICK ACTS

The following are actions for which improper performance does not have significant consequences:

- Acknowledge/reset/test of annunciators
- Bypass of a Nuclear Instruments (SRMs, IRMs, and APRMs) when directed by ARPs, Surveillances, or Other Procedures
- Selection and Driving of Nuclear Instruments (SRMs and IRMs) IN and OUT of the Core when directed by Startup and Shutdown Procedures
- Selection of a Peripheral Rod when using Core Flow for Power Reductions per SS approval
- Ackowledgement of “ROD OUT BLOCK” during Power Maneuvers if previously addressed on current shift
- Operation of plant communication systems
- Operation of plant computer systems to monitor plant parameters
- Operation of selector switches/pushbuttons to monitor plant parameters
- Changing chart paper or pens
- Changing light bulbs
- Blowdown of moisture from air receivers
- Operation of fire fighting equipment
- Swapping N2 bottles for HCU charging operations
- Starting and stopping sump pumps / transformer fans
- Advancing HVAC roll filters
- Matching switch position to the equipment condition
(ex; "red flagging" the RHR A pump control switch following an auto pump start)

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1 ---------NOTES------------ 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. -------------------------------- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	AND	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	-----------NOTE------------ Only applicable for Functions 3.a and 3.b. ------------------------------	
		Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	B.3	Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	-----------NOTES---------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 2.c, 2.d, and 2.f. ------------------------------	
		Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for feature(s) in both divisions
	<u>AND</u>		
	C.2	Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	------------NOTE------------ Only applicable if HPCI pump suction is not aligned to the suppression pool. ------------------------------- Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	<u>AND</u>		
	D.2.1	Place channel in trip.	24 hours
	<u>OR</u>		
	D.2.2	Align the HPCI pump suction to the suppression pool.	24 hours
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	-----------NOTES---------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.d and 2.g. ------------------------------- Declare supported feature(s) inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
	<u>AND</u>		
	E.2	Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	F.1	Declare Automatic Depressurization System (ADS) valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
	F.2	Place channel in trip.	96 hours from discovery of inoperable channel concurrent with HPCI or reactor core isolation cooling (RCIC) inoperable <u>AND</u> 8 days
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1	Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	<u>AND</u>		
	G.2	Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable <u>AND</u> 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1	Declare associated supported feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 5)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2, 3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -113 inches
b. Drywell Pressure - High	1, 2, 3	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.92 psig
c. Reactor Steam Dome Pressure - Low (Injection Permissive)	1, 2, 3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 390 psig and ≤ 476 psig
d. Core Spray Pump Discharge Flow - Low (Bypass)	1, 2, 3, 4(a), 5(a)	1 per subsystem	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 610 gpm and ≤ 825 gpm
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level - Low Low Low, Level 1	1, 2, 3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -113 inches
b. Drywell Pressure - High	1, 2, 3	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.92 psig

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Also required to initiate the associated diesel generator (DG) and isolate the associated plant service water (PSW) turbine building (T/B) isolation valves.

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 of seven 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 ≤ 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.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. AND C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days

(continued)

ACTIONS (continued)

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

78. 215003A2.05 001

Unit 2 Reactor startup is in progress.

The following IRM readings have been observed while switching up from range 6 to range 7:

IRM Channel	Range 6	Range 7
A	75	8.0
B	60	0.0
C	75	8.0
D	70	6.0
E	70	8.5
F	50	5.0
G	65	9.0
H	70	8.5

IAW 34GO-OPS-001-2, "Plant Startup", and with the above IRM data,

Acceptable overlap is confirmed on _________ IRMs.

The LOWEST level of authority required to continue the reactor startup is the _________ .

A.✔ six (6);
Shift Supervisor

B. six (6);
Shift Manager

C. seven (7);
Shift Supervisor

D. seven (7);
Shift Manager

Description:
34GO-OPS-011-2

7.2.24

Confirm there is overlap between IRM ranges 6 AND 7, by completing Attachment 9.

NOTE:

IRM range 6 to range 7 overlap criteria is obtained for calibration purposes only and is not used to determine operability of the IRMs. Operability of an IRM is demonstrated by the indication being on-scale and tracking power changes. During the transition from range 6 to range 7, operability of the IRM is only considered to be lost if the IRM indication goes off-scale or ceases to track power changes.

(The range 7 IRM reading must be $\pm$ 2 of the range 6 IRM reading divided by10)

7.2.26

To continue power ascension with any inoperable IRMs, obtain Shift Supervisor approval.

An SRO must have detailed administrative procedure knowledge of 34GO-OPS-001-2, to answer this question.

The "B" distractor is plausible because the first half is correct. The second half is plausible because the Shift Manger's permission is required to continue operation just above the point of criticality if conditions prohibit the withdrawl of control rods.

The "C" distractor is plausible if only "B" IRM reading zero (0) is recognized as failing the overlap with a differential of 6 and the "G" IRM reading nine (9) with a differential of 2.5 is not recognized. Second half is correct.

The "D" distractor is plausible if only "B" IRM reading zero (0) is recognized as failing the overlap with a differential of 6 and the "G" IRM reading nine (9) with a differential of 2.5 is not recognized. The second half is plausible because the Shift Manger's permission is required to continue operation just above the point of criticality if conditions prohibit the withdrawl of control rods.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

215003 Intermediate Range Monitor (IRM) System

A2. 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: (CFR: 41.5 / 45.6)

A2.05 Faulty or erratic operation of detectors/system 3.3 3.5

SRO only because of link to 10CFR55.43(b)(5):Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. 1) assessing plant conditions (normal, abnormal, or emergency) and then 2) selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed. One area of SRO level knowledge (with respect to selecting a procedure) is knowledge of the content of the procedure versus knowledge of the procedure's overall mitigative strategy or purpose.

LESSON PLAN/OBJECTIVE:
C51-IRM-LP-01202/EO 012.010.A.01

References used to develop this question:
34GO-OPS-011-2, Plant Startup, **Ver. 42.3**

Item 1: SRO ONLY Guideline
Item 2: 34GO-OPS-011-2, pages 26 & 73, **Ver. 42.3**

ORIGINAL QUESTION (LT-012010 003 HLT Bank)

A Unit 2 Reactor startup is in progress. The following IRM readings have been observed while switching up from range 6 to range 7:

IRM Channel	Range 6	Range 7
-----------	-------	----------
A	75	8.0 and slowly rising
B	80	0.0 and steady
C	75	8.0 and slowly rising
D	90	9.0 and slowly rising
E	90	10.0 and slowly rising
F	80	0.0 and steady
G	75	9.0 and slowly rising

H	80	8.5 and slowly rising

IAW 34GO-OPS-001-2, "Plant Startup", which ONE of the choices below completes BOTH of the following statements?

Based on the above IRM data, acceptable overlap on ALL IRMs _________ confirmed.
The minimum approval required to continue the reactor startup is the Shift _________ .

A. is;
Supervisor

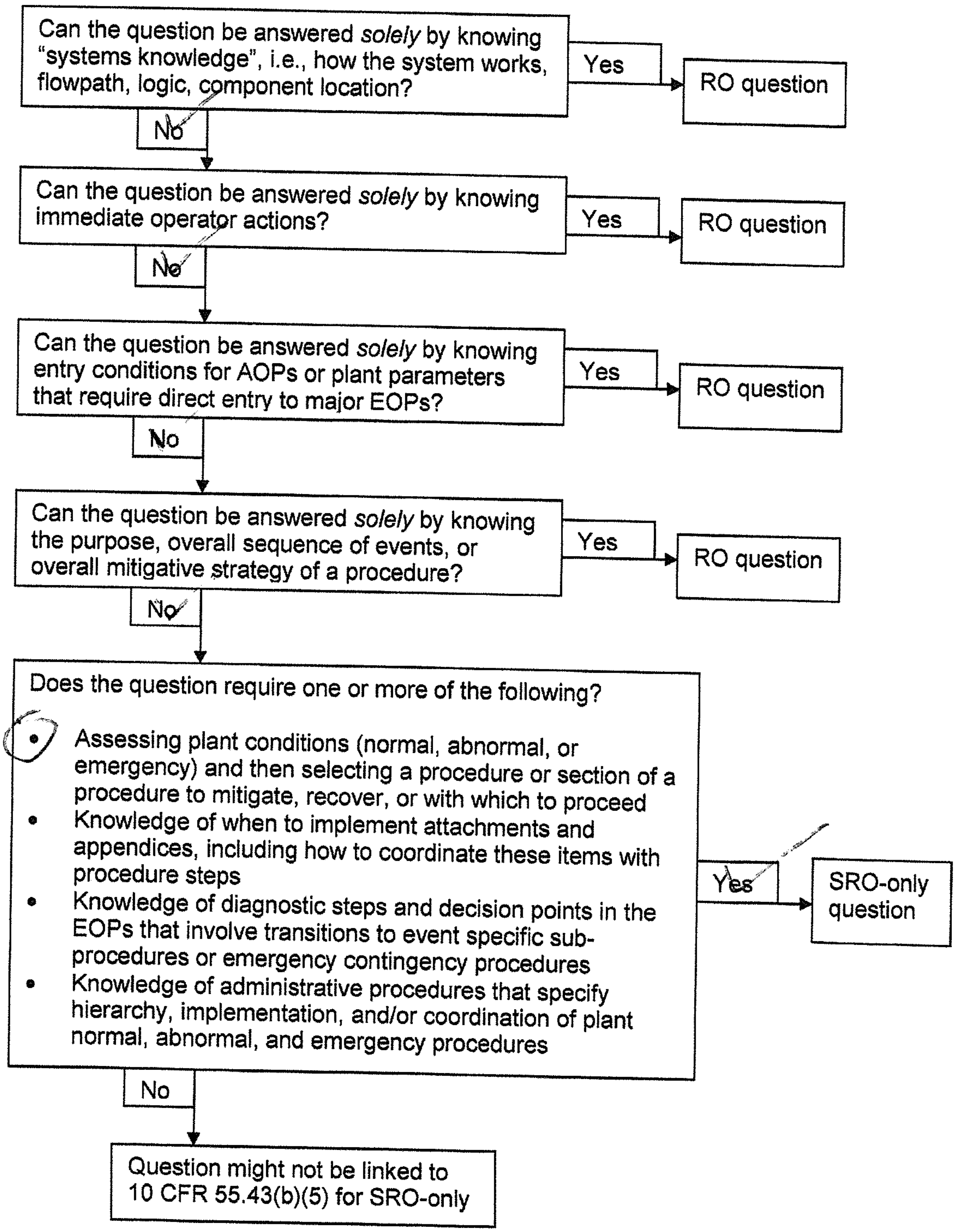
B. is ;
Manager

C.✓is NOT;
Supervisor

D. is NOT;
Manager

Q#78 K/A 215003 A2.05

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



CAUTION:	WHEN POSITIONING THE IRM RANGE SWITCHES, CARE MUST BE EXERCISED TO PREVENT A REACTOR SCRAM FROM OCCURRING. IF POSSIBLE, RANGE IRMS IN ONLY ONE RPS CHANNEL AT A TIME.

Critical

7.2.22 As reactor power increases,
range up the IRM Range Switches to maintain IRM indication on recorders between 10 AND 80 on the 0-125 scale. ____

7.2.23 WHEN all operable IRM channels are above range 3 AND
PRIOR to reaching range 7,
fully withdraw all operable SRM detectors. ____

7.2.24 **Confirm** there is overlap between IRM ranges 6 AND 7,
by completing Attachment 9. ____

NOTE:	IRM range 6 to range 7 overlap criteria is obtained for calibration purposes only and is not used to determine operability of the IRMs. Operability of an IRM is demonstrated by the indication being on-scale and tracking power changes. During the transition from range 6 to range 7, operability of the IRM is only considered to be lost if the IRM indication goes off-scale or ceases to track power changes.

7.2.25 **Record** any unacceptable IRM overlaps in the Control Room log. ____

7.2.25.1 **Notify** I&C Shop to adjust IRM preamplifiers. ____

7.2.26 To continue power ascension with any inoperable IRMs,
obtain Shift Supervisor approval. ____

CONTINUOUS

1.0 **Confirm** the overlap between IRM ranges 6 AND 7 is acceptable as follows:

1.1 **Record** readings from range 6 for each IRM channel.

1.2 **Record** readings from range 7 for each IRM channel.

1.3 **Divide** Range 6 readings (COLUMN 2) by 10 AND **enter** in Column 4.

COLUMN 1	COLUMN 2	COLUMN 3	COLUMN 4	COLUMN 5	
				SIGN-OFF	
IRM CHANNEL	RANGE 6 READING (Black Scale)	RANGE 7 READING (Red Scale)	(COLUMN 2) / 10	INITIALS	VERIFIED (LIC OPER)
A					
B					
C					
D					
E					
F					
G					
H					

1.4 **Confirm** that Column 3 = Column 4 ± 2 (on the red scale). _____

1.5 **Initial** AND **verify** the calculations. _____

VERIFY _____

NOTE:	One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second Licensed Operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

This question was
one of the five SRO
questions previously
submitted for review

SRO question
1 of 5

Question change required
Changes were made

79. 218000A2.05 001

Unit 2 was operating at 100% RTP when a transient resulted in the following:

- o 125 VDC 2A, 2R25-S001, deenergizes
- o 125 VDC 2B, 2R25-S002, deenergizes

- o MSIVs are closed
- o Reactor Pressure is 1080 psig and slowly increasing
- o Drywell Pressure is 2.0 psig and slowly increasing
- o Torus Water Level is 195 inches

If the control switch for 2B21-F013M, ADS valve, is placed to the open position, the 2B21-F013M _________ open.

With the above plant conditions IAW EOPs, Reactor Pressure will be reduced IAW __________ .

A. will;
31EO-EOP-108-2, Alternate RPV Depressurization

B. will;
31EO-EOP-107-2, Alternate RPV Pressure Control

C.✓ will NOT;
31EO-EOP-108-2, Alternate RPV Depressurization

D. will NOT;
31EO-EOP-107-2, Alternate RPV Pressure Control

Description:

Edwin, this was question 1 of 5 of the SRO questions that you have already reviewed. Any discussed changes have been incorporated.

The control logic is powered from the station service batteries through 125VDC bus 2A (2R25-S001) and 2B (2R25-S002). The "A" and "B" logic is normally powered from the 125VDC 2A bus. Only the "B" logic is alternately powered from 125VDC 2B bus upon failure of the 2A bus. With both 125VDC A & B lost, ADS SRV "M" will not have control power to energize its solenoid and will not open.

With the MSIVs closed, 31EO-EOP-107-2, Alternate RPV Pressure Control, will be entered to control reactor pressure. Reactor pressure control will be in this procedure until plant conditions change requiring an emergency depress due to Torus level >193 inches. Once the emergency depress is required, reactor pressure control will be transferred to CP-1 Pont G of

Depressurization, will be required to be entered to control reactor pressure.

The SRO must remember the RC/P leg and determine that reactor pressure will be controlled by 31EO-EOP-107 since the MSIVs are closed and remember that once an emergency depress is required, CP-1 Point G is entered to control reactor pressure. After transitioning to CP-1 Point G, the SRO must realize less than 5 SRVs will be open and then transition to 31EO-EOP-108 to control reactor pressure. This is above the overall mitigating strategy of the RC & CP-1 EOP flowcharts and requires SRO knowledge to answer this question.

The "A" distractor is plausible if the applicant confuses/does not remember the power supplies for the ADS valves and thinks "M" ADS valve has power to open. The second part is correct.

The "B" distractor is plausible if the applicant confuses/does not remember the power supplies for the ADS valves and thinks "M" ADS valve has power to open. The second part is plausible if the applicant confuses or does not recognize that CP-1 Point G is required, therefore since EOP-107 is already is in progress, the SRO will stay in this procedure to control reactor pressure.

The "D" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses or does not recognize that CP-1 Point G is required, therefore since EOP-107 is already is in progress, the SRO will stay in this procedure to control reactor pressure.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

218000 Automatic Depressurization System

A2. 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:
(CFR: 41.5 / 45.6)

A2.05 Loss of A.C. or D.C. power to ADS valves 3.4 3.6

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

B21-ADS-LP-03801, Automatic Depressurization System (ADS), **Ver. 4.0**/EO 038.001.A.02

References used to develop this question:

34SO-B21-001-2, Automatic Depressurization (ADS) And Low-Low Set (LLS) Systems, **Ver. 13.13**
31EO-EOP-010-2, RC (NON-ATWS), **Ver. 9**
31EO-EOP-015-1, CP-1, **Ver. 8**
Item 1: SRO ONLY Guideline
Item 2: 34SO-B21-001-2, page 24, **Ver. 13.13**
Item 3: 31EO-EOP-010-2, RC (NON-ATWS), **Ver. 9** & 31EO-EOP-015-1, CP-1, **Ver. 8**

Modified from HLT Database Q#218000K2.01-002

ORIGINAL QUESTION (Q#218000K2.01-002)

Given the following plant conditions:

Unit 2 is at rated power

A loss of 125 VDC 2A, 2R25-S001, has occurred

Which ONE of the following describes the effect on the Unit 2 Automatic Depressurization System (ADS) valves and ADS logic?

The NORMAL power supply to the __________ has been lost.

__________ Initiation Logic will be supplied ALTERNATE power.

A. "A" Initiation Logic ONLY
Both divisions of

B. "A" Initiation Logic ONLY

Only one division of

C. "A"& "B" Initiation Logic
Both divisions of

D.✓ "A"& "B" Initiation Logic
Only one division of

Q#79 K/A 218000A2.05

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

CONTROL SWITCH / BREAKER NUMBER	DESCRIPTION	NORMAL POSITION	CHECKED	VERIFIED
2R25-S001 (130TDT13)				
Brkr. 26	Auto Depressurization System "A" Normal Cntl Pwr (2B21C Sys)	CLOSED		
Brkr. 30	Remote Shutdown Panel 2C82-P001	CLOSED		
2R25-S002 (130TDT13)				
Brkr. 22	Core Spray B Relay & ADS "B" Logic (2E21 & 2B21 System)	CLOSED		
Brkr. 29	ATTS ECCS Cabinet 2H11-P928 Power Supply	CLOSED		
Brkr. 34	Remote Shutdown Panel 2C82-P001 (2B21 & 2E11)	CLOSED		
2R25-S065 120/208V DIST. CAB. 2C INSTRUMENT BUS 2B (130TGT12)				
Brkr. 6	Temp Recorders For Safety/Blowdown Valves 2B21 System (2B21-R614)	CLOSED		
2R25-S129 (130TET13)				
Brkr. 5	ATTS ECCS Cabinet 2H11-P927 Power Supply 2E21-K401C	CLOSED		
Brkr. 7	ATTS ECCS Cabinet 2H11-P927 Power Supply 2E21-K402C	CLOSED		
2R25-S130 (130TDT13)				
Brkr. 5	ATTS ECCS Cabinet 2H11-P928 Power Supply 2E21-K402D	CLOSED		

31EO-EOP-010-2, RC RPV RC/P Path

WHILE PERFORMING THE FOLLOWING	
IF torus water temperature CANNOT be maintained below the Heat Capacity Temperature Limit (Graph 2)	THEN maintain reactor pressure below the limit, irrespective of the resulting cooldown rate.
IF torus water level CANNOT be maintained below the SRV Tail Pipe Level Limit (Graph 6)	THEN maintain reactor pressure below the limit, irrespective of the resulting cooldown rate.
IF STEAM COOLING IS REQUIRED	THEN perform Steam Cooling

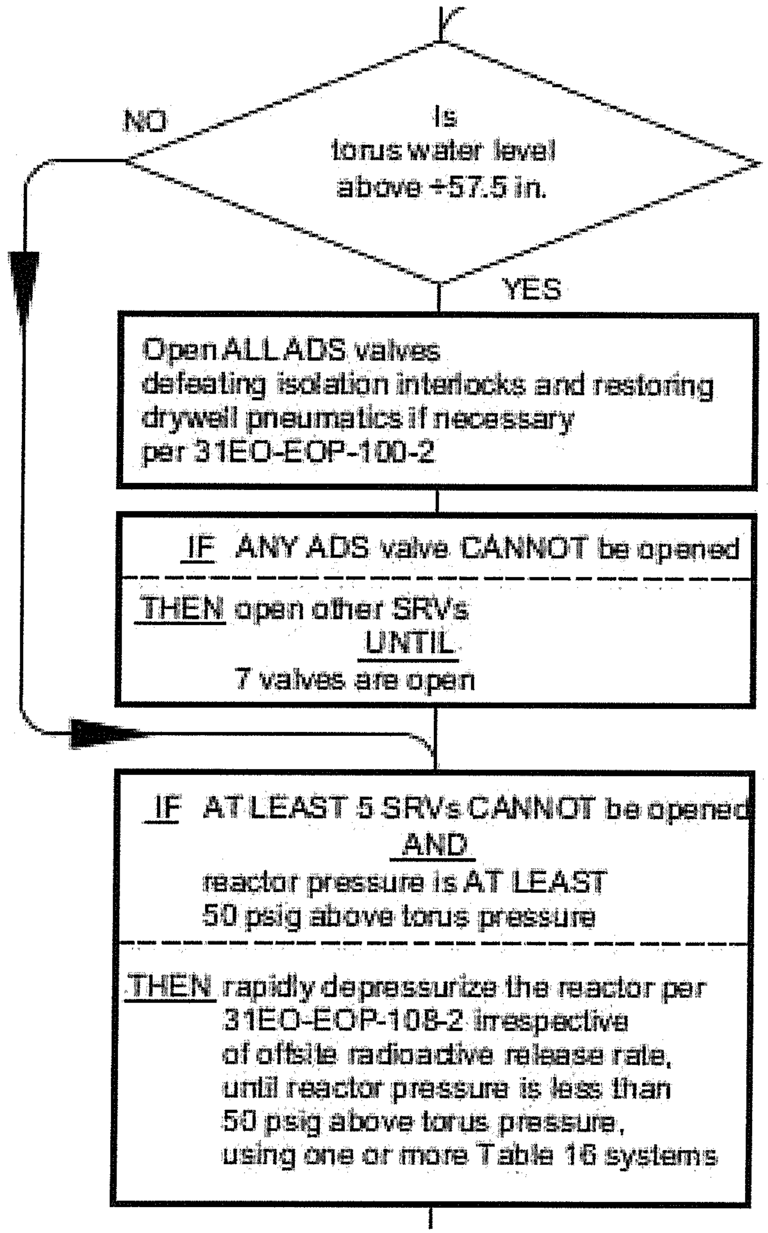
D

3 4 5

Stabilize reactor pressure below 1074 psig with main turbine bypass valves. If desired use one or more of the following:

- o Low Low Set
  If necessary, allow reactor pressure to exceed 1074 psig then CYCLE any SRV to initiate LLS
- o Alternate Reactor Pressure Control System(s) in Table 1 per 31EO-EOP-107-2

31EO-EOP-015-2, CP-1 POINT G Path



This question was one of the five SRO questions previously submitted for review

SRO question

2 of 5

Question change required
Changes were made

80. 261000A2.08 001

Unit 2 is operating at 100% RTP when the following occurs:

- o 10:00
  Fire alarm received on Unit 2 Control Building
  SO dispatched to investigate

- o 10:02
  Loss of 24/48V DC Cabinet 2A, 2R25-S015, occurs

- o 10:03
  SO reports 24/48V DC Cabinet 2A, 2R25-S015 is on fire

- o 10:17
  SO reports 2R25-S015 fire is EXTINGUISHED, however,
  2R25-S015 is severly damaged

IAW 34AB-R22-001-2, Loss Of DC Buses, at 10:05, without any operator actions, the TOTAL number of SBGT fans running is __________ .

IAW NMP-EP-110, Emergency Classification Determination and Initial Actions, the fifteen (15) minute clock for declaring an emergency STARTS at __________ .

A.✓ four (4);
10:00

B. four (4);
10:03

C. three (3);
10:00

D. three (3);
10:03

Description:

Edwin, this was question 2 of 5 of the SRO questions that you have already reviewed. Any discussed changes have been incorporated.

24/48V DC Cabinet 2A, 2R25-S015, provides DC power to various safety related equipment (SRMs, IRMs, RF Trip Units for PCIS logic and SBGT start).

Trip Auxiliary Units 2C51-Z2A & Z2C, 2H11-P606 (Radiation Monitor relays for secondary

2R25-S015, is de-energized.

IAW 34AB-R22-001-2, Loss Of DC Buses, Section 2.0 Automatic Actions, step 2.2 states "SBGT system auto start (Unit 1 and Unit 2)" which will start **both** SBGT fans on **each** unit. One of the SBGT fans on Unit 1 will automatically shutdown on low flow after a time delay. If the trains are running due to an auto initiation signal, the train that is in a low flow condition will be automatically shutdown after a given time delay (4 minutes for train "B" and 6 minutes for train "A")

The SRO will be required from memory to know the 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Per NMP-EP-110-GL02.

NMP-EP-110-GL02
The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, *Detection* is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene.

The "B" distractor is plausible since the first part is correct. The second part is plausible if the applicant only remembers the "visual observation and report by plant personnel" definition of *Detection,* IAW NMP-EP-11--GL02, and does not know the "sensor alarm indication" portion of the definition.

The "C" distractor is plausible if the applicant confuses the time delay shutdown for U1 SBGT on low flow and thinks one of the U1 SBGT fans has shutdown, leaving three (3) in service. The second part is correct.

The "D" distractor is plausible if the applicant confuses the time delay shutdown for U1 SBGT on low flow and thinks one of the U1 SBGT fans has shutdown, leaving three (3) in service. The second part is plausible if the applicant only remembers the "visual observation and report by plant personnel" definition of *Detection,* IAW NMP-EP-11--GL02, and does not know the "sensor alarm indication" portion of the definition.

A. **Correct** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

<u>K/A:</u>

261000 Standby Gas Treatment System

A2. Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6)

A2.08 D.C. electrical failure 2.4 2.7

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

<u>LESSON PLAN/OBJECTIVE:</u>

R42-ELECT-LP-02704, DC Electrical Distribution, EO 200.018.A.03
EP-LP-20101, Initial/Terminating Activities, 001.017.A

<u>References used to develop this question:</u>

34AB-R22-001-2, Loss of DC Buses, **Ver. 4.3**
NMP-EP-110, Emergency Classification Determination and Initial Actions, **Ver. 5.0**
Load list A-20159 for 2R25-S015 **Ver. 3.0**
NMP-EP-110-GL02, HNP EALs - ICs, Threshold Values and Basis,
(HA2 & HU2 Criteria), **Ver. 2.0**
U2 TRM LFD-2-SCIS-4
NMP-EP-110-GL-02, HNP EALs - ICs, Threshold Values and Basis, **Ver 2.0**

Item 1: SRO ONLY Guideline
Item 2: 34AB-R22-001-2, page 44, **Ver. 4.3**
Item 3: A-20159 Load List for 2R25-S015, page 5, **Ver. 3.0**
Item 4: HOT EALS HA2 & HU2, **Ver. 2.0**
Item 5: U2 TRM LFD-2-SCIS-4, **Rev. 0**
Item 6: NMP-EP-110-GL-02, **Ver 2.0**

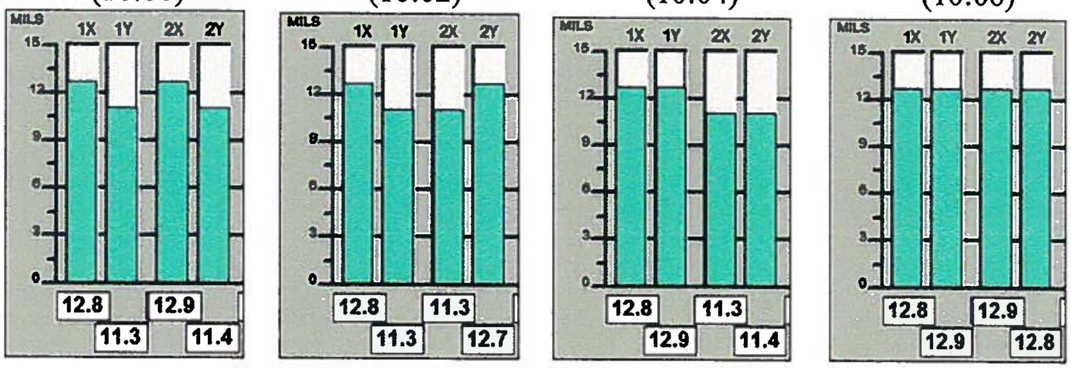
Modified from ILT-6 NRC Exam Q#85

ORIGINAL QUESTION (HLT-6 NRC Exam Q#85)

Unit 2 is operating at 400 GMWe. The following DEHC Mark VI vibration displays were taken for Main Turbine bearings #1 and #2 at the following times.

Time:

(10:00) (10:02) (10:04) (10:06)



Subsequently, the Unit 2 Main Turbine automatically trips.

A local Systems Operator reports that part of a turbine blade has been expelled from the Unit 2 Main Turbine and caused visible damage to the Unit 2 Reactor Building wall.

Based strictly on the above indications, which ONE of the following completes the statements below?

IAW 34SO-N30-001-2, Main Turbine Operation, the FIRST Unit 2 Main Turbine High Vibration trip signal was received __________ 10:03.

IAW NMP-EP-110, Emergency Classification Determination and Initial Actions, an emergency declaration __________ required.

A. prior to;
is NOT

B. after;
is NOT

C. prior to;
is

D.✓ after;
is

Q#80 K/A 261000A2.08

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

2.0 AUTOMATIC ACTIONS

2.1 IF OPEN, off gas Adsorbers Bypass valve, 2N62-F043, CLOSES

2.2 SBGT system auto start (Unit 1 and Unit 2)

2.3 Secondary Containment isolation

2.4 Half Scram Channel A (IF NOT in RUN)

2.5 Control Rod Withdrawal Block (in STARTUP or REFUEL only)

2.6 Group 2 Isolation Valves close
(except 2G11-F019, 2G11-F020, Drywell Equip. Drain Sump Isolation valves AND 2G11-F003, 2G11-F004, Drywell Floor Drain Sump Isolation valves)

3.0 IMMEDIATE OPERATOR ACTIONS

None

BREAKER 7 CABLE N/A

DESCRIPTION: Spare, 20 amp 2 pole breaker.

REFERENCE DRAWINGS:

S/L H-23635 Sh. 1, W/D H-23271

BREAKER 8 CABLE DAX704M01

DESCRIPTION:

1. Loss of power to 2C51A-Z2A and 2C51A-Z2C Trip Auxiliary Units in panel 2H11-P606. This loss simulates a refueling floor high radiation signal to PCIS logic and results in SBGT (Units 1 and 2) starting.
2. ERF input signal on Secondary Containment Auto Isolation Group Initiation.

REFERENCE DRAWINGS:

S/L H-23635 Sh. 1, W/D H-23271, W/D H-27848, E/D H-27158, E/D H-27159, E/D H-27620, E/D H-27621, E/D H-27625, E/D H-27631, E/D H-27634, B-27620 Sh. 2, E/D H-27629, E/D H-27767, E/D H-27769

BREAKER 9 CABLE N/A

DESCRIPTION: Spare, 20 amp 2 pole breaker.

REFERENCE DRAWINGS:

S/L H-23635 Sh. 1, W/D H-23271

BREAKER 10 CABLE N/A

DESCRIPTION: Spare, 20 amp 2 pole breaker.

REFERENCE DRAWINGS:

S/L H-23635 Sh. 1, W/D H-23271

<u>HA2</u> - FIRE **<u>OR</u>** EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown (Pg. 97)

1. FIRE **<u>OR</u>** EXPLOSION

   <u>AND</u>

   Affected system parameter indications show degraded performance

   <u>OR</u>

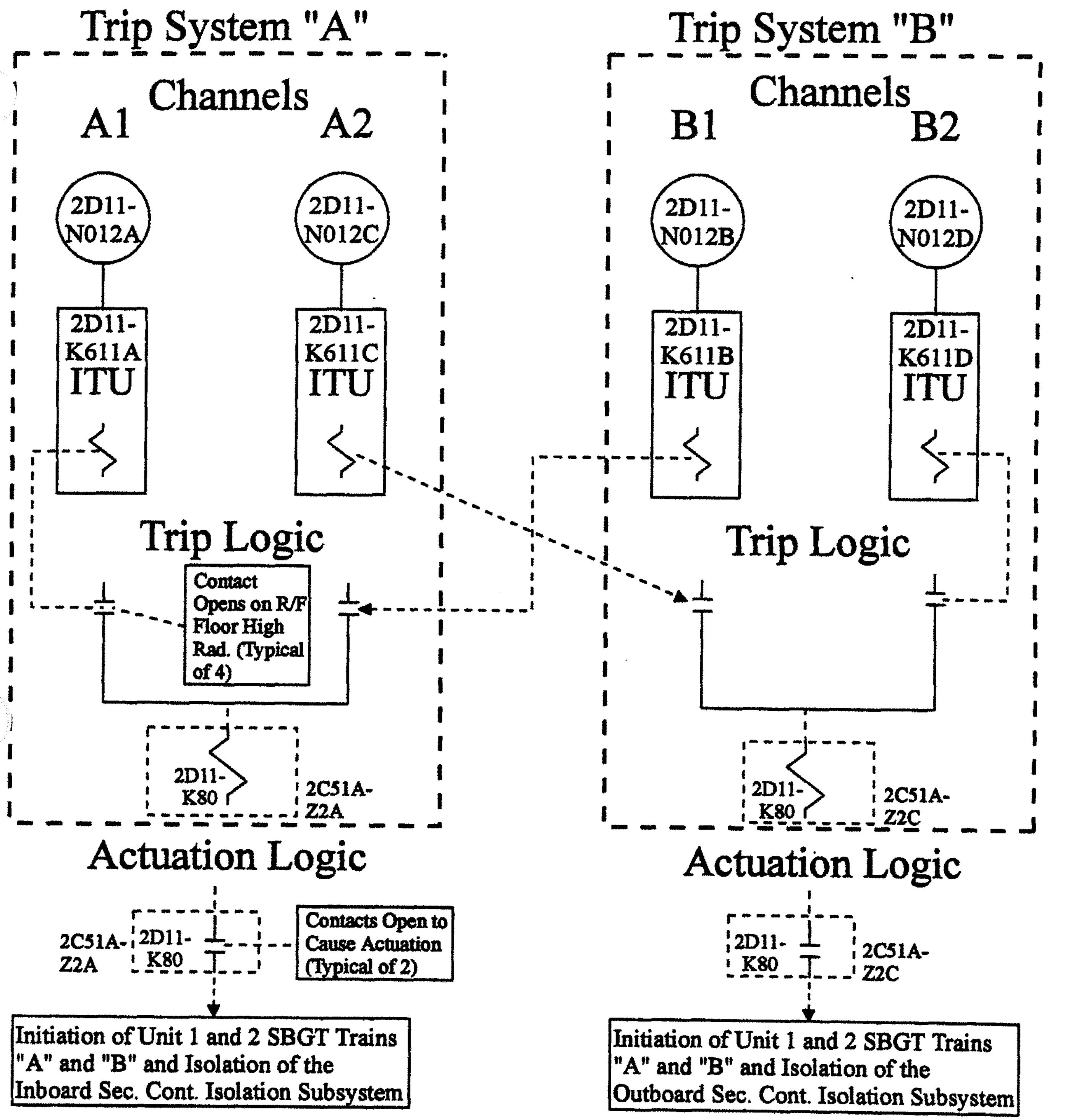
   Plant personnel report VISIBLE DAMAGE to permanent structures or safety related equipment in any of the following VITAL AREA:

   Primary Containment
   Reactor Building
   Diesel Generator Building
   Control Building
   Intake Structure

<u>HU2</u> - FIRE Within PROTECTED AREA Boundary **<u>NOT</u>** Extinguished Within 15 Minutes of Detection (Pg. 107)

1. FIRE in buildings or areas contiguous to any of the following areas **<u>NOT</u>** extinguished within 15 minutes of control room notification or control room alarm unless disproved by personnel observation within 15 minutes of the alarm:

   Primary Containment
   Reactor Building
   Diesel Generator Building
   Control Building
   Intake Structure



Minimum Channel Requirements for System Isolation/Initiation Capability:

In order to maintain Secondary Containment isolation capability and SBGT initiation capability on a Refueling Floor Exhaust Radiation - High signal, channels in one of the following combinations must be operable or maintained in the tripped condition.

Elem. Ref.

H-27597	H-27600	H-27734
H-27598	H-27629	H-27761
H-27599	H-27631	H-27767
	H-27732	

A1 and B1
OR
A2 and B2

Prepared By: T.L. Bruner

Reviewed By: Royce Clark

LFD-2-SCIS-04

TS 3.3.6.2-1, Item 4
R/F Floor Exhaust
Radiation-High

Rev. 0 10/20/94

Initiating Condition HU2

FIRE Within PROTECTED AREA Boundary **NOT** Extinguished Within 15 Minutes of Detection.

Operating Mode Applicability: All

Threshold Value:

1. FIRE in buildings or areas contiguous to any of the following areas **NOT** extinguished within 15 minutes of control room notification or control room alarm unless disproved by personnel observation within 15 minutes of the alarm:

Primary Containment
Reactor Building
Diesel Generator Building
Control Building
Intake Structure

Basis:

FIRE: is combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIREs. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

PROTECTED AREA: the area which normally encompasses all controlled areas within the security protected area fence.

The purpose of this IC is to address the magnitude and extent of FIREs that may be potentially significant precursors to damage to safety systems. As used here, *Detection* is visual observation and report by plant personnel or sensor alarm indication. The 15 minute time period begins with a credible notification that a FIRE is occurring, or indication of a VALID fire detection system alarm. Verification of a fire detection system alarm includes actions that can be taken with the control room to ensure that the alarm is not spurious. A verified alarm is assumed to be an indication of a FIRE unless it is disproved within the 15 minute period by personnel dispatched to the scene.

The intent of this 15 minute duration is to size the FIRE and to discriminate against small FIREs that are readily extinguished. The list is limited and applies to buildings and areas contiguous to plant VITAL AREAs or other significant buildings or areas.

81. 263000G2.4.8 001

Unit 2 is in Startup with the following conditions:

- o Reactor pressure is 80 psig and steady

An event occurs with the following:

- o The Supply breaker to 600 V Bus 2C trips and can NOT be re-closed
- o RWL is 15" increasing 1" per minute (lowest level reached 10")

Given these conditions, which ONE of the following completes both statements?

IAW Tech Specs, a Required Action Statement MUST be entered for __________ .

In order to restore the associated Station Service Battery Chargers to service and IAW 34AB-R23-001-2, "Loss of 600 Volt Emergency Bus", energizing 600 VAC bus "2C" using the 4160/600V "2CD" Transformer is __________ .

A.✓ 600VAC "2C" ONLY;
NOT allowed

B. 600VAC "2C" ONLY;
allowed

C. 600VAC "2C" AND also for Instrument Bus "2A";
NOT allowed

D. 600VAC "2C" AND also for Instrument Bus "2A";
allowed

Description:

IAW 34AB-R23-001-2, step 4.3 states "IF the affected 600 V bus is de-energized due to a loss of its 4160 V supply bus AND its 4160 V supply bus cannot be restored, ENERGIZE the 600 V bus from its alternate 4160 V supply per procedure 34SO-R23-001-2, 600V/480BV AC System. ONLY use 4160/600V 2CD transformer WHEN in plant condition 4 OR 5 OR when EOP's are entered AND THEN only IF 1B D/G loading permits.

With the supply breaker to 600 V Bus 2C open, 600 V 2C will remain de-energized. Since NO entry condition exists for the EOPs AND Unit 2 is in Mode 2, the procedure **DOES NOT allow** 600 V 2C to be energized through 4160/600V 2CD transformer.

IAW TS Bases 3.8.7, "Should one or more buses not listed in LCO 3.8.7 become inoperable due to a failure not affecting the OPERABILITY of a bus listed in LCO 3.8.7 (e.g., a breaker supplying a single MCC faults open), the individual loads on the bus would be considered individual loads would be entered. If however, one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in LCO 3.8.7 (e.g., loss of a 4.16 kV ESF bus, which results in de-energization of all buses powered from the 4.16 kV ESF bus), the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV ESF bus)." 600V Bus 2C is a tech spec required bus and feeds Instrument Bus 2A via Essential Cabinet 2A. However, **Tech Specs does not cascade**. A loss of the 600V Bus 2C **ONLY** requires the actions for 600V Bus 2C.

The SRO must apply Motherhood statement LCO 3.0.6 to properly answer this question. The 600V Bus 2C (support system) is a Tech Spec required bus and feeds Instrument Bus 2A (supported system). However, Tech Specs does not cascade. A loss of the 600V Bus 2C only requires a RAS for 600V Bus 2C. ROs are not responsible for the Motherhood Statements from memory and are above the RO knowledge level.

The "B" distractor is plausible since the first part is correct. The second part is plausible if the applicant does not remember the requirement for being in the EOPs or in Mode 4 or 5 condition (currrent Mode is 2). Loading considerations for the 1B EDG will be zero since there are no conditions indicating the 1B EDG is running loaded.

The "C" distractor is plausible if the applicant does not remember or confuses TS LCO 3.0.6 and thinks Instrument Bus 2A will also require a RAS to be generated since it is de-energized. The second part is correct.

The "D" distractor is plausible if the applicant does not remember or confuses TS LCO 3.0.6 and thinks Instrument Bus 2A will also require a RAS to be generated since it is de-energized. The second part is plausible if the applicant does not remember the requirement for being in the EOPs or in Mode 4 or 5 condition (currrent Mode is 2). Loading considerations for the 1B EDG will be zero since there are no conditions indicating the 1B EDG is running loaded.

A. **Correct** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

263000 D.C. Electrical Distribution

2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13) 3.8 4.5

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.

LESSON PLAN/OBJECTIVE:

R23-ELECT-LP-02703, 600/ 480/ 208 VAC Electrical, EO 027.019.A.02
LT-LP-30005, Technical Specifications, EO 300.006.A.22

References used to develop this question:

34AB-R23-001-2, Loss Of 600 Volt Emergency Bus, **Ver. 1.11**
34SO-R23-001-2, 600V/480V AC System, Limitation 5.2.2, **Ver. 7.3**
U2 TS 3.8.7 Distribution Systems - Operating, **Amendment No. 210**
U2 TS Bases 3.8.7 Distribution Systems - Operating, **Rev. 39**

Item 1: SRO ONLY Guideline
Item 2: 34AB-R23-001-2, page 2, **Ver. 1.11**
Item 3: 34SO-R23-001-2, page 6, **Ver. 7.3**
Item 4: U2 TS 3.8.7, page 3.8-37, **Amend. 210**
Item 5: U2 TS Bases 3.8.7, page B3.8-74, **Rev. 39**

Modified question used on HLT-4 NRC Exam Q#03

ORIGINAL QUESTION (HLT-4 NRC Exam Q#03)

Unit 2 was operating at 100% power when a Loss Of Coolant Accident occurred.

These conditions exist:

- o The reactor has scrammed and all control rods fully inserted
- o RPV pressure .... 960 psig, increasing at 4 psig/minute
- o Drywell (DW) pressure ... 7 psig (decreasing)
- o BOTH RHR loops............ DW spray mode
- o RPV level .... -145 inches, decreasing at 2"/minute
- o 4160 VAC bus "2E" has de-energized and cannot be re-energized

Given these conditions, IAW 34AB-R23-001-2, "Loss of 600 Volt Emergency Bus", energizing 600 VAC bus "2C" using the 4160/600V "2CD" Transformer is ___________________.

A. NOT allowed; the reactor must be in Mode 4

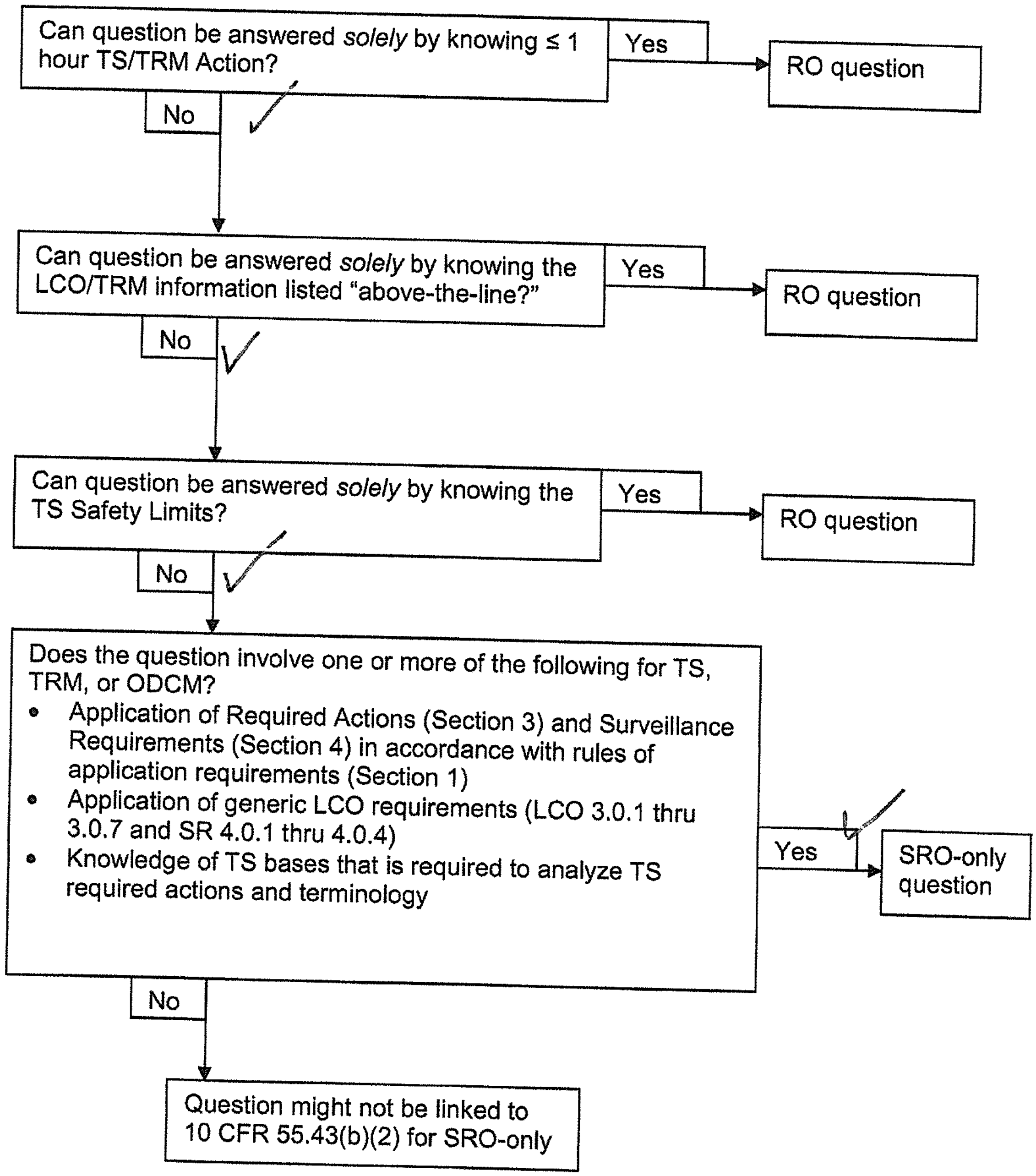
B. NOT allowed; the "1B" Emergency Diesel Generator would be overloaded

C.✓ allowed; ALL low pressure ECCS will NOT inject at rated flow when reactor pressure decreases to below the shutoff head

D. allowed; but is NOT needed since all low pressure ECCS will inject at rated flow when reactor pressure decreases to below the shutoff head

Q#81 K/A 263000 G2.4.8

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



4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 Monitor primary containment temperature and pressure. ☐

4.2 Notify Unit 1 Plant Operator to perform the following:

- OPEN the affected 1R24-S018A (1R24-S018B) MCC NORMAL SUPPLY BRKR, panel 1H11-P601. ☐
- Depending upon the prognosis of 2C (D) 600 Volt Bus,
  EITHER energize the affected 1R24-S018A (1R24-S018B) MCC from its ALTERNATE SUPPLY per 34SO-R24-003-1 ☐
  OR
  re-close its NORMAL SUPPLY BRKR
  WHEN its NORMAL 600 Volt power source becomes available. ☐

NOTES	• Bus link is normally installed in 2C and must be moved to supply 2D 600V Bus from the 2CD transformer. • The 600 volt supply breakers to the 2C (2D) bus are interlocked so that only one breaker, normal or alternate, can be closed at one time.

4.3 IF the affected 600 V bus is de-energized due to a loss of its 4160 V supply bus AND its 4160 V supply bus cannot be restored,
ENERGIZE the 600 V bus from its alternate 4160 V supply per procedure 34SO-R23-001-2, 600V/480BV AC System.
Only use 4160/600V 2CD transformer
WHEN in plant condition 4 OR 5 OR when EOP's are entered AND
THEN only IF 1B D/G loading permits. ☐

4.4 IF the Vital AC Bus is de-energized,
enter 34AB-R25-001-2, Loss of Vital AC Bus. ☐

4.5 Perform the following applicable abnormal procedures concurrently with this procedure:

PROCEDURE NUMBER		
34AB-R25-002-2	Loss of Instrument Buses	☐
34AB-R24-001-2	Loss of Essential AC Distribution Buses	☐
34AB-P51-001-2	Loss of Instrument and Service Air System	☐
34AB-T47-001-2	Complete Loss of Drywell Cooling	☐
34AB-C71-002-2	Loss of RPS	☐
34AB-P42-001-2	Loss of RBCCW	☐
34AB-R22-001-2	Loss of DC Buses	☐

5.0 PRECAUTIONS/LIMITATIONS

5.1 PRECAUTIONS

5.1.1 Observe safety rules and PPE requirements as provided in NMP-SH-003, Electrical Work Practices.

5.1.2 600V AC breakers are furnished with individual protective relays which initiate an automatic signal to trip the breaker during a fault OR abnormal condition. Breakers are tripped by overcurrent OR differential type relays in order to disconnect AND isolate the electrical fault AND protect the electrical equipment WHILE maintaining continuity of service on the remainder of the system.
Following a trip, breaker restoration must be accomplished in accordance with 31GO-OPS-021-0, Manipulation Of Controls And Equipment, AND NMP-OS-007-001, Conduct of Operations Standards and Expectations.

5.1.3 Use the applicable attachment, Attachment 3, 4, 5, or 6, to obtain the incident energy level for the bus to be racked, and refer to NMP-SH-003, Electrical Work Practices, for the protective gear requirements based on this energy level.

5.2 LIMITATIONS

5.2.1 Normal AND Alternate bus feed circuit breakers for 600V AC buses 2A, 2B, 2C, 2D, 2AA, AND 2BB will NOT trip on bus undervoltage. A tripped supply breaker indicates a bus fault. A bus fault must be corrected in accordance with 31GO-OPS-021-0, Manipulation Of Controls And Equipment, AND NMP-OS-007-001, Conduct of Operations Standards and Expectations.

5.2.2 Only use 4160/600V 2CD transformer:
WHEN in plant condition 4 or 5 AND only THEN IF 1B D/G loading permits.
OR
WHEN EOP's are entered AND only THEN IF 1B D/G loading permits.

5.2.3 Crossfeeding the 600V Bus 2B from 4160V Bus 2C will only be allowed WHEN the Plant is in:
COLD SHUTDOWN condition
OR
REFUELING MODE
OR
with no fuel in the vessel
AND
with 2C Condensate Pump AND 2C Condensate Booster Pump TAGGED OUT per NMP-AD-003, Equipment Clearances and Tagging. (Ref. 34SO-R23-004-2)

5.2.4 Due to bus loading evaluations not being performed, crossfeeding 600V Buses 2A, 2AA, and 2BB from 4160V Bus 2D is not allowed.

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Distribution Systems - Operating

LCO 3.8.7 The following AC and DC electrical power distribution subsystems shall be OPERABLE:

a. Unit 2 AC and DC electrical power distribution subsystems comprised of:

1. 4160 V essential buses 2E, 2F, and 2G;
2. 600 V essential buses 2C and 2D;
3. 120/208 V essential cabinets 2A and 2B;
4. 120/208 V instrument buses 2A and 2B;
5. 125/250 V DC station service buses 2A and 2B;
6. DG DC electrical power distribution subsystems; and

b. Unit 1 AC and DC electrical power distribution subsystems needed to support equipment required to be OPERABLE by LCO 3.6.4.3, "Standby Gas Treatment (SGT) System"; LCO 3.7.4, "Main Control Room Environmental Control (MCREC) System"; LCO 3.7.5, "Control Room Air Conditioning (AC) System"; and LCO 3.8.1, "AC Sources - Operating."

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Unit 1 AC or DC electrical power distribution subsystems inoperable.	A.1 Restore required Unit 1 AC and DC subsystem(s) to OPERABLE status.	7 days

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6 Containment Systems.

The OPERABILITY of the AC and DC electrical power distribution subsystems is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining distribution systems OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite power sources or all onsite AC electrical power sources; and

b. A postulated worst case single failure.

The AC and DC electrical power distribution system satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

LCO

The Unit 2 AC and DC electrical power distribution subsystems are required to be OPERABLE. The required Unit 2 electrical power distribution subsystems listed in LCO 3.8.7 ensure the availability of AC and DC electrical power for the systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA.

Should one or more buses not listed in LCO 3.8.7 become inoperable due to a failure not affecting the OPERABILITY of a bus listed in LCO 3.8.7 (e.g., a breaker supplying a single MCC faults open), the individual loads on the bus would be considered inoperable, and the appropriate Conditions and Required Actions of the LCOs governing the individual loads would be entered. If however, one or more of these buses is inoperable due to a failure also affecting the OPERABILITY of a bus listed in LCO 3.8.7 (e.g., loss of a 4.16 kV ESF bus, which results in de-energization of all buses powered from the 4.16 kV ESF bus), the Conditions and Required Actions of the LCO for the individual loads are not required to be entered, since LCO 3.0.6 allows this exception (i.e., the loads are inoperable due to the inoperability of a support system governed by a Technical Specification; the 4.16 kV ESF bus). In addition, since some components required by Unit 2 receive power through Unit 1 electrical power distribution subsystems (e.g., Standby Gas Treatment (SGT) System, Low Pressure Coolant Injection (LPCI) valve load centers, Main Control Room Environmental Control (MCREC) System,and Control Room Air Conditioning (AC) System), the Unit 1 AC and DC electrical power distribution subsystems needed to support the required equipment must also be OPERABLE.

(continued)

82. 290001A2.02 001

Unit 2 is operating at 100% RTP operating in TYPE "A" Containment.

A SO reports that one (1) of the Unit 2 Reactor Building (RB) Blowout panels is damaged and is NOT fully closed and sealed.

- o Unit 2 Reactor Building (RB) dP is +0.02 inches WC and steady

Operations and Engineering are reviewing recent performances of 34SV-T22-002-0, Secondary Containment Integrity, and determines the 31 day surveillance was last performed on July 19.

Todays date is September 19.

Entry into 34AB-T22-002-2, Loss Of Secondary Containment Integrity, __________ REQUIRED at this time.

IAW TS, without performing a Risk Evaluation, the LATEST time allowed to perform 34SV-T22-002-0 before requiring entry into a RAS, is __________ .

A.✓ is;
24 hours

B. is;
31 days

C. is NOT;
24 hours

D. is NOT;
31 days

Description:

TS B3.6.4.1, Secondary Containment, - An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and (continued) processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum (0.20 inch of vacuum) can be established and maintained. The secondary containment boundary required to be OPERABLE is dependent on the operating status of both units, as well as the configuration of doors, hatches, refueling floor plugs, SCIVs, and available flow paths to SGT Systems.

Verifying that secondary containment equipment hatches and one access door in each access maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides **adequate assurance that exfiltration from the secondary containment will not occur**. SR 3.6.4.1.1 also requires equipment hatches to be sealed. In this application, the term "sealed" has no connotation of leak tightness. Maintaining secondary containment OPERABILITY requires verifying one door in the access opening is closed. An access opening contains one inner and one outer door. The intent is **not to breach** the secondary containment at any time when secondary containment is required. This is achieved by maintaining the inner or outer portion of the barrier closed at all times. However, all secondary containment access doors are normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access opening.

Therefore, with the Reactor Building (RB) Blowout panel damaged, the integrity of Secondary Containment is lost, plus RB dP of -0.02 inches WC will result in entry condition alarms being received, requiring entry into 34AB-T22-002-2, Loss Of Secondary Containment Integrity.

TS SR 3.0.3 states "If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed > 24 hours and the risk impact shall be managed."

The SRO must apply LCO SR 3.0.3 in order to fully answer this question correctly. ROs are not responsible for the Motherhood Statements from memory and are above the RO knowledge level.

The "B" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses LCO SR 3.0.3 which allows performance from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. 31 days is the specified frequency but a risk evaluation must be performed to extend to this time.

The "C" distractor is plausible if the applicant does not realize that RB dP of -0.02 inches WC is an entry condition to the Abnormal procedure or confuses the layout of the RB Blowout panels and thinks they are in series similar to RB doors and other blowout devices. The second part is correct.

The "D" distractor is plausible if the applicant does not realize that RB dP of -0.02 inches WC is an entry condition to the Abnormal procedure or confuses the layout of the RB Blowout panels and thinks they are in series similar to RB doors and other blowout devices. The second part is plausible if the applicant confuses LCO SR 3.0.3 which allows performance from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. 31 days is the specified frequency but a risk evaluation must be performed to extend to this time.

A. **Correct** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

290001 Secondary Containment

A2. Ability to (a) predict the impacts of the following on the SECONDARY CONTAINMENT ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6)

A2.02 †Excessive outleakage 3.5 3.7

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.

LESSON PLAN/OBJECTIVE:

T22-SC-LP-01302, Secondary Containment, EO 300.006.C.02
LT-LP-20201, Introduction To Abnormal Procedures, EO LT-20201.002

References used to develop this question:

34AB-T22-002-2, Loss Of Secondary Containment Integrity, **Ver. 1.1**
34AR-654-001-2, RB Inside To Outside Air Diff Press Low, **Ver. 6.3**
34SV-T22-002-0, Secondary Containment Integrity, **Ver. 2.12**
U2 TS SR 3.0.3, Surveillance Requirement (SR) Applicability, **Amendment No. 194**

Item 1: SRO ONLY Guideline
Item 2: 34AB-T22-002-2, page 1, **Ver. 1.1**
Item 3: 34AR-654-001-2, **Ver. 6.3**
Item 4: 34SV-T22-002-0, Section 2.0, **Ver. 2.12**
Item 5: U2 TS SR 3.03, **Amend. 194**

Q# 82 K/A 290001A2.02

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

<table>
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PLANT E. I. HATCH</td><td colspan="3">DOCUMENT TYPE:
ABNORMAL OPERATING PROCEDURE</td><td>PAGE
1 OF 2</td></tr>
<tr><td colspan="3">DOCUMENT TITLE:
LOSS OF SECONDARY CONTAINMENT INTEGRITY</td><td>DOCUMENT NUMBER:
34AB-T22-002-2</td><td>VERSION NO:
1.1</td></tr>
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N/A</td><td colspan="3">APPROVALS:
DEPARTMENT MANANGER DRM DATE 1-23-97

NPGM/POAGM/PSAGM N/A DATE N/A</td><td>EFFECTIVE DATE:

5-19-08</td></tr>
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1.0 CONDITIONS

1.1 ANNUNCIATORS

1.1.1 REFUELING FLOOR OUTSIDE AIR DIFF PRESS LOW, 657-001

1.1.2 RB INSIDE TO OUTSIDE AIR DIFF PRESS LOW, 654-001

1.2 Failure of the required SBGT subsystem when Secondary Containment is required.

1.3 Inability to secure closed an inoperable ventilation system isolation valve necessary to maintain secondary containment integrity.

1.4 Inability to maintained closed at least one door in each access opening to the Secondary Containment.

1.5 Visual observation of failure of the reactor building to remain intact.

2.0 AUTOMATIC ACTIONS

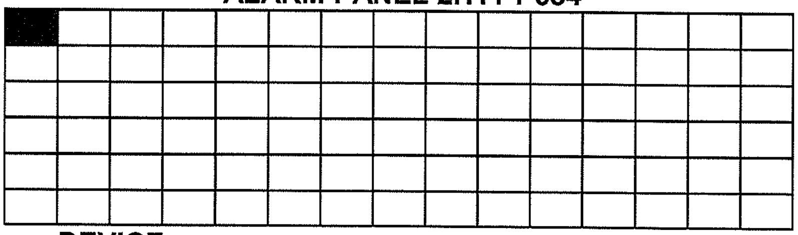
NONE

3.0 IMMEDIATE OPERATOR ACTIONS

NONE

1.0 IDENTIFICATION:

ALARM PANEL 2H11-P654



RB INSIDE TO
OUTSIDE AIR
DIFF PRESS LOW

DEVICE: 2T46-N600B

SETPOINT: -0.06" WC

2.0 CONDITION:

The Reactor Building to outside air differential pressure is no longer less than -0.06" WC

3.0 CLASSIFICATION: EQUIPMENT STATUS

4.0 LOCATION: 2H11-P654

5.0 OPERATOR ACTIONS:

5.1 Confirm Reactor Building to outside air differential pressure, is greater than -0.06" WC, as indicated on 2T46-R604A, Sec Cnmt Diff Press A, and 2T46-R604B, Sec Cnmt Diff Press B, on Panel 2H11-P700. ☐

5.2 Confirm Reactor Building Ventilation System is in operation per 34SO-T41-005-2, Reactor Building Ventilation System. ☐

5.3 IF the Reactor Building Ventilation has automatically isolated, confirm the SBGT System has started AND is aligned to Reactor Building per 34SO-T46-001-2, Standby Gas Treatment System. ☐

5.4 IF an abnormal radioactive release is occurring OR has occurred, enter 34AB-D11-001-2, Radioactivity Release Control. ☐

5.5 At 2R25-S065, confirm BRKR 15 is CLOSED. ☐

5.6 Enter 34AB-T22-002-2, Loss of Secondary Containment Integrity. ☐

5.7 Enter 34AB-T22-003-2, Secondary Containment Control. ☐

6.0 CAUSES:

6.1 Automatic Reactor Building Ventilation Isolation

6.2 Improper operation of the Reactor Building Ventilation System

6.3 Loss of Secondary Containment

7.0 REFERENCES

7.1 A-26464-T46A, Data Sheet
7.2 H-27769, SBGT 2T46 Elementary Diagram
7.3 A-26497, Instrument Setpoint Index

8.0 TECH. SPECS./TRM/ODCM/FHA:

8.1 3.6.4.1 Secondary Containment

SOUTHERN NUCLEAR PLANT E. I. HATCH	DOCUMENT TYPE: SURVEILLANCE PROCEDURE			PAGE 1 OF 20
DOCUMENT TITLE: SECONDARY CONTAINMENT INTEGRITY		DOCUMENT NUMBER: 34SV-T22-002-0		VERSION NO: 2.12
EXPIRATION DATE: N/A	APPROVALS: DEPARTMENT MGR J. I. Hammonds	DATE 10/24/96		EFFECTIVE DATE: 09/19/12
	SSM/PM N/A	DATE N/A		

1.0 OBJECTIVE

This procedure provides instructions for checking Secondary Containment Integrity as required by Unit 1 and 2 TS SR's 3.6.4.1.1, 3.6.4.1.2, and 3.6.4.2.1.(SNC27703)

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Section **Page**

Attachments

2.0 APPLICABILITY

This procedure applies to all penetrations NOT capable of being closed by OPERABLE containment automatic isolation valves AND required to be closed during accident conditions that are closed by valves, blind flanges OR deactivated automatic valves secured in position.

This procedure also confirms:
(1) All (Secondary Containment) equipment hatches are closed and sealed, AND
(2) At least one door in each access to the secondary containment is closed.

This procedure is performed at least every 31 days.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed > 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

83. 290003G2.4.4 001

At 08:58, **Unit 1** is operating at 7% RTP ready to transfer the Reactor mode switch to "RUN".

At 09:00, an event occurs causing Refueling Floor ARMs to indicate the following:

1D21-K601A, Reactor head laydown area,	52 mr/hr
1D21-K601B, Refueling Floor Stairway,	25 mr/hr
1D21-K601D, Refueling Floor,	24 mr/hr
1D21-K601E, Drywell Shield Plug,	22 mr/hr
1D21-K601M, Spent Fuel Pool & New Fuel Storage,	40 mr/hr

At 09:03, Unit 1 NPO reports 1Z41-C012B, Control Room HVAC Filter Fan, will NOT run. LCO 3.7.4 Main Control Room Environmental Control (MCREC) System, RAS is entered.

At 09:05, Refueling Floor HP notifies the control room that High Radiation trash was being moved on the Refueling Floor and had caused the higher than normal radiation conditions.

At 09:07, the High Radiation trash has been removed from the Refueling Floor. The ABOVE ARMs are now indicating NORMAL values.

At 09:15, the operating crew is ready to transfer the Reactor Mode switch to "RUN".

At 09:00, an entry condition existed for ___________ .

At 09:15, with 1Z41-C012B inoperable and without performance of a Risk Assessment, Tech Specs ___________ allow transferring the Reactor Mode switch to the "RUN" position.

Reference Provided

A. ONLY 34AB-T22-003-1, Secondary Containment Control;
will

B. ONLY 34AB-T22-003-1, Secondary Containment Control;
will NOT

C. BOTH 34AB-T22-003-1, Secondary Containment Control, AND
31EO-EOP-014-1, SC/RR, EOP Flowchart;
will

D✓ BOTH 34AB-T22-003-1, Secondary Containment Control, AND
31EO-EOP-014-1, SC/RR, EOP Flowchart;
will NOT

Description:

34AB-T22-003-1, Secondary Containment Control, AND 31EO-EOP-014-1, SC/RR, EOP Flowchart, both contain the same values for exceeding the Maximum Normal Radiation Levels (50 mr/hr). With 1D21-K601A, Reactor head laydown area, reaching 52 mr/hr, then both procedures entry conditions were exceeded.

At 09:03 LCO 3.7.4 Condition A was entered due to 1Z41-C012B, failing which Required Action A.1 requires the MCREC subsystem to be restored to OPERABLE status in 7 days.

Motherhood statement LCO 3.0.4 states: "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.

Without **performing** a risk assessment, the Reactor Mode switch can NOT be placed to "RUN".

The SRO must realize apply LCO 3.0.4 in order to fully answer this question correctly. ROs are not responsible for the Motherhood Statements from memory and are above the RO knowledge level.

The "A" distractor is plausible if the applicant does not know or confuses the entry condition values for Table 6. Also plausible if the applicant does not know that both procedures entry conditions are the same. The second part is plausible if the applicant confuses or does not properly apply LCO 3.0.4 and thinks the Reactor Mode switch can be transferred to "RUN".

The "B" distractor is plausible if the applicant does not know or confuses the entry condition values for Table 6. Also plausible if the applicant does not know that both procedures entry conditions are the same. The second part is correct.

The "C" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses or does not properly apply LCO 3.0.4 and thinks the Reactor Mode switch can be transferred to "RUN".

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
Att. 6 Secondary Containment Operating Radiation Levels of 34AB-T22-001-1 WITHOUT words "Max Normal Operating Value mR/hr" & "Max Safe Operating Value mR/hr".

TS 3.7.4 Main Control Room Environmental Control (MCREC) System, pages 3.7-8 thru 3.7-9

K/A:

290003 Control Room HVAC

2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.
(CFR: 41.10 / 43.2 / 45.6) 4.5 4.7

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.

LESSON PLAN/OBJECTIVE:

EOP-SCRR-LP-20325, Secondary Containment / Radioactivity Release Control,
EO 201.077.A.04
LT-LP-30005, Technical Specifications, EO 300.006.A.18

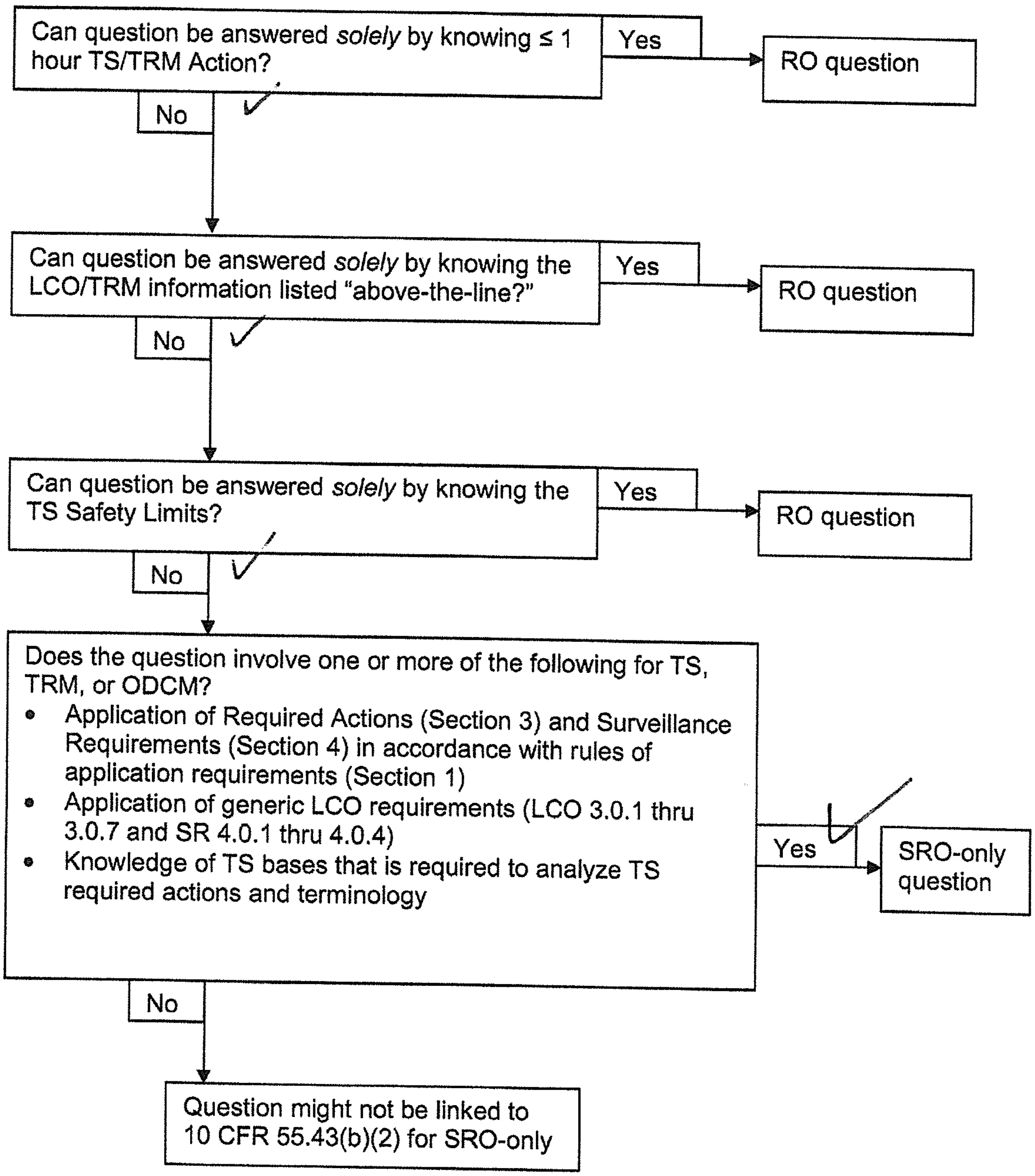
References used to develop this question:

34AB-T22-003-1, Secondary Containment Control
31EO-EOP-014-1, SC/RR, EOP Flowchart, **Ver. 12**
U1 TS 3.7.4 Main Control Room Environmental Control (MCREC) System, **Amend. 225**
U1 TS 3.0.4, pages 3.0-1 & 3.0-2, **Amend. 250 & 246**

Item 1: SRO ONLY Guideline
Item 2: 34AB-T22-003-1, pages 3 & 14, **Ver. 5.12**
Item 3: U1 SC RR Entry & Table 6, **Ver. 12**
Item 4: U1 TS 3.7.4, page 3.7-8, **Amend. 225**
Item 5: U1 TS 3.0.4, pages 3.0-1 & 3.0-2, **Amend. 250 & 246**

Q# 83 K/A 290003 G2.4.4

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



4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 Monitor Secondary Containment temperatures, pressures, radiation levels, AND sump levels.

4.2 IF at any time while performing this procedure, any of the following secondary containment parameters exceeds its maximum normal operating value in any area, enter 31EO-EOP-014-1, SC/RR Secondary Containment/Radioactivity Release Control:

- area ambient temperature (Attachment 2)
- area differential temperature (Attachment 3)
- differential pressure (Attachment 5)
- area radiation (Attachment 6)
- HVAC exhaust radiation (Attachment 6)
- area water level (Attachment 8)

4.3 IF an ambient temperature AND/OR differential temperature alarm is received, perform the following:

4.3.1 At panel 1H11-P614, on 1G31-R604 Temperature Recorder and 1G31-R608 Temperature Recorder, determine which sensor/area is in alarm.

4.3.2 Monitor Reactor Building AND Refueling Floor to outside air differential pressures at panel 1H11-P700, on 1T46-R604A & 1T46-R604B, Sec Cnmt Diff Press instruments.

4.3.3 Operate available area coolers.

4.3.4 IF secondary containment HVAC exhaust radiation level is below the secondary containment HVAC isolation setpoints (see Attachment 6), operate available secondary containment HVAC.

4.4 IF a secondary containment process radiation monitor alarm is received, perform the following:

4.4.1 At panel 1H11-P606, P645 or SPDS, determine/monitor actual radiation levels, including 1D11-R619, Stack Monitor.

4.4.2 IF secondary containment HVAC exhaust radiation level exceeds the secondary containment HVAC isolation setpoint (see Attachment 6), perform the following:

4.4.2.1 Confirm OR manually initiate isolation of secondary containment HVAC per Attachment 7.

4.4.2.2 Confirm initiation of OR manually initiate SBGT per 34SO-T46-001-1 and 34SO-T46-001-2, Standby Gas Treatment System.

4.4.2.3 PLACE the ON/OFF switches on 1D11-P010 and 1D11-P011, Fission Product Panels, in the OFF position. (Located at 158RHR05)

HVAC EXHAUST RADIATION ANNUNCIATORS ON 1H11-P601	MAXIMUM NORMAL OPERATING VALUE mr/hr
	HI-HI RADIATION ALARM
-RX BLDG POT CONTAM AREA (1D11-K609A, 1D11-K609B, 1D11-K609C, 1D11-K609D)	18
-REFUELING FLOOR VENT EXHAUST (1D11-K611A, 1D11-K611B, 1D11-K611C, 1D11-K611D)	18

AREA RADIATION MONITORS on 1H11-P600, 1D21-P600	Max Normal Operating Value mR/hr	Max Safe Operating Value mR/hr
REFUEL FLOOR AREA		
1 Reactor head laydown area (1D21-K601A)	50	1000
2 Refueling Floor Stairway (1D21-K601B)	50	1000
3 Refueling Floor (1D21-K601D)	50	1000
4 Drywell Shield Plug (1D21-K601E)	50	1000
5 Spent Fuel Pool & New Fuel Storage (1D21-K601M)	50	1000
203' ELEVATION AREA		
6 RB 203' Working Area (1D21-K601X)	50	1000
185' ELEVATION AREA		
7 Spent Fuel Pool Demin. Equip (1D21-K601C)	150	1000
8 Fuel Pool Demin. Panel (1D21-K617)	50	100
158' ELEVATION AREA		
9 RB 158' Working Area (1D21-K601K)	50	1000
10 Rx Wtr Sample Rack Area 158' (1D21-K601L)	50	1000
130' ELEVATION NORTH AREA		
11 TIP Area (1D21-K601F)	50	1000
12 North CRD HCU (1D21-K601P)	50	1000
13 TIP Probe Drives Area (1D21-K601U)	100	1000

31EO-EOP-014-1, SC/RR EOP Flowchart

SC - SECONDARY CONTAINMENT CONTROL

Area water level above Table 5 Maximum Normal Operating Water Level

Area or HVAC exhaust radiation level above Table 6 Maximum Normal Operating Radiation Level

Differential pressure at or above 0 in. of water

Table 6 SECONDARY CONTAINMENT OPERATING RADIATION LEVELS

HVAC EXHAUST RADIATION ANNUNCIATORS on 1H11-P601	Max Normal Operating Value mR/hr	Max Safe Operating Value mR/hr
	HI-HI Radiation Annunciator	N/A
- RB BLDG POT CONTAM AREA (1D11-K609 A-D)	18	N/A
- REFUELING FLOOR VENT EXHAUST (1D11-K611 A-D)	18	N/A
AREA RADIATION MONITORS on 1H11-P600, 1D21-P600	**Max Normal Operating Value mR/hr**	**Max Safe Operating Value mR/hr**
REFUEL FLOOR AREA		
1 Reactor head laydown area (1D21-K601A)	50	1000
2 Refueling Floor Stairway (1D21-K601B)	50	1000
3 Refueling Floor (1D21-K601D)	50	1000
4 Drywell Shield Plug (1D21-K601E)	50	1000
5 Spent Fuel Pool & New Fuel Storage (1D21-K601M)	50	1000

3.7 PLANT SYSTEMS

3.7.4 Main Control Room Environmental Control (MCREC) System

LCO 3.7.4 Two MCREC subsystems shall be OPERABLE.

---NOTE---
The main control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One MCREC subsystem inoperable.	A.1 Restore MCREC subsystem to OPERABLE status.	7 days
B. Two MCREC subsystems inoperable due to inoperable control room boundary in MODE 1, 2, or 3.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

(continued)

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1

LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, LCO 3.0.7, and LCO 3.0.8.

LCO 3.0.2

Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.

LCO 3.0.3

When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:

a. MODE 2 within 7 hours;

b. MODE 3 within 13 hours; and

c. MODE 4 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required.

LCO 3.0.3 is only applicable in MODES 1, 2, and 3.

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

(continued)

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

management actions, if appropriate; exceptions to this specification are stated in the individual Specifications, or

a. 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.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY, or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the required testing.

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, additional evaluations and limitations may be required in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is 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.

LCO 3.0.7

Special Operations LCOs in Section 3.10 allow specified Technical Specifications (TS) requirements to be changed to permit performance of special tests and operations. Unless otherwise specified, all other TS requirements remain unchanged. Compliance with Special Operations LCOs is optional. When a Special Operations LCO is desired to be met but is not met, the ACTIONS of the Special Operations LCO shall be met. When a Special Operations LCO is not desired to be met, entry into a MODE or other specified condition in the Applicability shall only be made in accordance with the other applicable Specifications.

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84. 295004AA2.02 001

Unit 2 was operating at 50% power when the 125VDC Distribution Cabinet 2D, 2R25-S129, was lost and the following annunciator was received:

o 602-110, ECCS/RPS DIVISION 1 TROUBLE

The SRO is in the process of identifying which analog transmitter trip system (ATTS) units were affected and is performing a loss of safety function determination IAW the Technical Requirements Manual (TRM).

Which ONE of the following identifies the affected analog transmitter trip units and describes the TRM Loss of Function Diagrams (LFD)?

Two _________ ATTS cabinets will be de-energized.

The loss of function statement found at the bottom of the LFD identifies the channel combinations which are __________ .

A. RPS;
REQUIRED to be operable to maintain the safety function

B. RPS;
NO longer available for the safety function

C.✔ ECCS;
REQUIRED to be operable to maintain the safety function

D. ECCS;
NO longer available for the safety function

Description:

Power Distribution to the ATTS panels.

a. RPS Bus A supplies 120 VAC power to ATTS panels P921 and P923.
b. RPS Bus B supplies 120 VAC power to ATTS panels P922 and P924.
c. 125 VDC Bus A supplies power to ATTS panels P925 and P927 from panel 2R25 S129 (1R25-S105 for Unit 1).
d. 125 VDC Bus B supplies power to ATTS panels P926 and P928 from panels 2R25 S002 and 2R25 S130 (1R25 S106 for Unit 1). One power supply in both ATTS panels P926 and P928 receives its power from 2R25 S002 and the other power supply in each ATTS panel receives its power from 2R25 S130.

Note: The loss of function statement typically found at the bottom of the LFD identifies the channel combinations required to be operable in order for instrument function

The SRO must know the purpose of the LFD.
TRM 11.0 LOSS OF FUNCTION DIAGRAMS

A. Purpose

Loss of Function Diagrams (LFDs) provide a means for evaluating the affects of the loss of one or more instrument channels on the capability of the associated instrument logic to perform its intended safety function. In fulfilling this purpose, the LFDs provide the following:

- o The number of channels associated with a given instrument function.
- o The configuration of the instrument channels in the trip systems.
- o The number and combinations of channels required to be operable in order for instrument function capability to be maintained.

The "A" distractor is plausible since RPS ATTS cabinets are similar to the ECCS cabinets therefore if the internal power suppplies inside the ATTS cabinets associated with RPS were powered from 125VDC Distribution Cabinet 2D, 2R25-S129. The second half is correct.

The "B" distractor is plausible since RPS ATTS cabinets are similar to the ECCS cabinets therefore if the internal power suppplies inside the ATTS cabinets associated with RPS were powered from 125VDC Distribution Cabinet 2D, 2R25-S129. The second half is plausible if applicant does not understand the content provided in LFDs.

The "D" distractor is plausible because the first half is correct. The second half is plausible if applicant does not understand the content provided in LFDs.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

295004 Partial or Complete Loss of D.C. Power

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.10 / 43.5 / 45.13)

AA2.02 Extent of partial or complete loss of D.C. power 3.5 3.9

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the TS and their bases for the TRM

LESSON PLAN/OBJECTIVE:
H11-ATTS-LP-10008, **Ver. 4.0**/EO 055.001.A.05

References used to develop this question:

34AR-602-110-2, ECCS/RPS Division 1,**Ver. 3.1**
TRM Section 11.0, Loss of Function Diagrams,**Rev. 13**

Item 1: SRO ONLY Guideline
Item 2: 34AR-602-110-2, **Ver. 3.1**
Item 3: H11-ATTS-LP-10008 LP, page 11, **Ver. 4.0**
Item 4: Item 4 TRM T 11.0 LFDs, page T11.0-1, **Rev. 13**

Bank Question used on HLT-3 NRC Exam Q#84

ORIGINAL QUESTION (HLT-3 NRC Exam Q#84)

Unit 2 was operating at 50% power when the 125VDC Distribution Cabinet 2D, 2R25-S129 was lost and the following annunciator was received:

ECCS/RPS DIVISION 1 TROUBLE (602-110)

The SRO is in the process of identifying which analog transmitter trip system (ATTS) units were affected and is performing a loss of safety function determination in accordance with the Technical Requirements Manual (TRM).

Which ONE of the following identifies the affected analog transmitter trip units and describes the TRM Loss of Function Diagrams (LFD)?

A. Two RPS ATTS cabinets will be de-energized.

The loss of function statement found at the bottom of the LFD identifies the channel combinations which are no longer available for the safety function.

B. Two RPS ATTS cabinets will be de-energized.

The loss of function statement found at the bottom of the LFD identifies the channel combinations required to be operable in order to maintain the safety function.

C.✓ Two ECCS ATTS cabinets will be de-energized.

The loss of function statement found at the bottom of the LFD identifies the channel combinations required to be operable in order to maintain the safety function.

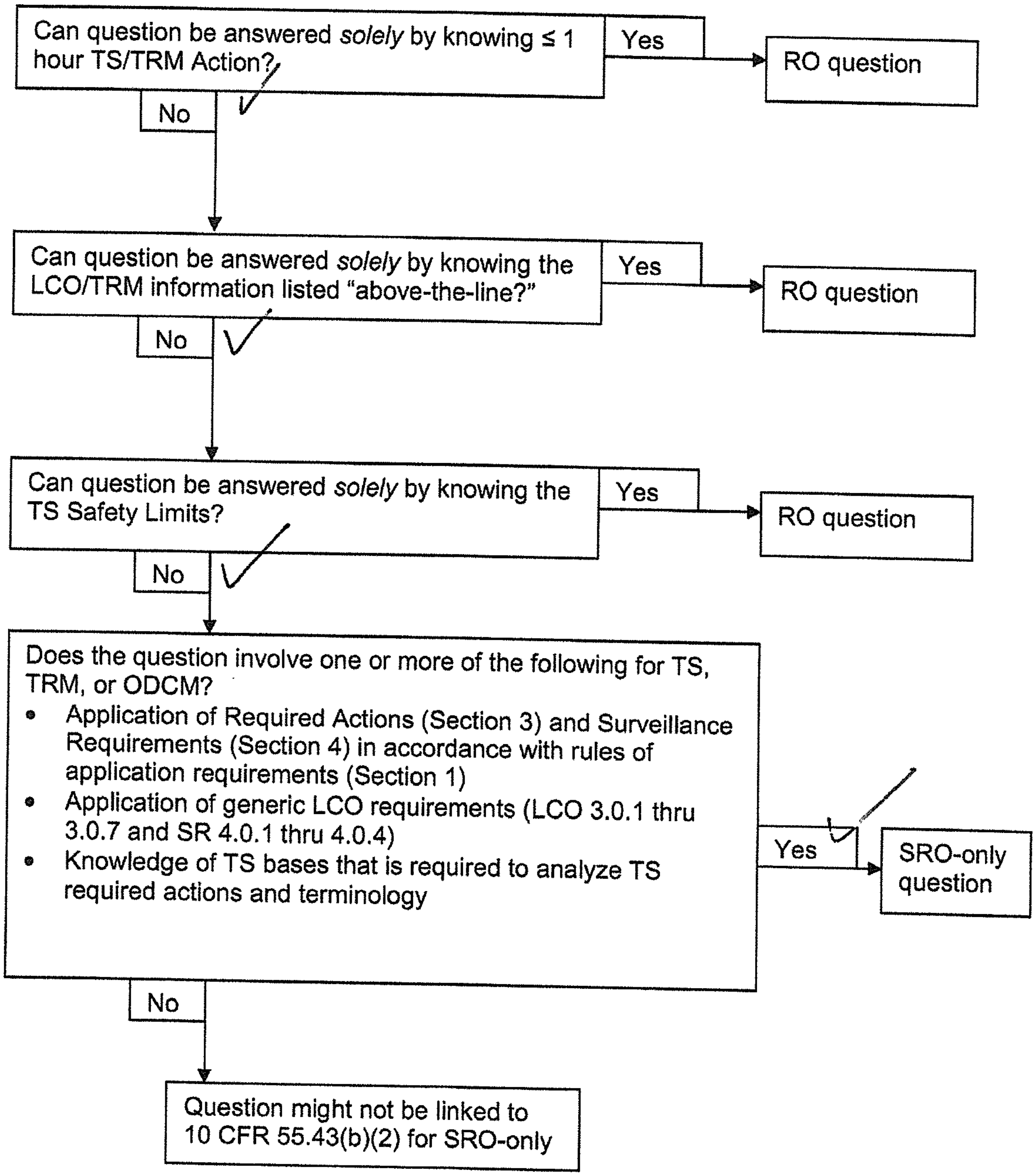
D. Two ECCS ATTS cabinets will be de-energized.

The loss of function statement found at the bottom of the LFD identifies the channel combinations which are no longer available for the safety function.

Q#84 K/A 295004AA2.02

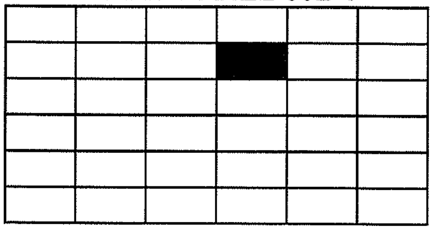
Clarification Guidance for SRO-only Questions
Rev 1 (03/11/2010)

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



1.0 IDENTIFICATION:

ALARM PANEL 602-1



ECCS/RPS

DIVISION I

TROUBLE

DEVICE: N/A

SETPOINT: N/A

2.0 CONDITION:

ATTS Panel 2H11-P921 OR 2H11-P923 OR 2H11-P925 OR 2H11-P927 has a trouble condition.

3.0 CLASSIFICATION:

AUXILIARIES

4.0 LOCATION:

2H11-P602 Panel 602-1

5.0 OPERATOR ACTIONS:

5.1 Confirm a surveillance OR calibration is NOT being performed on any trip unit. ☐

5.2 Confirm that no DIV. I ATTS panel red CARD-OUT-OF-FILE indicating light is ILLUMINATED. ☐

5.3 Confirm no DIV. I ATTS panel alarm TEST switch is DEPRESSED. ☐

5.4 Confirm both white POWER ON indicating lights for each DIV. I ATTS Panel are ILLUMINATED. ☐

5.5 Confirm no red TRIP UNIT GROSS FAILURE light is ILLUMINATED. ☐

6.0 CAUSES:

6.1 One of the trip units is in calibration mode

6.2 One of the panel test switches is depressed

6.3 One of the trip units has a card-out-of-file

6.4 One of the trip units is in gross failure

6.5 Loss of one or both panel power supplies

7.0 REFERENCES:

H-24401 thru H-24435, ATTS Elementary

8.0 TECH. SPECS./TRM/ODCM/FHA:

8.1 TS 3.3.1.1 Reactor Protection System Instrumentation

8.2 TS 3.3.5.1 ECCS Instrumentation

c. Remote meter (optional)

d. Up to seven (7) slave trip units (optional)

2. The slave trip units are required if more than one actuation/trip signal is desired from that master trip unit. Each slave trip unit can drive one additional trip relay, thus providing one additional actuation/trip signal per Slave Unit.

3. The master trip unit also provides a high and a low input gross fail trip. If the input from the transmitter fails either high or low, the master trip unit will initiate a gross fail trip, which will alert the operator to a possible transmitter failure.

C. The ATTS Major Divisions

The ATTS is divided up into two major divisions; the Reactor Protection System (RPS) division and the Emergency Core Cooling System (ECCS) division.

1. The Reactor Protection System (RPS) design is a two-division system (A and B) where each division has dual monitoring to meet redundancy criteria. Thus, this system must have four individual cabinets (P921, P922, P923, and P924) to assure separation between the redundant hardware. The four channels of the two divisions are identified as 1A, 2A, 1B, and 2B, with hardware for each channel housed in separate cabinets. The RPS division provides actuation and trip signals to the following:

a. RPS (scram signals)

b. PCIS (Groups 1, 2, 5, 6, 10, and 12)

c. Secondary Containment Isolation System

2. The Emergency Core Cooling System (ECCS) design is a two-division system (Div 1, Div 2) with hardware mounted in four separate cabinets (P925, P926, P927, and P928) to provide the required separation between divisions. Each cabinet houses trip units and trip relays which provide actuation and trip signals to the following:

a. PCIS for HPCI and RCIC (Groups 3 and 4)

b. HPCI

c. RCIC

d. Core Spray

e. RHR

T 11.0 LOSS OF FUNCTION DIAGRAMS

A. Purpose

Loss of Function Diagrams (LFDs) provide a means for evaluating the affects of the loss of one or more instrument channels on the capability of the associated instrument logic to perform its intended safety function. In fulfilling this purpose, the LFDs provide the following:

- The number of channels associated with a given instrument function.
- The configuration of the instrument channels in the trip systems.
- The number and combinations of channels required to be operable in order for instrument function capability to be maintained.

B. General Rules for Use:

- LFDs are "channel-based," that is, they are designed to be used to determine instrument function capability given a loss of one or more channels. For the purposes of determining loss of function, the LFDs show what constitutes a channel. However, in identifying the beginning and end of a channel for the purpose of determining channel functional test scope, the LFD should not be used for this purpose; instead, the TRM definition "Channel Functional Test Scope" should be used.
- As in typical elementary logic, the energy trace is from the sensor to the actuated device. Consequently, inoperability of a component in the energy trace can directly or indirectly affect the ability of a downstream component in the trace to function. However, the opposite is not always true; that is, the downstream component since it does not provide input to the upstream component does not affect the ability of the upstream component to function. As such, loss of a component anywhere other than in the channel cannot in all cases be traced back to evaluate the affect of the loss on a channel(s). Consequently, since the LFDs are "channel-based," in such cases, the LFD cannot be used to determine instrument function capability. Instead, the elementary logic must be consulted to determine the affect of the loss on the supported system.
- LFDs are designed to be used with the instrumentation specifications found in the Technical Specifications, the TRM, and the ODCM. Typically, an LFD is provided for each instrumentation specification line item. However, some instruments provide more than one instrument function and an LFD may not provide sufficient information to ascertain all of the functions provided by the instrument. In order to identify all instrument functions performed by a particular instrument, Table 10.1-1, Master Equipment Cross Reference, Sorted by MPL, must be consulted. For a given MPL, this sort will identify all LFDs for the instrument functions that are served by the instrument.
- The complete logic from sensor to the actuation logic/actuated device is not reflected in the LFDs. A dashed line is used to denote cases where the logic was not included. Elementary diagrams used to develop the LFD are referenced on the LFD in the event information on the omitted logic is needed.

85. 295007AA2.01 001

Unit 2 is operating at 100% RTP, when a reactor pressure transient occurs resulting in the following:

- o 603-114, REACTOR VESSEL PRESSURE HIGH illuminates

Subsequently, Drywell Floor Drain leakage increases to 55 gpm

The REACTOR VESSEL PRESSURE HIGH alarm setpoint is ____________ psig.

IAW NMP-EP-110, Emergency Classification Determination and Initial Action, the HIGHEST Emergency Classification that will be declared based on Drywell Floor Drain leakage is __________ .

Reference Provided

A. 1060;
an Alert

B. 1060;
a Notification of Unusual Event

C.✔ 1055;
an Alert

D. 1055;
a Notification of Unusual Event

Description:

603-114, REACTOR VESSEL PRESSURE HIGH
DEVICE:..SETPOINT:
2C32-R608..1055 PSIG increasing

LCO 3.4.10 The reactor steam dome pressure shall be . 1058 psig.

NMP-EP-110-GL02, Hot EAL Initiating Conditions
SU5- RCS Leakage
Unidentified or Pressure Boundary leakage > 10gpm
or
Identified leakage > 25 gpm

The SRO must know
NMP-EP-110-GL02, Fission Product Barrier
Potential Loss or Loss of Fuel Clad barrier or RCS barrier is an ALERT

The "A" distractor is plausible because it is above NOP 1045 psig and below the reactor scram set point 1074 psig. The second half is correct.

The "B" distractor is plausible because it is above NOP 1045 psig and below the reactor scram set point 1074 psig. Second half is plausible if the applicant confuses that the Containment barrier being lost as a NOUE and would be correct if asking Containment Barrier.

The "D" distractor is plausible because the first half is correct. Second half is plausible if the applicant confuses that the Containment barrier being lost as a NOUE and would be correct if asking Containment Barrier.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
FISSION PRODUCT BARRIER "RCS" PORTION ONLY

K/A:

295007 High Reactor Pressure

AA2. Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE : (CFR: 41.10 / 43.5 / 45.13)

AA2.01 Reactor pressure 4.1 4.1

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

<u>LESSON PLAN/OBJECTIVE:</u>
B11-RXINS-LP-04404,**Ver. 6.0/** EO 200.002.A.12

<u>References used to develop this question:</u>

34AR-603-114-2, Reactor Vessel Pressure High, **Ver. 3.1**
U2 TS 3.4.10, Reactor Steam Dome Press, **Amend. 210**
NMP-EP-110-GL02 – HNP EALs - ICs, Threshold Values and Basis, **Ver. 2.0**

Item 1: SRO ONLY Guideline
Item 2: 34AR-603-114-2, **Ver. 3.1**
Item 3: U2 TS 3.4.10, page 3.4-25, **Amend. 210**
Item 4: FPB "RCS" Portion Only, **Ver. 2.0**

Modified from bank question used on HLT-3 NRC Exam Q#42

ORIGINAL QUESTION (HLT-3 NRC Exam Q#42)

Unit 2 is at 99% power, ascending to rated power following a plant startup when the following alarm is received:

REACTOR VESSEL PRESSURE HIGH (603 -114)

Which ONE of the following identifies the alarm setpoint and the required EHC pressure set?

The alarm setpoint is ____________ psig.

For this power level, EHC pressure set should be set to approximately _________ psig.

A✓ 1055, 945
B. 1055,1040
C. 1064, 945
D. 1064, 1040

Q#85 K/A 295007AA2.01

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 ✓ ↓

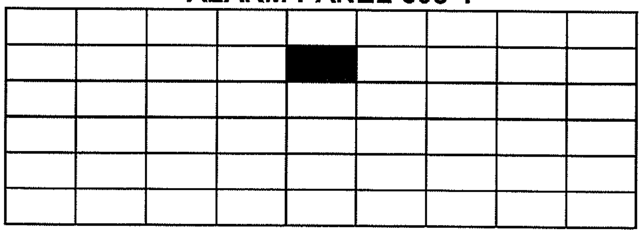
Does the question require one or more of the following?

- 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

1.0 IDENTIFICATION:

ALARM PANEL 603-1



REACTOR VESSEL
PRESSURE
HIGH

DEVICE:
2C32-R608

SETPOINT:
1055 PSIG increasing

2.0 CONDITION:

Reactor vessel pressure is at or above 1055 PSIG.

3.0 CLASSIFICATION:
AUXILIARIES

4.0 LOCATION:
2H11-P603 Panel 603-1

5.0 OPERATOR ACTIONS:

5.1 Confirm validity of the alarm using any of the following panel 2H11-P603 indicators:

5.1.1 2C32-R609, Rx Press/Turb Stm Flow recorder. ☐

5.1.2 2C32-R605C, Rx Press indicator. ☐

5.2 REDUCE power to prevent further pressure increase per 34G0-OPS-005-2, Power Changes. ☐

5.3 IF the MSIVs are OPEN, REDUCE pressure by lowering the pressure control setpoint. ☐

5.4 IF the MSIVs are CLOSED, REDUCE pressure by:

5.4.1 Utilizing the main steam line drains to the main condenser. ☐

5.4.2 Running the HPCI or RCIC systems in the Reactor Pressure Control Mode. ☐

5.4.3 Manually operating the safety relief valves. ☐

NOTE	Opening of any SRV with reactor pressure > 1074 PSIG will activate the low-low set (LLS) logic. The logic, once activated, will open all four 2B21-F013B, 2B21-F013D, 2B21-F013F, 2B21-F013G, LLS / Manual Relief Vlv's. These valves will open and close automatically to maintain pressure between 1036 and 851 PSIG.

6.0 CAUSES:

6.1 MSIV closure
6.2 Pressure control setpoint set too high
6.3 EHC System malfunction

7.0 REFERENCES:

7.1 H-27519 thru H-27524, FW Control System Elem.
7.2 57CP-CAL-029-2, GE/RMAX/Bailey Recorders

8.0 TECH. SPECS./TRM/ODCM/FHA:

N/A - Not applicable to this procedure

34AR-603-114-2
VER. 3.1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1058 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1058 psig.	In accordance with the Surveillance Frequency Control Program

NMP-EP-110-GL02, Hot Chart

<u>SU5</u> - RCS Leakage (Pg. 62)
1. UNIDENTIFIED **<u>OR</u>** pressure boundary leakage greater than 10 gpm. **<u>OR</u>** 2. IDENTIFIED leakage greater than 25 gpm.

NMP-EP-110-GL02, FPB Chart

ALERT	UNUSUAL EVENT
FA1 ANY Loss or Potential Loss of **EITHER** Fuel Clad **OR** RCS Barrier	**FU1** ANY Loss or Potential Loss of Containment Barrier

Potential Loss
3. RCS Leak Rate (Pg. 39) RCS leakage GREATER THAN 50 gpm inside the drywell **OR** Unisolable primary system leakage outside drywell as indicated by Secondary Containment operating temperatures or radiation levels above Max. Normal Operating Values (SC - Secondary Containment Control Flowchart – Table 4 & Table 6) 

86. 295012G2.4.11 001

Unit 2 was operating at 100% RTP when a loss of Drywell cooling occurred.

IAW 34AB-T47-001-2, Complete Loss of Drywell Cooling:

The crew is required to enter ___________ when any peak temperature listed in Attachment 1 has been exceeded for a MINIMUM of __________ .

A. 34GO-OPS-013-2, Normal Plant Shutdown;
1 hour

B. 34GO-OPS-013-2, Normal Plant Shutdown;
30 minutes

C. 34GO-OPS-014-2, Fast Reactor Shutdown;
1 hour

D✓ 34GO-OPS-014-2, Fast Reactor Shutdown;
30 minutes

Description:

34AB-T47-001-2 "Complete Loss of DW Cooling" contains a subsequent action that if any of the temperatures are exceeded in Attachment 1, then a 30 minute clock starts for restoring temperatures. If this time limit is exceeded, then a fast reactor shutdown will be initiated per 34GO-OPS-014-2.

Tech Spec 3.6.1.1, "Primary Containment Operability"

- o Restore primary containment to OPERABLE status within 1 hour
- o Mode 3 in 12 hours (a normal shutdown)
- o Mode 4 in 36 hours

The SRO must have knowledge of when to implement attachment 1.

The "A" distractor is plausible if the candidate thinks the high temperature affects Tech Spec 3.6.1.1, "Primary Containment Operability" (the actual design limit is 340° F). The LCO would require containment to be restored to operable status in 1 hour, Mode 3 in 12 hours, Mode 4 in 36 hours (a normal shutdown).

The "B" distractor is plausible if the candidate thinks the high temperature affects Tech Spec 3.6.1.1, "Primary Containment Operability" (the actual design limit is 340° F). The LCO would require containment to be restored to operable status in 1 hour, Mode 3 in 12 hours, Mode 4 in 36 hours (a normal shutdown). The second half is correct.

The "C" distractor is plausible because the first half is correct. The second half is plausible if the candidate thinks the high temperature affects Tech Spec 3.6.1.1, "Primary Containment Operability" (the actual design limit is 340° F). The LCO would require containment to be restored to operable status in 1 hour.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

K/A:

295012 High Drywell Temperature

2.4.11 Knowledge of abnormal condition procedures.
(CFR: 41.10 / 43.5 / 45.13) 4.0 4.2

SRO only because of link to 10CFR55.43(b)(5):
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

LESSON PLAN/OBJECTIVE:
P64-PCCCW-LP-01304,**Ver. 4.0**/EO 200.032.A.01

References used to develop this question:
34AB-T47-001-2, Complete Loss of Drywell Cooling, **Ver. 1.10**
Item 1: SRO ONLY Guideline
Item 2: 34AB-T47-001-2, page 3, **Ver. 1.10**

Bank question used on HLT-5 NRC Exam Q#87

ORIGINAL QUESTION (HLT-5 NRC Exam Q#87)

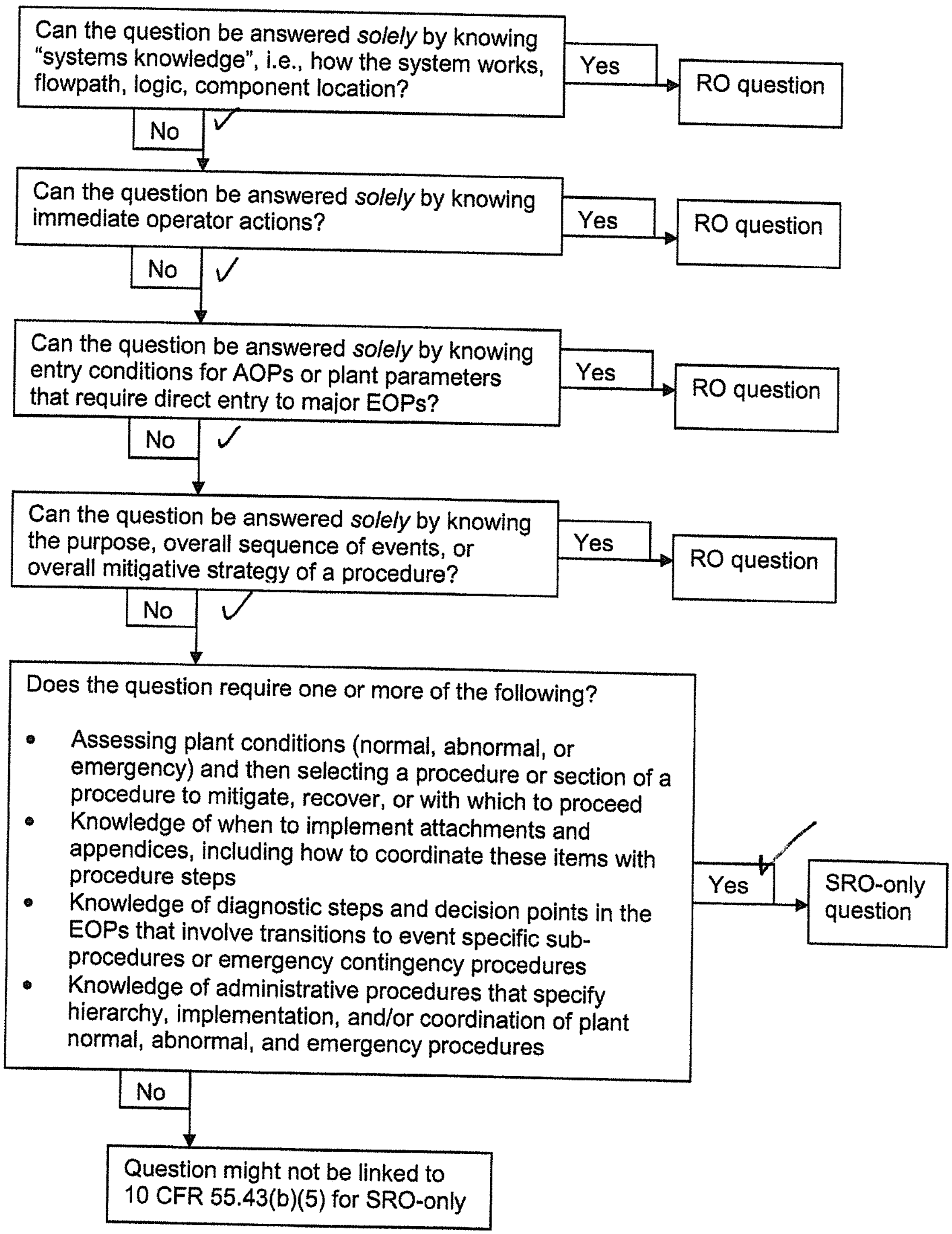
Which ONE of the choices below completes the following statement IAW 34AB-T47-001-2, Complete Loss of Drywell Cooling, Attachment 1, Peak Drywell Temperature?

The crew is required to enter __________ when any peak temperature listed in Attachment 1 has been exceeded for at least __________ .

A. 34AB-C71-001-2, Reactor Scram Procedure;
1 hour
B. 34AB-C71-001-2, Reactor Scram Procedure;
30 minutes
C. 34GO-OPS-014-2, Fast Reactor Shutdown;
1 hour
D✓. 34GO-OPS-014-2, Fast Reactor Shutdown;
30 minutes

Q#86 K/A 295012 G2.4.11

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



4.0 SUBSEQUENT OPERATOR ACTIONS

4.1 Monitor Drywell pressure and temperature during performance of this procedure and enter Technical Specifications as required. ☐

4.1.1 IF pressure approaches 0.75 PSIG,
vent the Drywell in accordance with 34SO-T48-002-2, Containment Atmospheric Control and Dilution System. ☐

4.1.2 IF average air temperature exceeds 150°F,
correct reactor water level indications as necessary for high Drywell temperatures per 34AB-B21-002-2, RPV Water Level Corrections. ☐

4.1.3 IF average air temperature exceeds 150°F,
enter 31EO-EOP-012-2, Primary Containment Control Flowchart. ☐

4.1.4 IF ANY peak Drywell temperature listed in Attachment 1
is exceeded for more than 30 minutes,
enter 34GO-OPS-014-2, Fast Reactor Shutdown. ☐

4.2 Select the appropriate condition from those listed below and perform the actions of the indicated step concurrently.

- Loss of drywell cooling fans — Step 4.3 ☐
- Loss of chilled water — Step 4.4 ☐

4.3 LOSS OF DRYWELL COOLING FANS

4.3.1 Perform the following at panels 2H11-P654 and 2H11-P657:

4.3.1.1 PLACE the control switches for all non-running in service fans in OFF. ☐

4.3.1.2 PLACE the control switches for all standby fans in RUN and
confirm fans START. ☐

4.3.1.3 IF fans did NOT start,
PLACE the control switches for non-running fans in OFF. ☐

4.3.2 Confirm 2R24-S011 and 2R24-S012 MCC's are energized. ☐

4.3.3 IF 2R24-S011 and/or 2R24-S012 MCC's are de-energized
attempt to restore by closing supply breaker on 2H11-P652 panel. ☐

4.3.4 Enter 34AB-R23-001-2, Loss of 600 Volt Emergency Bus, IF required. ☐

87. 295016G2.2.37 001

An event results in the Main Control Room being abandoned.

Control of **Unit 2** is established at the Unit 2 Remote Shutdown Panel (RSDP).

31RS-OPS-001-2, Shutdown From Outside Control Room, is in progress.

- o ALL RSDP Emergency Transfer Switches are in the "EMERGENCY" position

Subsequently, Unit 2 Drywell pressure increases to 3.0 psig.

A SO reports the following:

- o RHR pump 2A is NOT running

Maintenance reports RHR pump 2A Lockout Relay has TRIPPED.

The procedure that contains the guidance for whose AUTHORITY is required to reset the RHR pump 2A Lockout Relay is __________ .

With the RSDP Emergency Transfer Switches in the "EMERGENCY" position, RHR pump 2B is __________ .

A. 30AC-OPS-003, Plant Operations;
operable

B. 30AC-OPS-003, Plant Operations;
inoperable BUT available

C. 31GO-OPS-021, Manipulation of Controls and Equipment;
operable

D✓ 31GO-OPS-021, Manipulation of Controls and Equipment;
inoperable BUT available

Description:

31GO-OPS-021-0, step 7.3.1 states Lock-out relays and flags on protective relays that trip lock-out relays will NOT be reset UNTIL authorized by the **SS** and one of the following:

Shift Manager (SM) or higher
Engineering Supervisor or higher
Maintenance Team Leader (TL) (Supervisor)(Electrical) or higher

31RS-OPS-001-2, Shutdown From Outside Control Room on RHR system operation:

1.2 With transfer switch 2C82-S9 in the EMERG position, the 'B' RHR pump will NOT auto start on any of the LOCA signals

TS BASES 3.5.1 ECCS Operating
Each ECCS injection/spray subsystem are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems, each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started. RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. RHR pump 2B is inoperable but available because with RHR pump 2B controlled from the RSDP, the pump will not auto start and perform its intended function.

The SRO must have detailed knowledge of the authorization requirements to reset the lockout. An RO will know that the relay is required to have authorization, but the SRO will know who can and cannot authorize the reset. Since this procedure has different requirements for different types of relays, it will take additional knowledge from the SRO to answer this question.

The "A" distractor is plausible since the guidance to reset lockout relays was previously in this procedure and was recently changed to the new procedure. The second half is plausible if the student forgets the auto start feature is defeated.

The "B" distractor is plausible since the guidance to reset lockout relays was previously in this procedure and was recently changed to the new procedure The second half is correct.

The "C" distractor is plausible because The first half is correct. The second half is plausible if the student forgets the auto start feature is defeated.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

K/A:

295016 Control Room Abandonment

2.2.37 Ability to determine operability and/or availability of safety related equipment.
(CFR: 41.7 / 43.5 / 45.12) 3.6 4.6

SRO only because of link to 10CFR55.43(b)(5):Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures.

LESSON PLAN/OBJECTIVE:
C82-RSDP-LP-05201, Remote Shutdown Panel (RSDP),**Ver. 3.0**

References used to develop this question:
31GO-OPS-021-0, Manipulation of Controls and Equipment, **Ver.4.1**
31RS-OPS-001-2, Shutdown From Outside Control Room, **Ver. 6.20**
TS BASES 3.5.1 ECCS Operating, **Rev. 13/20**

Item 1: SRO ONLY Guideline
Item 2: 31GO-OPS-021-0, page 8, **Ver.4.1**
Item 3: 31RS-OPS-001-2, page 29, **Ver. 6.20**
Item 4: U2 TS BASES 3.5.1, page , **Rev. 13/20**

Modified from HLT-7 NRC Exam Q#79

ORIGINAL QUESTION (HLT-7 NRC Exam Q#79)

Unit 2 has automatically scrammed due to a small steam leak in the Drywell.

RHR "A" Loop is in Torus Spray Mode per the EOPs.

The following annunciator is received and RHR pump 2A trips:

601-212, RHR Pump A OVLD/LOCKOUT relay trip

Subsequently, the cause of the RHR "A" pump trip is identified and repaired.

With the above listed alarm, which ONE of the choices below completes the following statements?

The procedure that contains the guidance for whose authority is required to reset lockout relays and relay targets is __________ .

Of the listed individuals, the MINIMUM Authorization REQUIRED to reset the RHR "A" LOCKOUT relay is the Shift Supervisor AND any _______ .

A. 31GO-OPS-021, Manipulation of Controls and Equipment;
Work Control Center Supervisor

B.✓ 31GO-OPS-021, Manipulation of Controls and Equipment;
Plant Engineering Supervisor

C. 30AC-OPS-003, Plant Operations;
Work Control Center Supervisor

D. 30AC-OPS-003, Plant Operations;
Plant Engineering Supervisor

Q#87 K/A 295016 G2.2.37

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

7.3 RESET OF LOCK-OUT RELAYS AND RELAY TARGETS

INFORMATION

NOTE:	The requirements of 7.3.1 do NOT apply IF the reset of the lockout relay is directed by other procedures and there is no indication of an electrical fault on the affected equipment.

7.3.1 Lock-out relays and flags on protective relays that trip lock-out relays will NOT be reset
UNTIL authorized by the SS and one of the following:

- Shift Manager (SM) or higher
- Engineering Supervisor or higher
- Maintenance Team Leader (TL) (Supervisor)(Electrical) or higher

7.3.2 The Diesel Generator Loss of Off-Site Power Lock-out (LOSP) Relay may not be reset unless:

- authorized by the SS and one of the above listed staff support personnel
  OR
- authorized by the SS when the trip condition is known **OR** understood NOT to be an electrical fault or detrimental to the affected equipment.

7.3.3 The Recirculation Pump ASD Lock-out Relays
may be reset as authorized by the SS,
provided the trip cause has been investigated and considered NOT to be detrimental to the equipment.

7.3.4 Authorization will NOT be granted UNTIL corrective action is completed on any electrical fault or UNTIL the trip condition is known or understood NOT to be detrimental to the affected equipment.

7.3.5 Any member of the Maintenance staff (electrical), I&C staff, Operations staff, or Engineering Support staff who has been authorized by the SS may reset the relays and/or targets. Those persons conducting the resetting will inform the SS WHEN the activity is completed.

7.3.6 WHILE testing protective relays and schemes WHEN equipment or systems will NOT be affected, lock-out relays and relay targets may be reset as authorized by a member of the Maintenance group performing the test.

NOTE	Placing the Remote Shutdown Panel Transfer for RHR in Emergency has the following effects on RHR system operation: 1. The A, C, and D RHR pumps will automatically initiate in the LPCI mode on the following signals: 1.1 Reactor level ≥ -113 inches (actual setpoint = -101 inches) 1.2 High Drywell pressure ≤1.92 PSIG (actual setpoint = 1.85 PSIG). With transfer switch 2C82-S9 in the EMERG position, the 'B' RHR pump will NOT auto start on any of the above signals 2. Load shed AND overcurrent are still valid trip signals. 3. 2E11-F006A, 2E11-F006B, 2E11-F006C, 2E11-F006D, Pump Suction Valves, are still interlocked with their respective 2E11-F004 valve. 4. 2E11-F004B, Pump suction from the Torus, AND 2E11-F024B, Torus cooling, must be closed to open the SDC suction, 2E11-F006B. However, once 2E11-F006B is open, 2E11-F024B may be reopened. 5. The loss of suction valve alignment trip is defeated for the 2B RHR pump. 6. 2E11-F008 AND 2E11-F009, SDC Isol Vlvs, closure on high Rx pressure (138 PSIG) AND low Rx level (+3") is defeated. 7. 2E11-F007B, Min Flow Vlv, operates automatically. 8. 2E11-F048B, 3 minute LOCA interlock is defeated. 9. The LOCA interlock for closing the following valves is defeated: 2E11-F011B, 2E11-F016B, 2E11-F028B, 2E11-F027B. 10. Interlock to prevent opening both 2E11-F015B AND 2E11-F017B, Inbd AND Outbd Inj Vlvs, with rx. pressure ≥ 425 PSIG is defeated. 11. Interlock to automatically open 2E11-F015B AND 2E11-F017B, Inbd AND Outbd Inj Vlvs, on a LOCA signal (-101 RWL AND 1.85 PSIG drywell pressure) with Rx. pressure ≤ 425 PSIG is defeated. 12. Interlock to automatically close 2E11-F015B, Inbd Inj Vlv IF in Shutdown Cooling AND receive a PCIS Group II signal (+3 RWL OR 1.85 PSIG Drywell pressure) OR Rx pressure ≥ 138 PSIG is defeated. 13. Interlock to automatically trip 2E11-C001B AND 2E11-C001D, RHRSW pumps, on a LOCA signal is defeated. 14. LOSP and breaker trips for 2E11-C001B AND 2E11-C001D, RHRSW pumps, are still in effect. 15. 2E11-F017B, RHR Outbd Inj Vlv, 5 minute LOCA interlock is defeated.

NOTE	An RHR pump discharge pressure of greater than or equal to 112 PSIG (127 PSIG actual setpoint) OR a Core Spray pump discharge pressure of greater than or equal to 137 PSIG (152 PSIG actual setpoint) is the final permissive for an automatic depressurization initiation IF the ADS two minute delay has elapsed.

BASES

BACKGROUND (continued)

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS equipment.

The CS System is composed of two independent subsystems (Ref. 1). Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started when AC power is available. When the RPV pressure drops sufficiently, CS System flow to the RPV begins. A full flow test line is provided to route water from and to the suppression pool to allow testing of the CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the corresponding recirculation loop. The two LPCI subsystems can be interconnected via the RHR System cross tie valve; however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (all pumps immediately if power is provided by the 2D Startup Auxiliary Transformer (SAT), and if power is provided by the 2C SAT or the DGs, C pump within 1 second after AC power is available, and A, B, and D pumps approximately 10 seconds after AC power is available). RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the recirculation loops. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the corresponding recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for the four LPCI pumps to route water from the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) consists of a steam driven turbine pump unit, piping, and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the feedwater system line, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for

(continued)

This question was one of the five SRO questions previously submitted for review

SRO question
3 of 5

<u>Question Sat</u>

No changes required

No changes were made

88. 295019G2.1.28 001

Unit 2 was operating at 100% RTP when an event occurred resulting in the following:

- o Normal pneumatic supply to the Unit 2 Drywell is lost
- o ALL High pressure & Low pressure injection systems will NOT operate

Section 7.3.1, Emergency Nitrogen Supply Operation, of 34SO-P70-001-2, Drywell Pneumatics System, has been completed and is supplying Emergency Nitrogen to the Drywell.

- o RWL is -186 inches and slowly decreasing

After performing section 7.3.1, Emergency Nitrogen Supply Operation, the MAXIMUM number of SRVs that will be supplied Nitrogen from the Emergency Nitrogen Bottles is __________ .

With RWL at -186 inches and decreasing, Reactor pressure is REQUIRED to be controlled using EOP flowchart ________ .

A. 11;
RC RPV Control (Non-ATWS), RC/P path

B. 11;
CP-1 Point F, Steam Cooling path

C. 5;
RC RPV Control (Non-ATWS), RC/P path

D.✓ 5;
CP-1 Point F, Steam Cooling path

Description:

Edwin, this was question 3 of 5 of the SRO questions that you have already reviewed. NO changes were made since your review

When Emergency Nitrogen bottles are aligned for SRV operation, manual valves 2P70 F021 and F023 are closed to limit nitrogen to one header. 5 SRVs are served by this header..

Note from 34SO-P70-001-2, DW pneumatics System: The SRVs now being supplied with Nitrogen are 2B21-F013C, 2B21-F013D, 2B21-F013G, 2B21-F013H, & 2B21-F013M. No Inboard MSIVs will have Nitrogen supplied.

With the current RWL (-186") and trend (decreasing), IAW the CP-1 with the answer to the 2 previous decision diamonds (table 8, 2a, 9 systems aligned and operating) being NO. The next block on the CP-1 flowchart is the steam cooling red flag. This red flag directs the SS to the override at E-1 on the RC/P leg of the RC flowchart. The override states "I f Steam Cooling is Required THEN Perform Steam Cooling". An arrow from the override directs the SS to exit the RC/P leg of the RC flow chart transition to CP-1 Point F.

The SS must remember that the red flag on the CP-1 flowchart is linked to the override on the RC/P leg of the RC flowchart to know that the RVP will be controlled by the Steam Cooling leg of CP-1 flowchart.

The "A" distractor is plausible if the student mistakenly believes that the emergency nitrogen bottles supplies both drywell pneumatic headers. The second half is plausible if the student does not know that the steam cooling red flag is linked to the override on the RC/P leg of the RC flow chart.

The "B" distractor is plausible if the student mistakenly believes that the emergency nitrogen bottles supplies both drywell pneumatic headers. The second half is correct.

The "C" distractor is plausible because the first half is correct. The second half is plausible if the student does not know that the steam cooling red flag is linked to the override on the RC/P leg of the RC flow chart.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

K/A:

295019 Partial or Complete Loss of Instrument Air

2.1.28 Knowledge of the purpose and function of major system components and controls.
(CFR: 41.7) 4.1 4.1

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

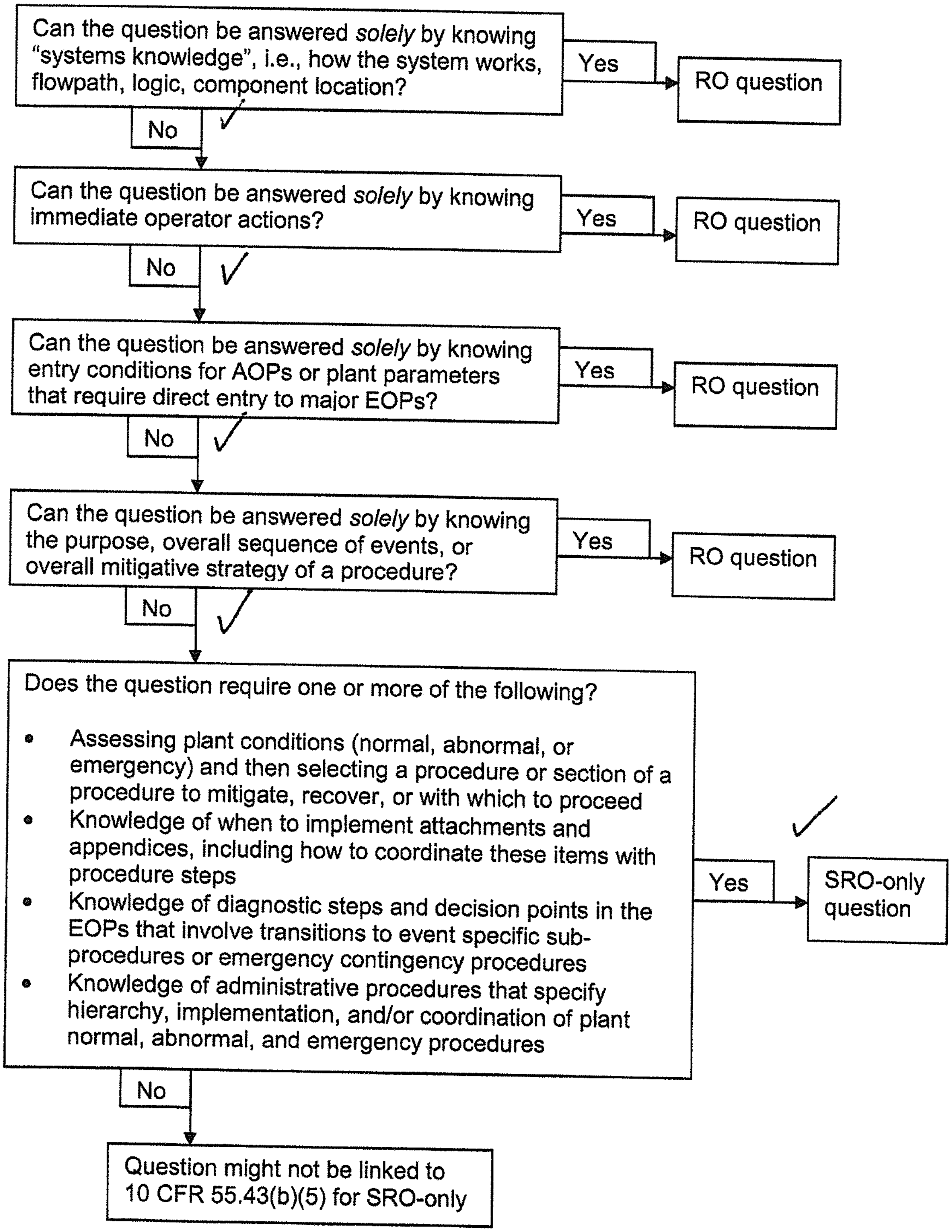
LESSON PLAN/OBJECTIVE:
P51-P52-P70-Plant Air-LP-03501,**Ver. 3.0**/EO 042.004.a.01

References used to develop this question:
34SO-P70-001-2, DW pneumatics System, **Ver. 10.7**
31EO-EOP-010-2, RPV CONTROL (NON ATWS),**Ver. 9.0**
31EO-EOP-015-2, CP-1 ALTERNATE LEVEL CONTROL, STEAM COOLING, & EMERGENCY RPV DEPRESSURIZATION,**Ver 8.0**

Item 1: SRO ONLY Guideline
Item 2: 34SO-P70-001-2, pages 8 & 9, **Ver. 10.7**
Item 3: U2 RC-P Override, **Ver 9.0**
Item 4: U2 CP-1, ALC path Steps, **Ver. 8.0**

Q#88 K/A 295019G2.1.28

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



7.3 INFREQUENT OPERATIONS

7.3.1 Emergency Nitrogen Supply Operation

CONTINUOUS

7.3.1.1 **Close** 2P70-F023, Drwl Pneu Sys Header Isol. ☐

7.3.1.2 **Close** 2P70-F021, Drwl Pneu Sys Header Isol, at 158RBR16. ☐

7.3.1.3 To place 2P70-A002A, Emergency Nitrogen Bottle, in service, **perform** the following:

NOTE:	All actions in this section are performed at 130RBR23 unless otherwise noted.

CAUTION:	DRYWELL PNEUMATIC HEADER PRESSURE MUST NOT EXCEED 120 PSIG. OVER PRESSURIZATION OF HEADER COULD RESULT IN INADVERTENT SRV ACTUATION OR PREVENT SRV ACTUATION IF REQUIRED. OPERATOR MUST CONSTANTLY MONITOR DRYWELL PNEUMATIC HEADER PRESSURE.

7.3.1.3.1 **Open** 2P70-F138A, 2P70-A002A Emergency Nitrogen Bottle Outlet Valve. ☐

7.3.1.3.2 **Confirm** 2P70-PCV-F140, Pressure Regulator, is adjusted to maintain 100-110 psig, as indicated on 2P70-PCV-F140. ☐

7.3.1.3.3 **Open** 2P70-F141, Emergency Nitrogen Bottles Pressure Control Valve, 2P70-F140, Isolation Valve. ☐

7.3.1.3.4 **Open** 2P70-F084, Emergency Nitrogen To Drywell Pneumatic Header Isolation Valve. ☐

7.3.1.3.5 **Confirm** Drywell Pneumatic System pressure is being maintained at 100-110 psig, as indicated on 2P70-PCV-F140. ☐

NOTE:	The SRVs now being supplied with Nitrogen are 2B21-F013C, 2B21-F013D, 2B21-F013G, 2B21-F013H, & 2B21-F013M. No Inboard MSIVs will have Nitrogen supplied.

7.3.1.3.6 WHEN 2P70-A002A, Emergency Nitrogen Bottle, pressure is less than 150 psig, **close** 2P70-F138A, Emergency Nitrogen Bottle, Outlet Valve, AND **open** next full Emergency Nitrogen Bottle Outlet Valve. ☐

7.3.1.3.7 **Close** 2P70-F139A, Header Isolation Valve, for the empty bottle. ☐

CAUTION:	FLEX HOSE CONTAINS GAS UNDER PRESSURE. SLOWLY LOOSEN FITTING AND LET DEPRESSURIZE PRIOR TO FULLY DISCONNECTING.

7.3.1.3.8 **Disconnect** the flex hose at 2P70-F138A. ☐

7.3.1.3.9 **Replace** depleted bottle. ☐

7.3.1.3.10 **Re-connect** flex hose at 2P70-F138A. ☐

7.3.1.3.11 **Open** 2P70-F139A for the replaced bottle AND **check** for leaks. ☐

7.3.1.3.12 IF leaks are present, **re-close** 2P70-F139A AND **repair** leaks. ☐

7.3.1.3.13 IF necessary, **reopen** 2P70-F139A. ☐

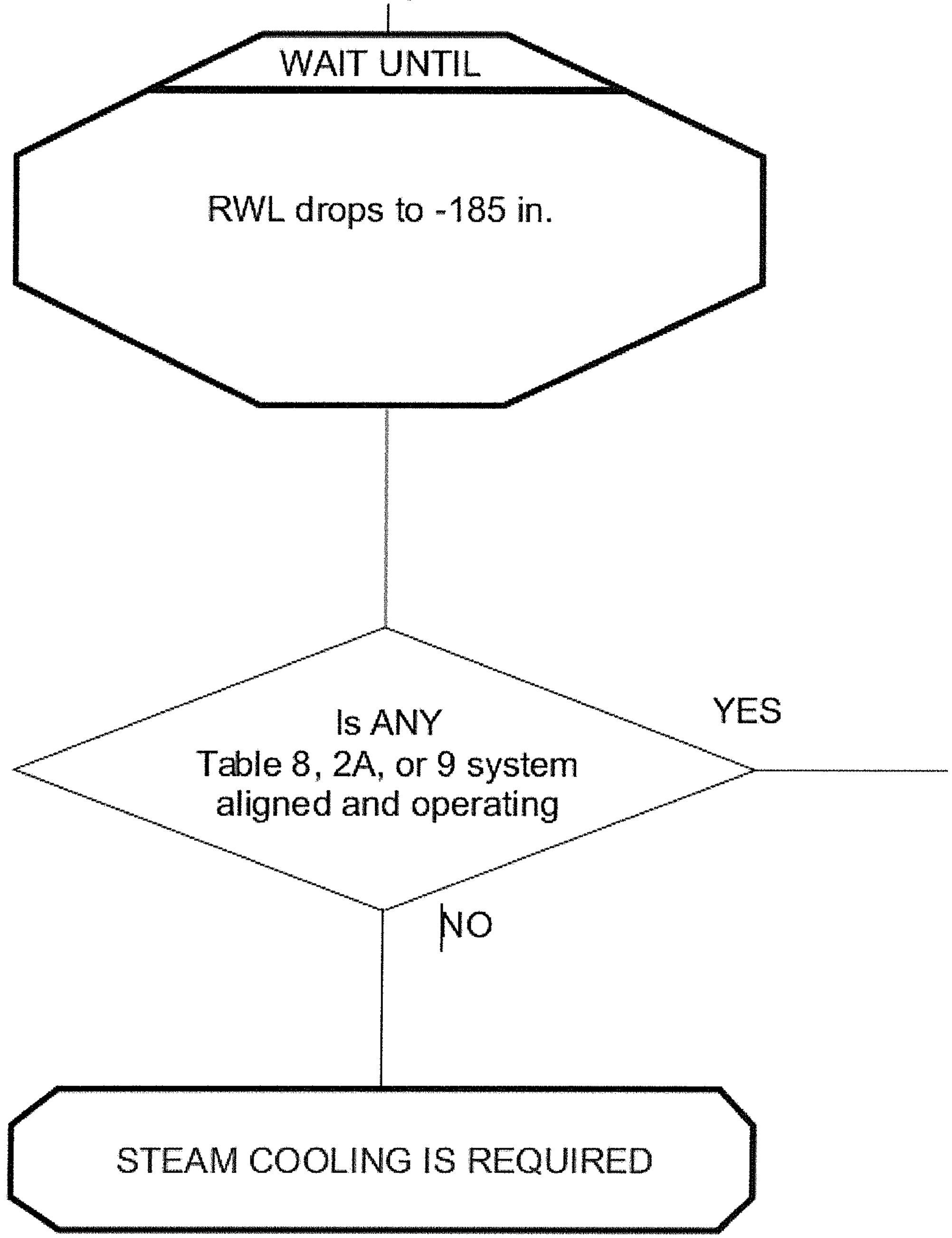
7.3.1.3.14 **Continue** exchanging bottles,
UNTIL Emergency Nitrogen is no longer required.
THEN **proceed to** step 7.3.1.6. ☐

31EO-EOP-010-2, RC RPV RC/P Path

WHILE PERFORMING THE FOLLOWING			
IF	torus water temperature CANNOT be maintained below the Heat Capacity Temperature Limit (Graph 2)	THEN	maintain reactor pressure below the limit, irrespective of the resulting cooldown rate.
IF	torus water level CANNOT be maintained below the SRV Tail Pipe Level Limit (Graph 6)	THEN	maintain reactor pressure below the limit, irrespective of the resulting cooldown rate.
IF	STEAM COOLING IS REQUIRED	THEN	perform Steam Cooling

GO TO CP-1
point F

31EO-EOP-015-2, CP-1 ALC Path



89. 295023AA2.02 001

Fuel movement is in progress on **Unit 1.**

Currently a fuel bundle is on the Main Grapple over the Fuel Pool area.

While over the Unit 1 Fuel Pool, the Main Grapple malfunctions releasing the irradiated fuel bundle and punctures the Fuel Pool liner.

Fuel Pool water level decreases and stabilizes at 22 feet.

The dropped fuel bundle is damaged and bubbles are observed floating to the surface.

Subsequently, a Secondary Containment isolation occurs due to the conditions on the Refuel Floor.

With the above Fuel Pool water level, LCO TS 3.7.8, Spent Fuel Storage Pool Water Level, __________ met.

IAW 31EO-EOP-0014-1, SC/RR, the Unit 1 Reactor Building HVAC fans _________ ALLOWED to be restarted.

A. is still;
are

B.✓ is still;
are NOT

C. is NOT;
are

D. is NOT;
are NOT

Description:

3.7 PLANT SYSTEMS

LCO 3.7.8 Spent Fuel Storage Pool Water Level

The spent fuel storage pool water level shall be ≥ 21 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

3.9 REFUELING OPERATIONS

LCO 3.9.6 Reactor Pressure Vessel (RPV) Water Level

RPV water level shall be ≥ 23 ft above the top of the irradiated fuel assemblies seated within the RPV.

Any one of the following will generate an isolation signal for the Unit 1 Reactor Zone Ventilation System:

a. Unit 1 or 2 Reactor Zone exhaust high radiation:
Unit 1: 18 mrem/hr on 1D11-K609 A-D
OR
Unit 2: 18 mrem/hr on 2D11-K609 A-D

b. Unit 1 or 2 Refueling Zone exhaust high radiation:

Unit 1. 18 mrem/hr on 1D11-K611-A-D
OR
Unit 2. 18 mrem/hr on 2D11-K611 A-D
OR
6.9 mrem/hr on 2D11-K634 A-D
OR
5.7 mrem/hr on 2D11-K635 A-D.

c. High drywell pressure (Either Unit): 1.85 psig

d. Low reactor water level (Either Unit): -35 inches

The only condition on the refueling flooring that could cause an isolation is high rads.

31EO-EOP-014-1, SC/RR Flowchart, Override
IF ANY Unit 1 or Unit 2 secondary containment HVAC exhaust radiation level exceeds the isolation setpoint (Table 14)
Then Confirm:
- o Unit 1 and Unit 2 Reactor Building HVAC isolation
- o Unit 1 and Unit 2 Refuel Floor HVAC isolation
- o Unit 1 and Unit 2 SBGT initiation

The SRO must be aware of the override on the SC/RR flowchart and determine its applicability.

The "A" distractor is plausible because the first half is correct. The second part is plausible if the student does not recognize that the only condition on the refueling flooring that could cause an isolation is high rads.

The "C" distractor is plausible if the student confuses the 23 ft RPV water level requirement during refueling operations (LCO 3.9.6) for the spent fuel storage pool water level limit. The second part is plausible if the student does not recognize that the only condition on the refueling flooring that could cause an isolation is high rads.

The "D" distractor is plausible if the student confuses the 23 ft RPV water level requirement during refueling operations (LCO 3.9.6) for the spent fuel storage pool water level limit. The second part is correct.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

295023 Refueling Accidents

AA2. Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : (CFR: 41.10 / 43.5 / 45.13)

AA2.02 Fuel pool level 3.4 3.7

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

LESSON PLAN/OBJECTIVE:
G41-FPC-LP-04501, Fuel Pool Cooling and Cleanup, **Ver. 5.0** /EO 200.076.A.01
HVAC-LP-01303, Secondary Containment HVAC Systems, **Ver. 2.0**

References used to develop this question:
LCO 3.7.8 Spent Fuel Storage Pool Water Level, **Amendment 266**
LCO 3.9.6 Reactor Pressure Vessel (RPV) Water Level, **Amendment 266**

Item 1: SRO ONLY Guideline
Item 2: U1 TS 3.7.8, page 3.7-19, **Amend. 266**
Item 3: U1 TS 3.9.6, **Amend. 266**

Fuel movement is in progress on Unit 1.

Currently a fuel bundle is on the Main Grapple.

o The Main Grapple is in the Normal Up position

Subsequently, the Unit 1 Main Steam line plugs fail causing the Reactor Cavity and Fuel Pool water levels to decrease.

Which ONE of the following completes these statements?

IAW 34AB-G41-002-1, Decreasing Rx Well/Fuel Pool Water Level, the grappled fuel bundle can be placed __________ in-core location.

When water level drops to the Main Steam lines, the fuel seated in the Fuel Pool racks will __________ .

A. into any;
still be covered

B. ONLY in its proper;
be uncovered

C. into any;
be uncovered

D✓. ONLY in its proper;
still be covered

Q# 89 K/A 295023AA2.02

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

3.7 PLANT SYSTEMS

3.7.8 Spent Fuel Storage Pool Water Level

LCO 3.7.8 The spent fuel storage pool water level shall be ≥ 21 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY: During movement of irradiated fuel assemblies in the spent fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Spent fuel storage pool water level not within limit.	A.1 ------------NOTE----------- LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of irradiated fuel assemblies in the spent fuel storage pool.	 Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.8.1	Verify the spent fuel storage pool water level is ≥ 21 ft over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.	In accordance with the Surveillance Frequency Control Program

3.9 REFUELING OPERATIONS

3.9.6 Reactor Pressure Vessel (RPV) Water Level

LCO 3.9.6 RPV water level shall be ≥ 23 ft above the top of the irradiated fuel assemblies seated within the RPV.

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.6.1	Verify RPV water level is ≥ 23 ft above the top of the irradiated fuel assemblies seated within the RPV.	In accordance with the Surveillance Frequency Control Program

90. 295024EA2.02 001

UNIT 1 was operating at 100% RTP when a steam leak in the Drywell resulted in the following:

- o Drywell average temperature is .... 255°F
- o Drywell pressure is .... 5.0 psig
- o Torus pressure is .... 4.0 psig

With the above plant conditions, if Drywell Sprays are INITIATED, there is an increased risk of __________ .

Subsequently, the NPO reports that ALL RWL instruments are simultaneously displaying erratic indication, an Emergency Depress will be ordered from __________ ONLY.

Reference Provided

A. damaging the Primary Containment Vent system due to exceeding the capacity of the Torus to Drywell Vacuum Breakers;

31EO-EOP-016-1, CP-2 RPV Flooding

B. damaging the Primary Containment Vent system due to exceeding the capacity of the Torus to Drywell Vacuum Breakers;

31EO-EOP-015-1, CP-1 Emergency RPV Depressurization

C.✓ de-inerting the containment due to opening the Reactor Building to Torus vacuum breakers before the operator can secure sprays;

31EO-EOP-016-1, CP-2 RPV Flooding

D. de-inerting the containment due to opening the Reactor Building to Torus vacuum breakers before the operator can secure sprays;

31EO-EOP-015-1, CP-1 Emergency RPV Depressurization

Description:

Drywell Spray Initiation Limit

At higher Drywell pressures, the rate of pressure reduction can be beyond the capacity of the Torus-to- Drywell vacuum breakers. Differential pressures between the Drywell and suppression chamber may exceed design, causing failure of boundary between the Drywell and the Torus.

At lower Drywell pressures, the Drywell to Torus differential pressure is not limiting. At these pressures, the concerns become:

a) Reducing Drywell pressure below its negative design before the operator can secure sprays.

b) Popping open Reactor Building to Torus vacuum breakers, which could de-inert the containment, before the operator can secure sprays.

Drywell pressure is $\geq$1.85 psig therefore an entry condition into the RC chart exist. Both the RC/P and RC/L legs have overrides, for loss of RWL indication, transitioning to CP-2 RPV Flooding.

The RC/P leg of the RC flowchart has an "Emergency Depress" override just above the "RWL cannot be determined" override. This override will transition to CP-1 Emergency RPV Depressurization

The SRO must understand that a Emergency Depress flag has not been meet and that the "RWL cannot be determined" override is the only override applicable.

The "A" distractor is plausible because it is correct at higher drywell pressures (pressures > 10 psig). The second part is correct

The "B" distractor is plausible if because it is correct at higher drywell pressures (pressures > 10 psig). The second part is plausible if the student only remembers the Emergency Depress override on RC/P leg going to CP-1.

The "D" distractor is plausible because the first part is correct. The second part is plausible if the student only remembers the Emergency Depress override on RC/P leg going to CP-1.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
Unit 1 Graph 8 Drywell Spray Initiation Limit curve

K/A:

295024 High Drywell Pressure

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.10 / 43.5 / 45.13)

EA2.02 Drywell temperature 3.9 4.0

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

LESSON PLAN/OBJECTIVE:
EOP-CURVES-LP-20306,**Ver. 1.0**/EO 201.076.A.15

References used to develop this question:

31EO-EOP-010-1, RC RPV Control, **Ver. 10.0**
31EO-EOP-015-1, CP-1, **Ver. 7.0**
31EO-EOP-016-1, CP-2 RPV Flooding,**Ver. 8.0**

Item 1: SRO ONLY Guideline
Item 2: 31EO-EOP-010-1, RC RPV, **Ver. 10.0**
Item 3: 31EO-EOP-015-1, CP-1, **Ver. 7.0**
Item 4: 31EO-EOP-016-1, CP-2, **Ver. 8.0**
Item 5: U1 Drywell Spray Initiation Curve Graph 8

Modified from 2010 Nile Mile Point NRC Exam Q#76

ORIGINAL QUESTION (NMP 2010 NRC Exam Q#76)

A steam leak in the Drywell has resulted in the following:

o The mode switch is in SHUTDOWN
o Drywell average temperature is 299°F and rising slowly

o Drywell pressure is 11 psig and rising slowly
o Torus pressure is 9 psig and rising slowly
o Torus water level is 12 feet and rising slowly
o RPV pressure is 875 psig and lowering slowly
o All available Drywell Cooling is in service

Which one of the following describes the next action required to be taken?

A. Enter EOP-8, RPV Blowdown, and open three ERVs.
B. Enter EOP-1 Attachment 15 and lower Torus water level.
C✓. Enter EOP-1 Attachment 17 and initiate Containment Spray.
D. Enter EOP-2, RPV Control, and rapidly depressurize the RPV.

Q# 90 K/A 295024EA2.02

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

31EO-EOP-010-1, RC RPV RC/P Path

RC/P

2

WHILE PERFORMING THE FOLLOWING

IF	Emergency Depress is anticipated, EXCEPT for low RWL	THEN	rapidly depress with main turbine bypass valves, irrespective of the resulting cooldown rate.
IF	EMERGENCY DEPRESS IS, OR HAS BEEN, REQUIRED	THEN	perform Emergency Depress
IF	RWL CANNOT be determined	THEN	perform RPV Flooding
IF	drywell pressure is above 1.85 psig	THEN	prevent injection from CS and LPCI pumps per 31EO-EOP-114-1 EXCEPT when required for adequate core cooling

GO TO CP-2
point J

31EO-EOP-010-1, RC RPV RC/L Path

WHILE PERFORMING THE FOLLOWING	
IF RWL CANNOT be determined	THEN perform RPV Flooding
IF primary containment water level and torus pressure CANNOT be maintained below Primary Containment Pressure Limit (Graph 13) AND adequate core cooling can be assured	THEN terminate injection into the RPV from sources external to the primary containment

GO TO CP-2
point J

31EO-EOP-015-1, CP-1

EMERGENCY DEPRESS

G

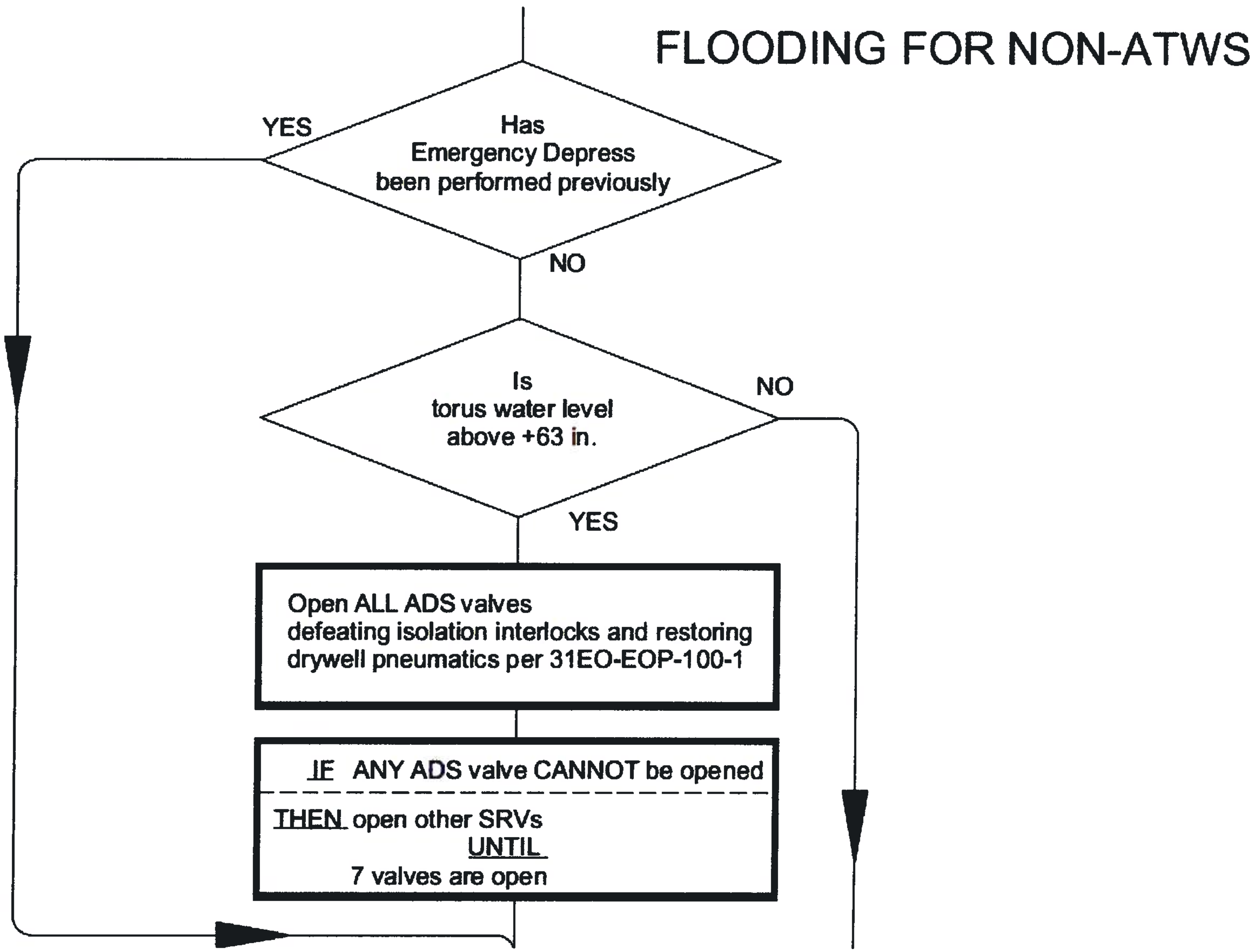
WHILE PERFORMING THE FOLLOWING

IF	THEN
IF PRIMARY CONTAINMENT FLOODING IS OR HAS BEEN REQUIRED	THEN exit the EOPs and enter the Severe Accident Guidelines
IF RWL CANNOT be determined	THEN perform RPV Flooding
IF it is anticipated that primary containment water level will rise above 42.3 ft.	THEN open 1B21-F016 before primary containment water level reaches 42.3 ft. defeating isolation interlocks if necessary per 31EO-EOP-100-1

GO TO SAGs

GO TO CP-2 point J

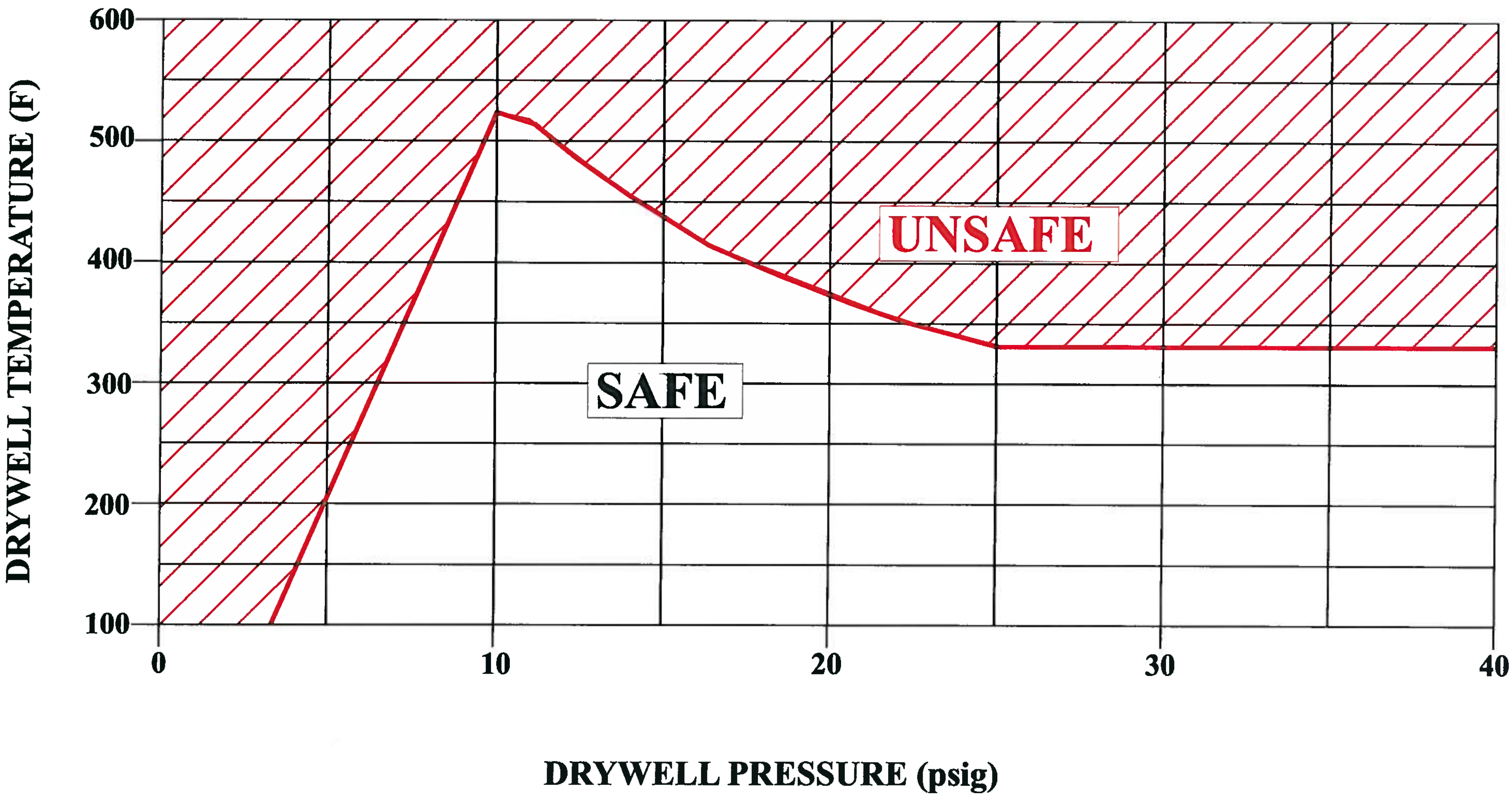
31EO-EOP-016-1, CP-2



GRAPH 8

UNIT 1

DRYWELL SPRAY INITIATION LIMIT



NOTE: May use SPDS Emergency Displays in place of this Graph.

This question was one of the five SRO questions previously submitted for review

SRO question

4 of 5

Question Sat

No changes required

No changes were made

91. 295032EA2.03 001

Unit 1 is operating at 100% RTP, when an unisolable steam leak occurs in the plant.

- o Main Control Room indications and alarms indicate rapidly increasing temperature in the Southwest Diagonal
- o A NPO reports the temperature in the Southwest Diagonal is above Maximum Safe Operating Temperature

This increasing temperature is a result of a steam leak on the _________ system.

IAW 31EO-EOP-014-1, SC/RR, EOP flowchart, the SS is REQUIRED to perform ________ .

A. HPCI;
34GO-OPS-014-1, Fast Reactor Shutdown

B. HPCI;
point A of the RC EOP flowchart

C. RCIC;
34GO-OPS-014-1, Fast Reactor Shutdown

D✓ RCIC;
point A of the RC EOP flowchart

Description:

Edwin, this was question 4 of 5 of the SRO questions that you have already reviewed. NO changes were made since your review

Location
- o Unit 1 RCIC Southwest Diagonal.
- o Unit 2 RCIC Northwest Diagonal
- o Unit 1 HPCI Northeast Diagonal
- o Unit 2 HPCI Southeast Diagonal

IAW the SC/T leg of the 31EO-EOP-014-1, SC/RR, EOP flowchart

PERFORM CONCURRENT

WAIT UNTIL	WAIT UNTIL
primary system is discharging reactor coolant into secondary containment (From Table 7 RCIC is a primary system)	area ambient or differential temperature is above Maximum Safe Operating Temperature in more than one area

BEFORE

ANY area ambient or differential temperature reaches Maximum Safe Operating Temperature	Shut down reactor per 34GO-OPS-013-1 or 34GO-OPS-014-1

PERFORM CONCURRENTLY
RC(A) point A

The SRO must have detailed knowledge of the SC/RR EOP flowchart. First remember that RCIC is considered a primary system per table 7. Then continue down the primary discharge leg of the SC/T leg of the SC/RR EOP flowchart to the "BEFORE" decision box to perform RC/(A) currently.

The "A" distractor is plausible if the Unit 2 Southeast Diagonal is confused with Unit 1 Southwest Diagonal. The second part is plausible if the "BEFORE" decision box was thought to be in the shutdown leg of the SC/T leg of the SC/RR EOP flowchart.

The "B" distractor is plausible if the Unit 2 Southeast Diagonal is confused with Unit 1 Southwest Diagonal . The second part is correct.

The "C" distractor is plausible because the first part is correct. The second part is plausible if the "BEFORE" decision box was thought to be in the shutdown leg of the SC/T leg of the SC/RR EOP flowchart.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

K/A:

295032 High Secondary Containment Area Temperature

EA2. Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : (CFR: 41.10 / 43.5 / 45.13)

EA2.03 Cause of high area temperature 3.8 4.0

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

LESSON PLAN/OBJECTIVE:
EOP-SCRR-LP-20325, SC/RR, **Ver.2.1**, EO 201.079.A.12

References used to develop this question:

31EO EOP 014 2, SC/RR,**Ver. 12.0**
E41-HPCI-LP-00501, **Ver. 5.0**
E51-RCIC-LP-03901,**Ver. 5.0**
Item 1: SRO ONLY Guideline
Item 2: U1 SCRR Temp Path, **Ver. 12.0**
Item 3: IE41-HPCI-LP-00501, HPCI LP, page 22, **Ver. 6.0**
Item 4: E51-RCIC-LP-03901, RCIC LP, page 27, **Ver. 6.1**

Q# 91 K/A 295032 EA2.0.3

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

PERFORM CONCURRENTLY

WAIT UNTIL

primary system
is discharging reactor coolant
into secondary containment
(Table 7)

BEFORE

ANY area ambient or differential temperature
reaches Maximum Safe Operating
Temperature (Table 4)

PERFORM CONCURRENTLY
RC(A) point A

WAIT UNTIL

area ambient or differential temperature
is above
Maximum Safe Operating Temperature
in more than one area
(Table 4)

Shut down reactor per 34GO-OPS-013-1
or 34GO-OPS-014-1

- The Lube Oil Cooler receives cooling water from the HPCI Booster Pump discharge. The water exiting the Lube Oil Cooler discharges to the suction of the HPCI booster pump. **(LT 3b)** **(SO 5f, 6f)** *Figure 01*

- The Aux Oil Pump is provided with a Pull-to-Lock control switch on the H11-P601. This lockout feature is used to prevent HPCI from starting if injection is not required. **(LT 12b)** The Aux Oil Pump is powered from R24-S022.

5. The HPCI Turbine Stop Valve provides quick isolation to the HPCI Turbine in the event of a trip or isolation. It is a hydraulically operated valve (HOV) which is spring closed and hydraulically opened. The TSV shuts on all HPCI trips and isolations, fails closed on a loss of oil pressure, and can be tripped locally at the HPCI Turbine. **(LT 3c)** *Figure 01*

6. The HPCI Turbine Control Valve provides Turbine speed control by throttling Turbine steam flow. It is hydraulically operated with spring pressure to close and hydraulics to open. The hydraulic pressure to the valve is controlled by the HPCI flow controller to maintain the desired HPCI pump discharge flowrate. To prevent overspeeding the turbine on an initiation signal, the Ramp Generator keeps the Control Valve closed until F001 and the Turbine Stop Valve are both off their full closed seat. (The Control Valve will not open if the F001 valve is not open.) **(LT 3c)** **(SO 21a)** *Figure 01*

7. The HPCI Turbine supplies the motive force for the HPCI pump and booster pump. It is a two stage impulse turbine, which is designed to reach rated speed and load within 70 seconds (75 sec per FSAR) of an initiation signal. Steam from the turbine is supplied from Main Steam Line "C" ("B" - Unit 1) and is exhausted to the Suppression Pool. The HPCI Turbine is located in the HPCI Pump Room 87' elevation SE diagonal (NE diagonal - Unit 1). **(LT 3d)** **(SO 5a, 6a)** *Figure 01*

   The Turbine is rated at 750-4100 hp between 2025 and 4060 RPM. The HPCI turbine is designed to deliver rated flow to the reactor between 162 psid and 1169 psid (pump suction to reactor vessel).

NOTE: Minimum recommended speed for Turbine operation is 2000 rpm based on maintaining adequate oil pressure for governor operation and bearing lubrication. Above this speed there is also sufficient steam flow through the Turbine to prevent turbine exhaust valve chatter . **(LT 6)(LCT 24)** **(SO 4)** **(EN 3)**

8. The Exhaust Line Drain Pot removes condensation from the HPCI Turbine Exhaust line drain when the HPCI system is in standby. Level in the Drain Pot is controlled automatically by drain valve F053. F053 is interlocked closed IF BOTH F001 AND TSV ARE NOT FULLY CLOSED. EITHER F001 OR the TURBINE STOP VALVE must be closed for F053 to open (both units). The drain pot discharges to the Barometric Condenser.

8. The RCIC Turbine (2E51-C002) converts thermal energy in steam into mechanical rotation to drive the pump. It is a single stage, horizontal, noncondensing, Terry (water wheel) turbine. This type of turbine is very reliable but not very efficient. The turbine is rated at 485 hp at 1135 psia inlet pressure and 95 hp at 165 psia. It will supply 100% rated flow at speeds between 2000 and 4500 rpm. The turbine is located in the U2 RB, 87' elev., NW diagonal (U1 RB, 87' elev., SW diagonal) **(LT 10f) (SO 6f)**

9. The Exhaust Line Drain Pot provides for condensation removal from the RCIC Turbine Exhaust line drains to the Barometric Condenser through manual valve F027. **(LT 10b) (SO 6b)**

10. RCIC System Rupture Disks (D001 and D002) provide protection for the RCIC Turbine casing from excessive exhaust pressure. The two diaphragms are in series and are designed to rupture at 150 psig. The space between them is vented to the Torus area through an orifice. **(LT 10c) (SO 6c)**

    a. High pressure between the diaphragms will cause a RCIC System Isolation at 10 psig.

    b. Located in U2 at RB Torus area 120' elev. (120RBR17), and in U1 RB Torus area 87' elev. (87RAR07).

11. RCIC Exhaust Line Vacuum Breakers (F102 and F103) prevent vacuum from being formed in the exhaust line by steam condensation following shutdown of the turbine thus siphoning Suppression Pool water into the RCIC Exhaust line. **(LT 10e) (SO 6e)**

    F102 and F103 can be isolated from the exhaust line by two isolation MOVs F104 and F105.

    - F104 is an AC MOV powered from S011
    - F105 is an AC MOV powered from S012

    The F104 and F105 auto close on a combined signal of Hi DW Pressure (1.85#) and Low RCIC Steam Line Pressure (95#). They are located in the exhaust line just prior to the Torus U2 RB 120' elev. (120RBR18) & U1 RB 122' elev. (122RBR07).

12. The Barometric Condenser condenses leakage from the Turbine Labyrinth Steam Seals, and drains from Trip and Throttle Valve, Governor Valve, Steam Supply Line, and Turbine Exhaust Line. **(LT 10d) (SO 6d)**

92. 295037G2.4.49 001

Unit 1 was operating at 100% power when a transient occurred resulting in the following:

- o All control rods did not fully insert
- o Reactor power 8%
- o Reactor Water Level.............. 9 inches
- o Drywell pressure .... 2.2 psig
- o Torus water temperature........ 125°F
- o Both Recirculation pumps are operating at minimum speed

IAW RC-1 and based on the above conditions, the OATC __________ REQUIRED to trip the Recirculation pumps.

Based on the above conditions, and IAW EOP Flowcharts RCA and CP-3 Overrides, _________ REQUIRED to be entered.

NOTE:

31EO-EOP-113-1, Terminating And Preventing Injection Into The RPV
31EO-EOP-114-1, Preventing Injection Into The RPV From Core Spray And LPCI

Reference Provided

A. is NOT;

ONLY EOP-114-1 is

B. is NOT;

BOTH EOP-113-1 and EOP 114-1 are

C. is;

ONLY EOP-114-1 is

D✓ is;

BOTH EOP-113-1 and EOP 114-1 are

Description:
34AB-C71-001, Scram Procedure
RC-1: IMMEDIATE SCRAM REACTIVITY CONTROL ACTIONS

1. INSERT MANUAL SCRAM.

2. PLACE MODE SWITCH to SHUTDOWN.
3. IF BLUE SCRAM LIGHTS are NOT ILLUMINATED, MANUALLY INITIATE ARI.
4. CONFIRM ALL RODS IN by observing FULL IN LIGHTS, SPDS, OR RWM DISPLAY.
5. NOTIFY SS of ROD POSITION CHECK.
6. PLACE SDV ISOL. VLV SW to "ISOL" & CONFIRM CLOSED.
7. IF NOT TRIPPED, PLACE RECIRC PUMPS at MINIMUM SPEED.
8. **IF REACTOR POWER IS ABOVE 5%, TRIP THE RECIRC PUMPS.**
9. INSERT SRMS AND IRMS.
10. IF REACTOR POWER REMAINS ABOVE 5%, INJECT SBLC.
11. SHIFT RECORDERS to read IRMS, when required.
12. RANGE IRMS to bring reading on Scale.
13. NOTIFY SS when above actions are complete.

IAW RC(A) Flowchart RC/P leg Override
if drywell pressure is above 1.85 psig THEN prevent injection from CS and LPCI pumps per 31EO-EOP-114-1 EXCEPT when required for adequate core cooling

IAW CP-3 RWL leg Override
if ALL the following exist:

- o Reactor power is above 5% or CANNOT be determined.................................. 8%
- o RWL is above -155 in... 9 inches
- o Torus water temperature is above Boron Injection Initiation Temperature........ (125°F/ 8%)
- o Drywell pressure is above 1.85 psig.. 2.2 psig

Then Terminate And Prevent Injection IAW 31EO-EOP-113-1

The SRO must diagnose the CP-3 RWL leg Override (including plotting on the BIIT curve) plus apply the RC(A) Flowchart RC/P leg Override for drywell pressure.

The "A" distractor is plausible if power is ≤ 5%. The second part is plausible if the CP-3 RWL leg Override is diagnosed as not being meet

The "B" distractor is plausible if power is ≤ 5%. The second part is correct.

The "C" distractor is plausible because the first is correct. The second part is plausible if the CP-3 RWL leg Override is diagnoised as not being meet

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
Unit 1 Graph 5 BIIT Curve

K/A:

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
(CFR: 41.10 / 43.2 / 45.6) 4.6 4.4

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of diagnostic steps and decision points in the emergency operating procedures (EOP) that involve transitions to event specific subprocedures or emergency contingency procedures.

LESSON PLAN/OBJECTIVE:
EOP-SCRAM-LP-20301, **Ver. 1.0**/LO LR-20301.001

References used to develop this question:
34AB-C71-001, Scram Procedure, **Ver. 12.5**
31EO-EOP-011-1, RCA RPV CONTROL (ATWS), **Ver. 10.0**
31EO-EOP-017-1, CP-3 ATWS LEVEL CONTROL, **Ver. 11.0**
Item 1: SRO ONLY Guideline
Item 2: 34AB-C71-001-1 page 25 **Ver. 12.5**
Item 3: RCA RCA-P Path Override, **Ver. 10.0**
Item 4: CP-3 Override, **Ver. 11.0**

Q#92 K/A 295037 G2.4.49

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

The following placards will be placed on the control boards in view of the operator.
These placards will be performed as directed by the body of this procedure.

RC-1: IMMEDIATE SCRAM REACTIVITY CONTROL ACTIONS

1. INSERT MANUAL SCRAM. ☐
2. PLACE MODE SWITCH to SHUTDOWN. ☐
3. IF **BLUE** SCRAM LIGHTS are NOT ILLUMINATED, MANUALLY INITIATE ARI. ☐
4. CONFIRM **ALL RODS IN** by observing FULL IN LIGHTS, SPDS, OR RWM DISPLAY. ☐
5. NOTIFY SS of ROD POSITION CHECK. ☐
6. PLACE SDV ISOL. VLV SW to "ISOL" & CONFIRM **CLOSED**. ☐
7. IF NOT TRIPPED, PLACE RECIRC PUMPS at MINIMUM SPEED. ☐
8. IF REACTOR POWER IS ABOVE 5%, TRIP THE RECIRC PUMPS. ☐
9. INSERT SRMS AND IRMS. ☐
10. IF REACTOR POWER REMAINS ABOVE 5%, INJECT SBLC. ☐
11. SHIFT RECORDERS to read IRMS, when required. ☐
12. RANGE IRMS to bring reading on Scale. ☐
13. NOTIFY SS when above actions are complete. ☐

Ref: 34AB-C71-001-1

WHILE PERFORMING THE FOLLOWING			
IF	EMERGENCY DEPRESS IS, OR HAS BEEN, REQUIRED	THEN	perform Emergency Depress
IF	RWL CANNOT be determined	THEN	perform RPV Flooding
IF	drywell pressure is above 1.85 psig	THEN	prevent injection from CS and LPCI pumps per 31EO-EOP-114-1 EXCEPT when required for adequate core cooling

WHILE PERFORMING THE FOLLOWING

	1	2	3	
IF EMERGENCY DEPRESS IS, OR HAS BEEN, REQUIRED				THEN perform the following:
IF ALL the following exist:	1	2	3	THEN perform the following:
o Reactor power is above 5% or CANNOT be determined				
o RWL is above -155 in.				
o Torus water temperature is above Boron Injection Initiation Temperature (Graph 5)				
o Drywell pressure is above 1.85 psig OR ANY SRV is open or opens				

This question was one of the five SRO questions previously submitted for review

SRO question

5 of 5

Question Sat

No changes required

No changes were made

93. 700000AA2.07 001

Unit 1 is operating at 100% RTP.

The load dispatcher reports degraded grid conditions with the following indications present for the LAST ONE MINUTE:

- o Generator frequency .... 59.7 hertz
- o 1H11-P653 VOLTMETER 1S40-R600 .... 225 KV

- o 4160 VAC BUS 1E .... 3695 volts
- o 4160 VAC BUS 1F .... 3690 volts
- o 4160 VAC BUS 1G .... 3685 volts

- o 652-122, 4160V BUS 1E VOLTAGE LOW ILLUMINATED
- o 652-222, 4160V BUS 1F VOLTAGE LOW ILLUMINATED
- o 652-322, 4160V BUS 1G VOLTAGE LOW ILLUMINATED

With the above plant conditions, __________ .

IAW 34AB-S11-001-0, Operation With Degraded System Voltage, after 30 minutes, a MINIMUM of __________ REQUIRED to be supplied from the Emergency Diesel Generator(s) on **Unit 1**.

A. Main Turbine blade damage may occur due to off frequency operation;

one (1) 4160 V Emergency bus is

B. Main Turbine blade damage may occur due to off frequency operation;

two (2) 4160 V Emergency busses are

C.✓ Emergency Bus loads may be damaged by degraded voltages;

one (1) 4160 V Emergency bus is

D. Emergency Bus loads may be damaged by degraded voltages;

two (2) 4160 V Emergency busses are

Description:

Edwin, this was question 5 of 5 of the SRO questions that you have already reviewed. NO changes were made since your review

Voltage) page **B3.3-188** states "A reduced voltage condition on a 4.16 kV emergency bus indicates that, while offsite power may not be completely lost to the respective emergency bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function."

Operating the Main Turbine with an under frequency condition could result in turbine blade degradation, therefore the Main Turbine is provided with under frequency trip protection to prevent turbine blade damage.

IAW 34AB-S11-001-0, "Operation With Degraded System Voltage" step 4.4.3 places only the "1E" 4160 V Emergency Bus on its emergency power source. This procedure will align one emergency bus to its emergency power source for both units, therefore having two as a distractor is plausible.

The SRO must have detailed knowledge of the abnormal procedure 34AB-S11-001-0 including the 3825 volt acceptance criteria which is based on TS knowledge and the consequences of opeating with degraded voltage.

The "A" distractor is plausible if the applicant remembers the Main Turbine could be damaged if allowed to operate beyond 59.5 Hz. and 60.5 Hz and confuses this with operating with a degraded voltage. The second part is correct.

The "B" distractor is plausible if the applicant remembers the Main Turbine could be damaged if allowed to operate beyond 59.5 Hz. and 60.5 Hz and confuses this with operating with a degraded voltage. The second part is plausible since this procedure will align one emergency bus to its emergency power source for both units and the applicant confusing this and selecting 2 Emergency Busses supplied by EDGs.

The "D" distractor is plausible since the first part is correct. The second part is plausible since this procedure will align one emergency bus to its emergency power source for both units and the applicant confusing this and selecting 2 Emergency Busses supplied by EDGs.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Correct** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

700000 Generator Voltage and Electric Grid Disturbances

AA2. Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID| DISTURBANCES:
(CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)

AA2.07 Operational status of engineered safety features 3.6 4.0

SRO only because of link to 10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

S11-LP-02706, Basic Grid Operating Concepts, EO 200.116.A.04

References used to develop this question:

U1 TS BASES 3.3.8.1 LOP Instrumentation, page B3.3-188, **Rev. 1.0**
34AB-S11-001-0, Operation With Degraded System Voltage, **Ver. 4.0**
Item 1: SRO ONLY Guideline
Item 2: U1 TS Bases 3.3.8.1 page B3.3-188 **Rev. 1**
Item 3: 34AB-S11-001-0 pages 3 & 4 **Ver. 4.0**

Modified from HLT-5 NRC Exam Q#93

ORIGINAL QUESTION (HLT-5 NRC Exam Q#93)

At 1200 the Northern Control Center (NCC) notified the Control Room Operator that the 230KV Bus voltage cannot be maintained above the normal minimum voltage.

The following parameters currently exist on Unit 2:

- o Main Generator H2 pressure .... 43 psig
- o Main Generator Megawatt ...... 860 MWe
- o Main Generator Megavar ..... +280 MVar
- o "2E" 4160 V Emergency Bus volts .... 3820 volts
- o "2F" 4160 V Emergency Bus volts .... 3820 volts
- o "2G" 4160 V Emergency Bus volts .... 3815 volts

Q#93 K/A 700000 AA2.0.7

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

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV emergency bus indicates that, while offsite power may not be completely lost to the respective emergency bus, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS Function. Therefore, power supply to the bus is transferred from offsite power to onsite DG power when the voltage on the bus drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the large ECCS motors. The Time Delay Allowable Values are long enough for the offsite power supply to usually recover.

This minimizes the potential that short duration disturbances will adversely impact the availability of the offsite power supply. Manual actions are credited in the range of 78.8 to 92% of 4.16 kV to restore bus voltages or to initiate a plant shutdown. The range specified for manual actions indicates that sufficient power is available to the large ECCS motors; however, sufficient voltage for equipment at lower voltages required for LOCA conditions may not be available.

Two channels of 4.16 kV Emergency Bus Undervoltage (Degraded Voltage) Function per associated bus are only required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. (Two channels input to each of the three emergency buses and DGs.) Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

3. 4.16 kV Emergency Bus Undervoltage (Anticipatory Alarm)

A reduced voltage condition on a 4.16 kV emergency bus indicates that, while offsite power is adequate for normal operating conditions, available power may be marginal for some equipment required for LOCA conditions. Therefore, the anticipatory alarms actuate when the 4.16 kV bus voltages approach the minimum required voltage for normal; i.e., non-LOCA conditions. This ensures that manual actions will be initiated to restore the bus voltages or to initiate a plant shutdown.

(continued)

NOTE:	Sustained low voltage conditions may cause the 4160 VAC emergency busses to trip (LOSP degraded voltage relaying). The following section transfers one bus to its associated diesel. The 1E(2E) bus was chosen because of the plant impact of losing loads on 600 VAC bus 1C(2C).

4.4.3 IF the 4160 VAC Bus voltages are NOT RESTORED to acceptable levels WITHIN 30 minutes, **perform** the following to maintain 4160V 1E emergency bus voltage. (Two handed operations will be necessary):

4.4.3.1 **Start** the 1R43-S001A D/G, using the start switch, panel 1H11-P652. ☐

4.4.3.1.1 **Override** 1P41-F310A AND 1P41-F310D, per 34AB-P41-001-1. ☐

4.4.3.1.2 **Open** AND **hold** the following control switches for 1R22-S005, 4160V 1E Bus UNTIL the emergency supply breaker closes:

- ACB 135712, Normal Supply, 4160V Bus 1E ☐
- ACB 135711, Alternate Supply, 4160 V Bus 1E. ☐

4.4.3.1.3 **Load** 1A D/G as necessary AND **perform** applicable abnormal procedures for:

- loss of 4160 V emergency busses ☐
- loss of 600V emergency busses ☐
- loss of essential busses ☐
- loss of instrument busses ☐
- loss of RPS busses ☐

4.4.3.1.4 **Reset** 4160V bus 1E LOSP LOCKOUT (86) Relay. ☐

4.4.3.1.5 **Place** the Overrides for 1P41-F310A AND 1P41-F310D in NORMAL. ☐

4.4.3.2 **Start** the 2R43-S001A D/G, using the start switch, panel 2H11-P652. ☐

4.4.3.2.1 **Override** 2P41-F316A AND 2P41-F316D per 34AB-P41-001-2. ☐

4.4.3.2.2 **Open** AND **hold** the following control switches for 2R22-S005, 4160V 2E Bus UNTIL the emergency supply breaker closes:

- ACB 135554 Unit 2, Normal Supply, 4160V Bus 2E ☐
- ACB 135544 Unit 2, Alternate Supply, 4160V Bus 2E. ☐

4.4.3.2.3 **Load** 2A D/G as necessary AND
perform applicable abnormal procedures for:

- loss of 4160 V emergency busses ☐
- loss of 600V emergency busses ☐
- loss of essential busses ☐
- loss of instrument busses ☐
- loss of RPS busses ☐

4.4.3.2.4 **Reset** 4160V bus 2E LOSP LOCKOUT (86) Relay. ☐

4.4.3.2.5 **Place** the Overrides for 2P41-F316A AND 2P41-F316D in NORMAL. ☐

4.4.4 IF the 4160VAC Bus voltages are NOT restored to acceptable levels WITHIN one hour, an orderly plant SHUTDOWN will be initiated
with the intent of reaching MODE 2 within 7 hours,
MODE 3 within 13 hours AND MODE 4 within 37 hours.
Refer To NMP-EP-110, Emergency Classification Determination and Initial Action. ☐

4.5 IF ANY of the conditions in 4.4 are met,
it may be necessary to ENTER into and **perform** the following:

- **Enter** 34AB-R22-003-1, Station Blackout ☐
- **Enter** 34AB-R22-003-2, Station Blackout ☐
- **Enter** 34AB-R22-002-1, Loss of 4160V Emergency Bus ☐
- **Enter** 34AB-R22-002-2, Loss of 4160V Emergency Bus ☐
- **Enter** 34GO-OPS-013-1, Normal Plant Shutdown ☐
- **Enter** 34GO-OPS-013-2, Normal Plant Shutdown ☐

4.6 During a degraded system voltage condition,
it may become necessary to enter several procedures, as well as those listed in 4.5.
Follow the ARPs AND
take the actions necessary to mitigate any transient. ☐

4.7 **See** attachment 1 for list of essential equipment affected by degraded voltage. ☐

94. G2.1.3 001

At 13:59, **Unit 2** was operating at 100%RTP when a Feedwater transient occurred.

At 14:00, the OATC inserted a manual scram.

At 17:45, the ON-COMING Shift Supervisor (SS) is reviewing the shift logs.

- o The ON-COMING SS previously worked seven (7) days ago

Which ONE of the choices below completes both statements?

IAW NMP-OS-007-001, "Conduct of Operations Standards and Expectations", prior to assuming shift, the ON-COMING SS is REQUIRED to review the previous __________ of shift's log.

IAW REG-025, One, Four, and Eight Hour Reporting Requirements of 10 CFR 50.72, the NRC must be notified of this event NO LATER THAN __________ .

A. three (3) days;
21:59

B.✓ three (3) days;
17:59

C. seven (7) days;
21:59

D. seven (7) days;
17:59

Description:

IAW NMP-OS-007-001, "Conduct of Operations Standards and Expectations", section 6.15, Shift Turnover, states:

6.15.2 Expectations

6.15.2.1 Routine Turnover

Because the proper turnover of information is important for safe and efficient operation, the following apply:

The off-going watch stander remains responsible until properly relieved, and does not relinquish the watch until satisfied that the on-coming watch stander is fully briefed and prepared.

proper turnover that prepares him/her adequately.
Operators assume duties only if they are physically and mentally fit to do so.
The on-coming watch stander reviews applicable unit operating logs, turnover sheets, and temporary orders for **at least the duration of his absence** or **3 days**, whichever is less.

00AC-REG-001-0, Form REG-0025, Item 2.14, states "Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is ***critical*** except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation." Item 2.14 is a 4 hour report, since the reactor was critical when RPS was actuated.

At 14:00, RPS Actuation occurred, therefore the NRC must be notified within 4 hours or 18:00. 17:59 is the "NO LATER THAN" time. 21:59 is the "NO LATER THAN" time for an eight (8) report.

The SRO must remember the Turnover requirements, procedure requirements for Reporting Requirement and determine which notification must be made. Reporting Requirements are above the RO knowledge level.

The "A" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses the 4 hr vs. 8 hr reporting requirement and would be correct if RPS was actuated with the reactor sub critical.

The "C" distractor is plausible if the applicant remembers only reviewing from "at least the duration of his absence" and forgets the "**whichever is less**". The second part is plausible if the applicant confuses the 4 hr vs. 8 hr reporting requirement and would be correct if RPS was actuated with the reactor sub critical.

The "D" distractor is plausible if the applicant remembers only reviewing from "at least the duration of his absence" and forgets the "**whichever is less**". The second part is correct.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

2.1.3 Knowledge of shift or short-term relief turnover practices.
(CFR: 41.10 / 45.13) 3.7 3.9

SRO only because of link to 10CFR55.43(b)(1): Conditions and limitations in the facility license. (Reporting Requirements)

LESSON PLAN/OBJECTIVE:

LT-LP-30004, Administrative Procedures, EO 300.004.B.2

References used to develop this question:

NMP-OS-007-001, Conduct of Operations Standards and Expectations, **Ver. 13.0**
REG-0025, One, Four, and Eight Hour Reporting Requirements of 10 CFR 50.72, **Ver. 8.0**

Item 1: SRO ONLY Guideline
Item 2: NMP-OS-007-001 Section 6.15.2 **Ver. 13.0**
Item 3: REG-0025, Page **Ver. 8.0**

Q#94 K/A G2.1.3

II. Some examples of additional knowledge and abilities as they pertain to an SRO license and the 10 CFR 55.43(b) topics [ES-401, Section D.1.c]:

A. Conditions and limitations in the facility license. [10 CFR 55.43(b)(1)]

Some examples of SRO exam items for this topic include:

- Reporting requirements when the maximum licensed thermal power output is exceeded.
- Administration of fire protection program requirements such as compensatory actions associated with inoperable sprinkler systems, fire doors, etc.
- The required actions for not meeting administrative controls listed in Technical Specification (TS) Section 5 or 6, depending on the facility (e.g., shift staffing requirements).
- National Pollutant Discharge Elimination System (NPDES) requirements, if applicable.
- Processes for TS and FSAR changes.

Note: The analysis and selection of required actions for TS Sections 3 and 4 may be more appropriately listed in the following 10 CFR 55.43 topic.

B. Facility operating limitations in the TS and their bases. [10 CFR 55.43(b)(2)]

Some examples of SRO exam items for this topic include:

- Application of Required Actions (Section 3) and Surveillance Requirements (SR) (Section 4) in accordance with rules of application requirements (Section 1).
- Application of generic Limiting Condition for Operation (LCO) requirements (LCO 3.0.1 thru 3.0.7; SR 4.0.1 thru 4.0.4).
- Knowledge of TS bases that are required to analyze TS required actions and terminology.
- Same items listed above for the Technical Requirements Manual (TRM) and Offsite Dose Calculation Manual (ODCM).

SRO-only knowledge generally cannot be claimed for questions that can be answered *solely* based on knowledge of ≤ 1 hour action statements and the safety limits since Reactor Operators (ROs) are typically required to know these items.

SRO-only knowledge generally cannot be claimed for questions that can be answered *solely* based on expected RO TS knowledge. RO's are typically expected to know the LCO statements and associated applicability information, i.e., the information above the double line separating the

6.15 Shift Turnover

6.15.1 Standard

On-coming and off-going shift operators participate in a comprehensive exchange of information to ensure an awareness of planned activities and operational challenges.

6.15.2 Expectations

6.15.2.1 Routine Turnover

Because the proper turnover of information is important for safe and efficient operation, the following apply:

- The off-going watch stander remains responsible until properly relieved, and does not relinquish the watch until satisfied that the on-coming watch stander is fully briefed and prepared.
- The on-coming watch stander assumes responsibilities only after conducting a proper turnover that prepares him/her adequately.
- Operators assume duties only if they are physically and mentally fit to do so.
- The on-coming watch stander reviews applicable unit operating logs, turnover sheets, and temporary orders for at least the duration of his absence or 3 days, whichever is less.
- The on-coming control room watch stander walks down the control boards and displays thoroughly.
- A complete turnover includes work in progress, status of equipment and alarms, activities recently completed and planned, and a review of logs.
- Whenever SM, SS, OATC, or UO shift relief occurs, the on-coming individual will give a Crew Update stating they have the position. Due to the physical layout of the FNP control room, it is acceptable for the SS, OATC, and UO to communicate via a 3-way communication when shift relief has occurred.

6.15.2.2 Special Circumstances

During activities that demand special attention, such as Infrequently Performed Tests and Evolutions (IPTE), reactor startups or transients, turnover is delayed so as to minimize distractions and enhance continuity. In such cases, the Shift Manager or Shift Supervisor is responsible for determining the timing of individual watch station relief.

6.16 Watch Standing Practices

6.16.1 Standard

Shift operators monitor the condition of plant equipment continually and thoroughly. They expect reliable equipment and are intolerant of equipment problems.

6.16.2 Expectations

Item No.	Reporting Requirement(s)	Report Description	Type of Report	Date/Time by which report must be submitted	Report Normally Initiated by. . .	Report Normally Approved by. . .	Report Normally Sent to. . .
2.11 2.12 2.13 2.14 2.15 2.16 2.17	10 CFR 50.54(z) 10 CFR 50.72(b)(2)(i) 10 CFR 50.72(b)(2)(iv)(A) 10 CFR 50.72(b)(2)(iv)(B) 10 CFR 50.72(b)(2)(xi) 10 CFR 72.75(b)(1) 10 CFR 72.75(b)(2)	FOUR-HOUR REPORTS	ENS	Within four hours	SOS	SOS	NRC

2.11 10 CFR 50.54(z)

(z) Each licensee with a utilization facility licensed pursuant to sections 103 or 104b. of the Act shall immediately notify the NRC Operations Center of the occurrence of any event specified in § 50.72 of this part.

2.12 10 CFR 50.72(b)(2)(i)

(b) Non-emergency events-- (See item 1.14 for requirements for follow-up notifications.)
(2) Four-hour reports. If not reported under paragraphs (a) or (b)(1) of this section, the licensee shall notify the NRC as soon as practical and in all cases, within four hours of the occurrence of any of the following:
(i) The initiation of any nuclear plant shutdown required by the plant's Technical Specifications.

3 10 CFR 50.72(b)(2)(iv)(A)

(iv)(A) Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

2.14 10 CFR 50.72(b)(2)(iv)(B)

(B) Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

2.15 10 CFR 50.72(b)(2)(xi)

(xi) Any event or situation, related to the health and safety of the public or onsite personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an onsite fatality or inadvertent release of radioactively contaminated materials. Note: the Chemistry Manager or Corporate Environmental Affairs should be consulted for reporting applicability prior to making a four (4) hour report involving groundwater contamination incidents when time allows.

2.16 10 CFR 72.75(b)(1)

(b) *Non-emergency notifications:* Four-hour reports. Each licensee shall notify the NRC as soon as possible but not later than four hours after the discovery of any of the following events or conditions involving spent fuel, HLW, or reactor-related GTCC waste:
(1) An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under this part when the action is immediately needed to protect the public health and safety, and no action consistent with license or certificate of compliance conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.

Item No.	Reporting Requirement(s)	Report Description	Type of Report	Date/Time by which report must be submitted	Report Normally Initiated by. . .	Report Normally Approve d by. . .	Report Normally Sent to. . .
3.11 3.12 3.13 3.14 3.15 3.16 3.17 3.18 3.19	10 CFR 50.54(z) 10 CFR 50.72(b)(3)(ii) 10 CFR 50.72(b)(3)(iv) 10 CFR 50.72(b)(3)(v) 10 CFR 50.72(b)(3)(xii) 10 CFR 50.72(b)(3)(xiii) 10 CFR 72.75(c)(1) 10 CFR 72.75(c)(2) 10 CFR 72.75(c)(3)	EIGHT-HOUR REPORTS	ENS	Within eight hours	SOS	SOS	NRC

3.11 10 CFR 50.54(z)

(z) Each licensee with a utilization facility licensed pursuant to sections 103 or 104b. of the Act shall immediately notify the NRC Operations Center of the occurrence of any event specified in § 50.72 of this part.

3.12 10 CFR 50.72(b)(3)(ii)(A)
10 CFR 50.72(b)(3)(ii)(B)

(b) Non-emergency events—(See item 1.14 for requirements for follow-up notifications.)

(3) Eight-hour reports. If not reported under paragraphs (a), (b)(1) or (b)(2) of this section, the licensee shall notify the NRC as soon as practical and in all cases within eight hours of the occurrence of any of the following:

(ii) Any event or condition that results in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; which can include conditions resulting from exceeding Tech Spec safety limits, or

(B) The nuclear power plant being in an unanalyzed condition that significantly degrades plant safety including Exceeding Tech Specs Safety limits.

3.13 10 CFR 50.72(b)(3)(iv)

(iv)(A) Any event or condition that results in valid* actuation of any of the systems listed in paragraph (b)(3)(iv)(B) of this section, except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.

(B) The systems to which the requirements of paragraph (b)(3)(iv)(A) of this section apply are:

(1) Reactor protection system (RPS) including: Reactor scram and reactor trip.

(2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs).

(3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: High-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems.

(4) ECCS for boiling water reactors (BWRs) including: High-pressure and low-pressure core spray systems; high-pressure coolant injection system; low pressure injection function of the residual heat removal system.

(5) BWR reactor core isolation cooling system; isolation condenser system; and feedwater coolant injection system.

(6) PWR auxiliary or emergency feedwater system.

(7) Containment heat removal and depressurization systems, including containment spray and fan cooler systems.

(8) Emergency ac electrical power systems, including: Emergency diesel generators (EDGs); hydroelectric facilities used in lieu of EDGs at the Oconee Station; and BWR dedicated Division 3 EDGs.

95. G2.1.36 001

Unit 2 is in REFUEL with core reload in progress.

IAW 34FH-OPS-001-0, Fuel Movement Operation, which ONE of the choices below correctly completes the following statements?

For core reload, the Unit 2 Reactor Mode switch is REQUIRED to be in the __________ .

The fuel movement prerequisites must be completed __________ .

A.✓ Refuel position and LOCKED;
at EACH shift change (12 hour shift) during fuel movement

B. Refuel position and LOCKED;
ONLY once during the refueling outage (prior to the initial fuel movement)

C. Refuel position ONLY (NOT locked);
at EACH shift change (12 hour shift) during fuel movement

D. Refuel position ONLY (NOT locked);
ONLY once during the refueling outage (prior to the initial fuel movement)

Description:

IAW 34FH-OPS-001-0, Limitation 5.2.2 states "Fuel movements in the reactor vessel may be performed only WHEN the Reactor Mode switch is **LOCKED** in the **REFUEL** position.

Prerequisite 6.3, states, Prerequisites shall be performed PRIOR to moving any fuel in or above the RPV or movement of irradiated fuel in the Secondary Containment AND **at each shift change (12 hour shift)**.

The SRO must know detailed knowledge of 34FH-OPS-001-0 prerequisite requirements and 34SV-F15-001-2 to obtain the correct answer to this question.

The "B" distractor is plausible since the first part is correct and the second part is plausible if the applicant does not remember the requirement or confuses this with 34SV-F15-001-2 requirement for performing the Hoist Limit Checks, which requires only once (prior to) during the refueling outage.

The "C" distractor is plausible if the applicant does not remember the limitation setforth in 34FH-OPS-001-0 or thinks that since the switch is in the position for the circuit to provide the necessary interlocks/rod blocks, it is performing its intended function and not required to be locked. The second part is correct.

The "D" distractor is plausible if the applicant does not remember the limitation setforth in 34FH-OPS-001-0 or thinks that since the switch is in the position for the circuit to provide the necessary interlocks/rod blocks, it is performing its intended function and not required to be locked. The second part is plausible if the applicant does not remember the requirement or confuses this with 34SV-F15-001-2 requirement for performing the Hoist Limit Checks, which requires only once (prior to) during the refueling outage.

A. **Correct** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

2.1.36 Knowledge of procedures and limitations involved in core alterations.

(CFR: 41.10 / 43.6 / 45.7) 3.0 4.1

SRO only because of link to 10CFR55.43(b)(7): Fuel handling facilities and procedures.

LESSON PLAN/OBJECTIVE:

F15-RF-LP-04502, Refueling, EO 045.018.A.03 & EO 300.044.A.01

References used to develop this question:

34FH-OPS-001-0, Fuel Movement Operation, **Ver. 24.7**
34SV-F15-001-2, Refueling Interlocks And Hoist Limit Checks, **Ver. 18.3**
Item 1: SRO ONLY Guideline
Item 2: 34FH-OPS-001-0 pages 7, 12, 31, 34 **Ver. 24.7**
Item 3: 34SV-F15-001-2, page 2, **Ver. 18.3**

Modified from HLT-6 NRC Exam Q#94

ORIGINAL QUESTION (HLT-6 NRC Exam Q#94)

Unit 1 is in REFUEL with core reload in progress.

The Control Room informs the Refueling SRO that the individual on the headset with them has to be relieved.

IAW 34FH-OPS-001-0, Fuel Movement Operation, which ONE of the choices below completes the following statements?

The individual who relieves the person in the Main Control Room __________ REQUIRED to have a NRC License.

The fuel movement prerequisites must be completed __________ .

A. is;
ONLY once during the refueling outage (prior to the initial fuel movement)

B. is NOT;
ONLY once during the refueling outage (prior to the initial fuel movement)

C.✓ is;
at EACH shift change (12 hour shift) during fuel movement

D. is NOT;
at EACH shift change (12 hour shift) during fuel movement

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.

G. Fuel handling facilities and procedures. [10 CFR 55.43(b)(7)]

Some examples of SRO exam items for this topic include:

- Refuel floor SRO responsibilities.
- Assessment of fuel handling equipment surveillance requirement acceptance criteria.
- Prerequisites for vessel disassembly and reassembly.
- Decay heat assessment.
- Assessment of surveillance requirements for the refueling mode.
- Reporting requirements.
- Emergency classifications.

This does not include items that the RO may be responsible for at some sites such as fuel handling equipment and refueling related control room instrumentation operability requirements, abnormal operating procedure immediate actions, etc. For example, an RO is required to stop the refueling process when communication is lost between the control room and the refueling floor, therefore, this is a task that is both an RO and SRO responsibility and is not SRO-only.

5.1.9 When moving fuel through the transfer canal slot and the scorpion is in use, in order to maintain ALARA, personnel working in the trough will be restricted from working within 4 feet of either side of the transfer canal slot OR as directed by HP personnel.

5.1.10 If the associated Reactor Cavity is drained, then prior to placing an irradiated fuel bundle in the Unit 1 fuel storage rack # 22 OR Unit 2 fuel storage rack # 9, in order to maintain ALARA, personnel will be restricted from working in the Reactor Cavity adjacent to the 'Fuel Pool to Reactor Cavity' gate.

5.1.11 If the Fuel Pool Transfer Canal is drained, then prior to placing an irradiated fuel bundle in Unit 1 fuel storage rack # 17 OR Unit 2 fuel storage rack # 1, in order to maintain ALARA, personnel will be restricted from working in the Fuel Pool Transfer Canal adjacent to the associated 'Fuel Pool to Fuel Pool Transfer Canal' gate.

5.2 LIMITATIONS

5.2.1 Fuel may NOT be moved in the reactor vessel UNLESS all rods are fully inserted and any jumpers inhibiting the refueling interlocks are removed in all cells containing fuel. NOTE: Step 4.3.11 is performed per the Special Operations section of TS 3.10.6.

5.2.2 Fuel movements in the reactor vessel may be performed only WHEN the Reactor Mode switch is LOCKED in the REFUEL position.

5.2.3 Reactor Vessel water level shall be maintained >23' and Fuel Pool Water level shall be maintained >21 feet, above the top of the fuel assemblies seated in the Vessel and Fuel Pool. Fuel Pool level readings can be obtained from 1T24-R001 and 2T24-R001, Fuel Pool level indicators, located in the Fuel Pools.

5.2.4 Visual contact with the fuel bundle/blade guide being moved must be maintained at all times, except for momentary obstacle and destination checks.
IF visual contact is lost, the movement must be stopped immediately.
A camera MUST be used to assist in bundle/blade guide location, orientation, and verification (including checking for bail handle damage).

5.2.5 All operations of the Refueling Platform must be made in a controlled, deliberate manner to ensure safe operations.

5.2.6 No Fuel may be moved without a channel UNLESS:
approved by Plant Management
AND
the fuel bundle has been discharged from the core for at least 45 days.

5.2.7 Irradiated fuel must NOT be ungrappled in any Fuel Preparation Machine (FPM) UNLESS the FPM is in the full down position.

6.0 PREREQUISITES

6.1 Prerequisites for Unit 1 are contained on Attachment 1 and Attachment 8.

6.2 Prerequisites for Unit 2 are contained on Attachment 2 and Attachment 9.

6.3 Prerequisites shall be performed PRIOR to moving any fuel in or above the RPV or movement of irradiated fuel in the Secondary Containment AND at each shift change (12 hour shift).

6.4 Reactor Engineering has provided an approved Fuel Movement Sheet that meets all the requirements of 42FH-ERP-014-0, Fuel Movement.

6.5 IF the Reactor Vessel has been disassembled in preparation for Refueling,
ensure the Cattle Chute has been installed in accordance with 52GM-MME-015-1 / 52GM-MME-015-2, Reactor Vessel Disassembly, subsection 7.14.
IF Cattle Chute is NOT installed,
Fuel Movement between the Rx Vessel and Fuel Pool is allowable IF:

1) HP denies all access to Drywell

   AND

2) All access points into Drywell are secured.
   (i.e. All Hatches in place, all Shield Blocks in place and Drywell Airlock Doors locked.)

6.6 IF performing this procedure for the purpose of core alterations,
THEN an IPTE Briefing will be performed:

__________________________________ / __________

Brief has been performed by applicable Management representative / Date

1.0 UNIT 2 FUEL MOVEMENT PREREQUISITES

CONTINUOUS

1.1 PRIOR to moving fuel with the Unit 2 Refuel Bridge in Secondary Containment AND at each shift change (12 hour shift) ensure that the following prerequisites have been met:

1.1.1 Verify ALL steps of Attachment 9 have been signed off as completed or N/A'ed:

Day Shift:_______/_______ Date Time Night Shift:_______/_______ Date Time

Init._______ Init._______

1.1.2 Clearances which would affect performance of this procedure have been released, the system is operational for this procedure as determined by the Shift Supervisor and the Shift Supervisor has approved release of the system to perform this procedure.

Day Shift:_______/_______ Date Time Night Shift:_______/_______ Date Time

Init._______ Init._______

1.1.3 Unit 1 and Unit 2 Standby Gas Treatment Systems are operable in accordance with Technical Specifications. (TS 3.6.4.3)

Day Shift:_______/_______ Date Time Night Shift:_______/_______ Date Time

Init._______ Init._______

NOTE:	"WITHIN 7 days PRIOR to" means "prior to the start of the campaign", with a campaign being considered a Refueling Outage or a Dry Cask Storage Campaign, as long as no issues or component problems have arisen to invalidate the hoist / load cell checks.

1.1.4 The appropriate (any hoists to be used) Hoist Limit Checks have been performed in accordance with approved Operating procedures (34SV-F15-001-2). (WITHIN 7 days PRIOR to start of fuel movement of fuel assemblies or control rods within the RPV (T3.9.3).

NOTE:	If Hoist Checks are current, 'Performed By (*)Init.' Step may be marked N/A.

Performed: _______/_______ Date Time (*)Init. _______

Day Shift:_______/_______ Date Time Night Shift:_______/_______ Date Time

Init._______ Init._______

NOTE:	Complete sign offs above upon obtaining HP Control Point phone number and N/A thereafter until Items 1) or 2) above occur.

1.2.4 If in place, the Cattle Chute Swing-gate has been lowered into the DOWN position

Day Shift:_______/_______ Date Time   Night Shift:_______/_______ Date Time

Init._______   Init._______

NOTE	Shorting links are required to be removed in Mode 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies and SDM NOT demonstrated per 42CC-ERP-010-0, Shutdown Margin Demonstration, for current core configuration. Provisions are made in Attachment 4 to allow installation OR removal of the links as required.

1.2.5 IF required, shorting links, as shown on Attachment 4, have been removed (WITHIN 30 min. PRIOR to entering applicability).

NOTE:	If this condition does not apply, this step may be marked N/A.

Day Shift:_______/_______ Date Time   Night Shift:_______/_______ Date Time

Init._______   Init._______

1.2.6 Reactor Mode Switch is locked in the REFUEL position and key removed. (TS 3.9.2)

Day Shift:_______/_______ Date Time   Night Shift:_______/_______ Date Time

Init._______   Init._______

2.0 APPLICABILITY

This procedure applies to 2F15-E003, Unit 2 Refueling Platform, AND associated hoists used for core alterations in the Unit 2 RPV AND:

- The HOIST LIMIT CHECKS subsection is to be performed WITHIN 7 days PRIOR to start for the applicable hoist(s) that move:
  - control rods OR fuel assemblies within the RPV.
  - fuel assemblies within the fuel pool.
- The REFUELING INTERLOCKS FUNCTIONAL TEST subsection is to be performed for the applicable components:
  - PRIOR to any core alterations AND every 7 days thereafter UNTIL completion of core alterations.
  - PRIOR to resuming core alterations AFTER completion of any repair, maintenance OR replacement of any component that could affect the refueling interlocks.
- Re-test of weight(s) will be performed IAW subsection 7.1.3 following any damage OR modification to weight(s).

3.0 REFERENCES

3.1 90AC-OAM-001-0, Test And Surveillance Control

3.2 40AC-ENG-016-0, Reactivity Management Program

3.3 Unit 2, Technical Specifications, TS SR 3.9.1.1 and 3.9.2.2

3.4 Unit 2, TRM, TSR 3.9.3.1, TSR 3.9.3.2, TLCO 3.9.4, and TSR 3.9.4.1

3.5 FSAR, Unit 2, Sections 7.6.1, Refueling Interlocks

3.6 SX-28057, Instruction Manual Refueling Platform

3.7 SX-25741 thru SX-25747, Elementary Diagram Refueling Platform

3.8 H-27499 thru H-27514, RMCS, C11 Elementary Diagram, Shts 1 thru 16 of 19

3.9 57CP-CAL-009-0, Refueling Platform Load Cell and Indicator/Controller Calibration

3.10 42FH-ERP-012-0, New Fuel & New Channel Handling

3.11 SX-25743, Refueling Platform Basic Logic Diagram

96. G2.2.18 001

IAW 31GO-OPS-024-0, Outage Safety Assessment, which ONE of the following completes both statements?

The individual responsible for completing the Outage Safety Assessment is the ________ .

Planned entry into an ORANGE (moderate risk) condition REQUIRES approval of the __________ .

A. respective Unit Operator at the Controls (OATC); Work Management Director ONLY

B.✓ respective Unit Operator at the Controls (OATC); Work Management Director AND the Plant Manager

C. Shift Technical Advisor (STA); Work Management Director ONLY

D. Shift Technical Advisor (STA); Work Management Director AND the Plant Manager

Description:

IAW 31GO-OPS-024-0, Outage Safety Assessment, Section 4.0 Responsibility, step 4.4 states "The respective **Unit Operator at the Controls** (OATC) is responsible for completing the Outage Safety Assessment. The STA is part of the Outage Safety Assessment process , just not responsible for completing. They are responsible for distribution of the OSA Checklist as directed by the Shift Manager.

Step 5.3 states:

The following list provides the definitions of the color codes used in the OSA and what actions, IF necessary, will be taken for each case:

Green (minimal risk) - This condition represents full safety function redundancy and does not require special actions.

Yellow (low risk) - This condition represents reduced but adequate safety function redundancy. This condition requires Shift Outage Manager approval. The operating shift shall be notified of the reduction in redundancy, no other contingency actions or plans are required.

Orange (moderate risk) - This condition represents a reduction in the capability to perform the safety function with little or no redundancy remaining. Steps shall be taken to minimize time spent in this condition. Planned entry into this condition requires detailed compensatory

Plant Manager. If unplanned entry into orange condition occurs immediate actions will be taken to restore Yellow status.

Red (high risk) - This condition represents a potential loss of one or more Key Safety Functions. Planned entry into this condition is NOT allowed. If unplanned entry into Red condition occurs immediate actions will be taken to restore Yellow status.

The SRO must have detailed knowledge of this procedure including authorizations for planned entry into risk situations. This detailed procedure knowledge is above the RO knowledge level.

The "A" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses that planned low risk activities only requires one person for approvals with planned medium risk, which requires two (2) different approvals.

The "C" distractor is plausible if the applicant confuses the STAs responsibility for distribution of the OSA Checklist with actual performance of the assessment. The second part is plausible if the applicant confuses that planned low risk activities only requires one person for approvals with planned medium risk, which requires two (2) different approvals.

The "D" distractor is plausible if the applicant confuses the STAs responsibility for distribution of the OSA Checklist with actual performance of the assessment. The second part is correct.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.
(CFR: 41.10 / 43.5 / 45.13) 2.6 3.8

SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

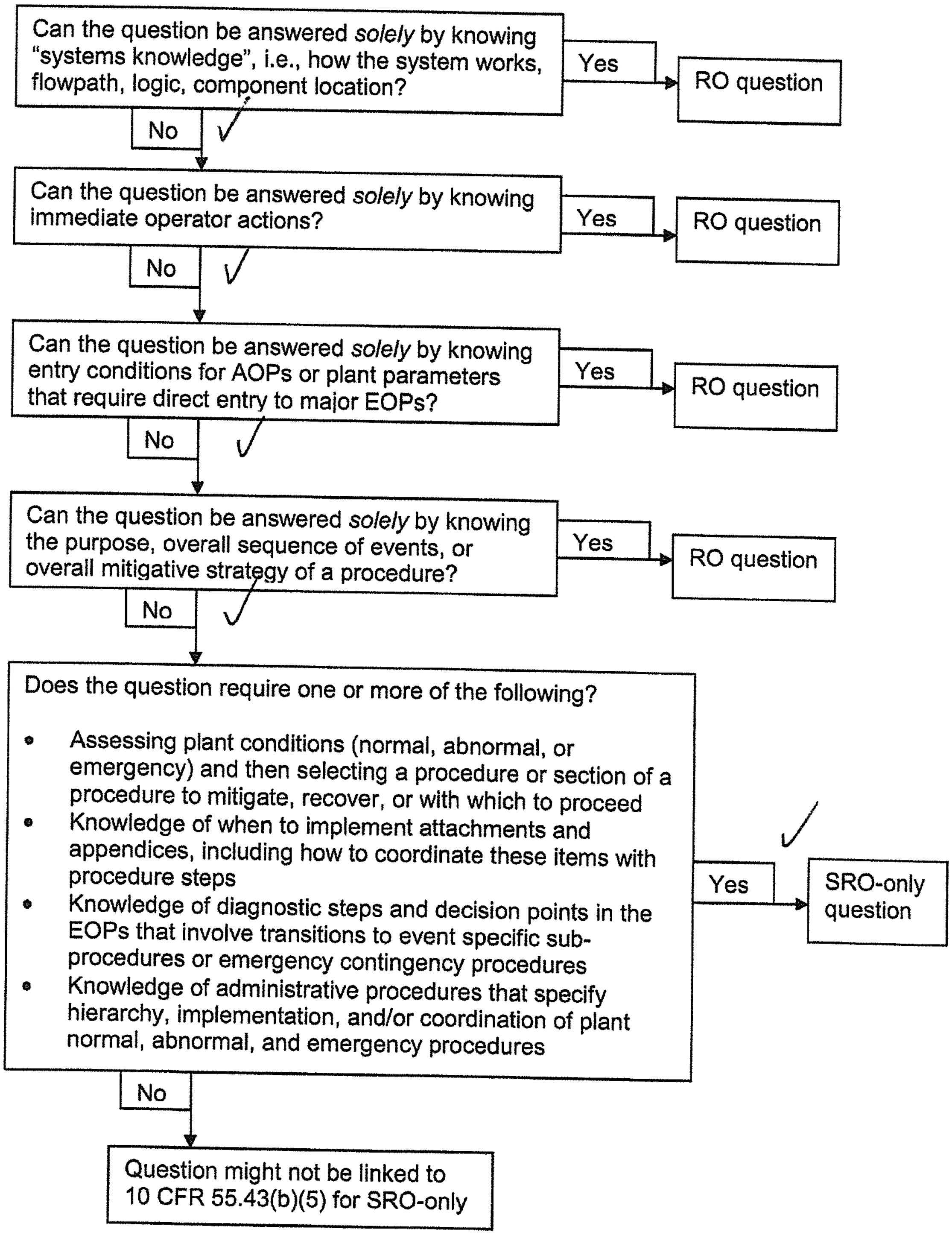
LT-LP-30007, Shift Operations And Evolutions, TO 500.003.A

References used to develop this question:

31GO-OPS-024-0, Outage Safety Assessment, **Ver. 3.3**
Item 1: SRO ONLY Guideline
Item 2: 31GO-OPS-024-0, pages 2 & 4, **Ver. 3.3**

Q#96 K/A G2.2.18

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



3.0 REFERENCES

- NUMARC 91-06 Guidelines for Industry Actions to Assess Shutdown Management (December 1991)
- NUMARC 93-01, Section 11, Assessment of Risk Resulting from Performance of Maintenance Activities (February 2000)
- Regulatory Guide 1.182, Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants.
- INPO 06-005, Guidelines for the Conduct of Outages at Nuclear Power Plants.
- INPO SER 02-08, Reduced Shutdown Safety Margins.
- INPO SOER 09-1, Shutdown Safety.
- NMP-GM-021-002, Plant Hatch Switchyard Access and Maintenance Controls.
- 90AC-OAM-003-0, Outage Risk Management.
- NMP-OM-002, Shutdown Risk Management
- NMP-OS-010, Protected Train/Division And Protected Equipment Program

4.0 RESPONSIBILITY

4.1 Operations Management is responsible for reviewing the daily Outage Safety Assessment (OSA) checklist, providing direction and support to Shift Personnel to ensure plant configurations with reduced redundancy are kept to a minimum, and ensuring the initiation of a CR for unplanned entry into conditions other than GREEN.

4.2 The Shift Manager (SM) is responsible for ensuring the accuracy and distribution of the OSA Checklist and informing Operations Management and Outage Management of changes in OSA status.

4.3 The respective Unit Shift Supervisor (SS) is responsible for ensuring that the availability is recorded for equipment necessary to ensure the plant is maintained in a safe condition and to ensure all applicable Technical Specifications are met.

4.4 The respective Unit Operator at the Controls (OATC) is responsible for completing the Outage Safety Assessment.

4.5 The Shift Technical Advisor (STA) or Backup Shift Supervisor (BUSS) is responsible for distribution of the OSA Checklist as directed by the SM.

4.6 The responsibility for reviewing and approving schedules & contingency plans for planned entry into YELLOW or ORANGE conditions will be as required by 90AC-OAM-002-0.

5.1.2.4 For credit to be taken for the 1G71 system for either unit the physical piping must be installed to the Fuel Pool for the unit taking credit for the system. The system must be filled and either running or available to start. IF the Alternate Decay Heat Removal System diesel generator is not available, reduce the availability number by one.

5.2 Protected Equipment or Protected Area:
Equipment or an area containing equipment required to ensure availability/reliability of systems needed for the Critical Safety Functions of Decay Heat Removal, Reactivity Control, Coolant Inventory, Secondary Containment or Power Availability (i.e. the Control Room panels when Shutdown Cooling is in service before the cavity is flooded, a Diesel and associated switchgear when it is the only operable Diesel Generator, or any area identified in the Shutdown Risk Assessment).

5.3 The following list provides the definitions of the color codes used in the OSA and what actions, IF necessary, will be taken for each case:

- Green (minimal risk) - This condition represents full safety function redundancy and does not require special actions.
- Yellow (low risk) - This condition represents reduced but adequate safety function redundancy. This condition requires Operations Outage Manager approval. The operating shift shall be notified of the reduction in redundancy, no other contingency actions or plans are required.
- Orange (moderate risk) - This condition represents a reduction in the capability to perform the safety function with little or no redundancy remaining. Steps shall be taken to minimize time spent in this condition. Planned entry into this condition requires detailed compensatory actions and contingency plans, and approval by the Work Management Director and the Plant Manager. If unplanned entry into orange condition occurs immediate actions will be taken to restore Yellow status.
- Red (high risk) - This condition represents a potential loss of one or more Key Safety Functions.
  Planned entry into this condition is **NOT** allowed.
  If unplanned entry into Red condition occurs immediate actions will be taken to restore Yellow status.

5.4 Time to saturation calculations will be done per 34AB-E11-001-1 OR 34AB-E11-001-2, Loss of Shutdown Cooling.

97. G2.2.39 001

Unit 1 is operating at 100% RTP.

At 0800, the 1A Diesel Generator is declared inoperable.

IAW TS 3.8.1 AC Sources - Operating, the LATEST ALLOWABLE time to complete the initial TS REQUIRED portions of 34SV-SUV-013-0, Weekly Breaker Alignment Checks, due to the inoperable Diesel Generator is __________ .

IAW TS SR 3.0.2, the subsequent performance time of 34SV-SUV-013-0 is MET if it is completed within _________ times the interval specified in the RAS Completion Time.

A. 0829;
2.0

B. 0829;
1.25

C. 0859;
2.0

D✓ 0859;
1.25

Description:

IAW TS 3.8.1 AC Sources - Operating, Required Action C.1, Perform SR 3.8.1.1 (34SV-SUV-013-0, Weekly Breaker Alignment Checks) for OPERABLE required offsite circuit(s) within one (1) hour and once per 8 hours thereafter.

IAW TS SR 3.0.2, The specified Frequency for each SR is met if the Surveillance is performed within **1.25** times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance **after** the initial performance. Exceptions to this Specification are stated in the individual Specifications.

Two (2.0) times is associated with SR 3.03 which states "If it is discovered that a Surveillance was not performed within its **specified Frequency**, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the **limit of the specified Frequency, whichever is greater.**" In this case within the specified in a sense is like 2.0 times the specified frequency ex. if specified frequency is 4 hours and you add four hours to it then you have doubled (2.0 times) the specified frequency. This example will apply in this case to be 2.0 times the TS frequency.

The SRO must apply Motherhood Statement, SR 3.0.2 & SR 3.0.3, in order to select the correct answer to the question. ROs are not responsible for the Motherhood Statements from memory and are above the RO knowledge level.

The "A" distractor is plausible since TS contain Required Actions with a Completion time of 30 minutes and would be a correct answer if asking different TS such as the time to restore parameter(s) to within limits IAW TS LCO 3.4.9 RCS Pressure and Temperature (P/T) Limits. The second part is plausible if the applicant confuses the TS SR 3.0.3 with TS SR 3.0.2 in applying extension time rules.

The "B" distractor is plausible since TS contain Required Actions with a Completion time of 30 minutes and would be a correct answer if asking different TS such as the time to restore parameter(s) to within limits IAW TS LCO 3.4.9 RCS Pressure and Temperature (P/T) Limits. The second part is correct.

The "C" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses the TS SR 3.0.3 with TS SR 3.0.2 in applying extension time rules.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

Equipment Control

2.2.39 **Knowledge of less than or equal to one hour Technical Specification action statements for systems.** (CFR: 41.7 / 41.10 / 43.2 / 45.13) 3.9 4.5

SRO only because of link to 10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.

<u>LESSON PLAN/OBJECTIVE:</u>

Reference(s) used to develop this question:

U1 TS 3.8.1 AC Sources - Operating, **Amend. 246**
U1 TS 3.4.9 RCS Pressure and Temperature (P/T) Limits, **Amend. 246**
U1 TS SR 3.0.2 & SR 3.0.3
Item 1: SRO ONLY Guideline
Item 2: U1 TS 3.8.1 pages 3.8-1 thru 3.8-4, **Amend 246**
Item 3: U1 TS 3.4.9 page 3.4-18 **Amend. 266**
Item 4: U1 TS SR3.02 & SR3.0.3 page 3.0-1 **Amend 250**

Modified from G2.2.39 001

ORIGINAL QUESTION (G2.2.39 Q#001)

Unit 1 is operating at 100% RTP.

At 0800, the 1A Diesel Generator is declared inoperable.

IAW TS 3.8.1 AC Sources - Operating, which ONE of the choices below is the LATEST ALLOWABLE time to complete the TS REQUIRED portions of 34SV-SUV-013-0, Weekly Breaker Alignment Checks, due to the inoperable Diesel Generator?

A. 0814
B. 0819
C. 0829
D.✓ 0859

98. G2.3.4 001

A Nuclear Plant Operator (NPO) is being sent out to isolate a radioactive leak on the 158' elevation of the **Unit 1** Reactor Building. Dose rates in the area are significantly higher than normal.

IAW 60AC-HPX-001-0, Radiation Exposure Limits, the MAXIMUM Administrative Annual TEDE exposure that the NPO can receive, without requiring written approval from an HP Supervisor, is __________ .

The _________ is the MINIMUM level of qualification necessary to declare a Radiological Event AND will make immediate decisions concerning Emergency Call List notifications.

A. 4,000 mrem;
Shift Manager

B. 4,000 mrem;
Control Room Shift Supervisor

C. 2,000 mrem;
Shift Manager

D✓ 2,000 mrem;
Control Room Shift Supervisor

Description:

IAW 60AC-HPX-001-0, Radiation Exposure Limits, step 8.2.1 lists the following administrative limits:

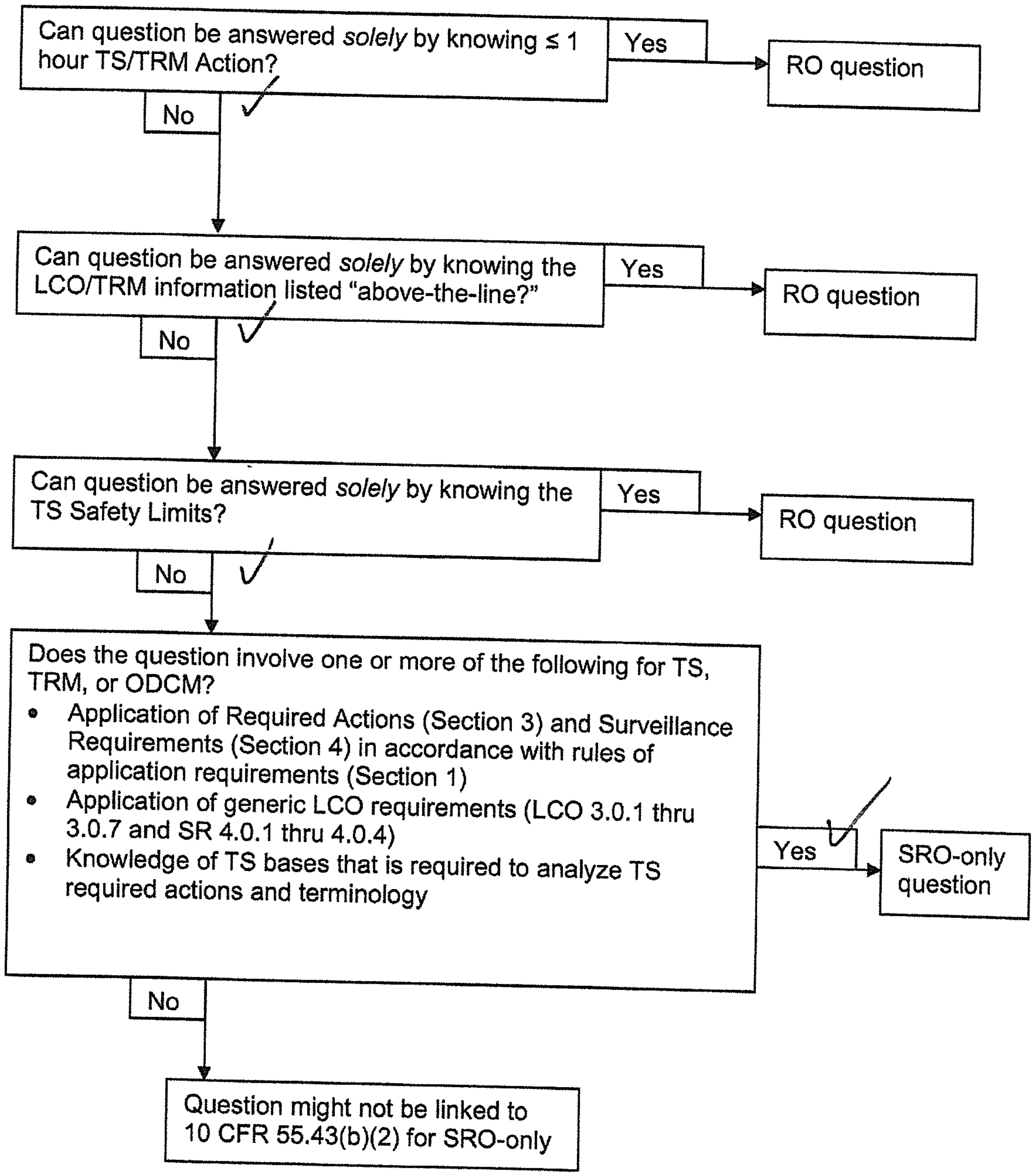
TYPE OF EXPOSURE	ANNUAL ADMINISTRATIVE GUIDELINE (mrem)		
	TIER 1	TIER 2	TIER 3
Total Effective Dose Equivalent (TEDE)	2,000	4,000	4,500
Deep Dose Equivalent + Committed Dose Equivalent (DDE + CDE)	20,000	40,000	45,000
Lens Dose Equivalent (LDE)	6,000	12,000	13,500
Shallow Dose Equivalent (SDE)	20,000	40,000	45,000

The administrative TIER 2 limit is 2,000 mrem.

Step 8.2.2 "Authorization Required for Assignment of Administrative Exposure Tiers" contains the following:

Q#97 K/A G 2.2.39

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



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 Unit 1 onsite Class 1E AC Electrical Power Distribution System;

b. Two Unit 1 diesel generators (DGs);

c. The swing DG;

d. One Unit 2 DG capable of supplying power to one Unit 2 Standby Gas Treatment (SGT) subsystem required by LCO 3.6.4.3, "SGT System;"

e. One qualified circuit between the offsite transmission network and the Unit 2 onsite Class 1E AC Electrical Power Distribution subsystem(s) needed to support the Unit 2 SGT subsystem(s) required by LCO 3.6.4.3;

f. Two DGs (any combination of Unit 2 DGs and the swing DG), each capable of supplying power to one Unit 1 low pressure coolant injection (LPCI) valve load center; and

g. One qualified circuit between the offsite transmission network and the applicable onsite Class 1E AC electrical power distribution subsystems needed to support each Unit 1 LPCI valve load center required by LCO 3.5.1, "ECCS - 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	Perform SR 3.8.1.1 for OPERABLE required offsite circuits.	1 hour AND Once per 8 hours thereafter
	AND		
	A.2	Declare required feature(s) with no offsite power available inoperable when the redundant required feature(s) are inoperable.	24 hours from discovery of no offsite power to one 4160 V ESF bus concurrent with inoperability of redundant required feature(s)
	AND		
	A.3	Restore required offsite circuit to OPERABLE status.	72 hours AND 17 days from discovery of failure to meet LCO 3.8.1.a, b, or c
B. One Unit 1 or the swing DG inoperable.	B.1	Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND		

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Declare required feature(s), supported by the inoperable 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)
	<u>AND</u>	
	B.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	<u>OR</u>	
	B.3.2 Perform SR 3.8.1.2.a for OPERABLE DG(s).	24 hours
	<u>AND</u>	
	B.4 Restore DG to OPERABLE status.	72 hours for a Unit 1 DG with the swing DG not inhibited or maintenance restrictions not met <u>AND</u> 14 days for a Unit 1 DG with the swing DG inhibited from automatically aligning to Unit 2 and maintenance restrictions met <u>AND</u> 72 hours for the swing diesel with maintenance restrictions not met

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 (continued)	AND 14 days for the swing diesel with maintenance restrictions met AND 17 days from discovery of failure to meet LCO 3.8.1.a, b, or c
C. One required Unit 2 DG inoperable	C.1 Perform SR 3.8.1.1 for OPERABLE required offsite circuit(s).	1 hour AND Once per 8 hours thereafter
	AND	
	C.2 Declare required feature(s), supported by the inoperable DG, inoperable when the redundant required feature(s) are inoperable.	4 hours from discovery of Condition C concurrent with inoperability of redundant required feature(s)
	AND	
	C.3.1 Determine OPERABLE DG(s) are not inoperable due to common cause failure.	24 hours
	OR	
	C.3.2 Perform SR 3.8.1.2.a for OPERABLE DG(s).	24 hours

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE------------- Required Action A.2 shall be completed if this Condition is entered. ----------------------------------- Requirements of the LCO not met in MODES 1, 2, and 3.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	12 hours 36 hours
C. -------------NOTE------------- Required Action C.2 shall be completed if this Condition is entered. ----------------------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1

SRs shall be met during the MODES or other specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.

SR 3.0.2

The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met.

For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance.

Exceptions to this Specification are stated in the individual Specifications.

SR 3.0.3

If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed > 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.

(continued)

98. G2.3.4 001

A Nuclear Plant Operator (NPO) is being sent out to isolate a radioactive leak on the 158' elevation of the **Unit 1** Reactor Building. Dose rates in the area are significantly higher than normal.

IAW 60AC-HPX-001-0, Radiation Exposure Limits, the MAXIMUM Administrative Annual TEDE exposure that the NPO can receive, without requiring written approval from an HP Supervisor, is __________ .

The _________ is the MINIMUM level of qualification necessary to declare a Radiological Event AND will make immediate decisions concerning Emergency Call List notifications.

A. 4,000 mrem;
Shift Manager

B. 4,000 mrem;
Control Room Shift Supervisor

C. 2,000 mrem;
Shift Manager

D✔ 2,000 mrem;
Control Room Shift Supervisor

Description:

IAW 60AC-HPX-001-0, Radiation Exposure Limits, step 8.2.1 lists the following administrative limits:

TYPE OF EXPOSURE	ANNUAL ADMINISTRATIVE GUIDELINE (mrem)		
	TIER 1	TIER 2	TIER 3
Total Effective Dose Equivalent (TEDE)	2,000	4,000	4,500
Deep Dose Equivalent + Committed Dose Equivalent (DDE + CDE)	20,000	40,000	45,000
Lens Dose Equivalent (LDE)	6,000	12,000	13,500
Shallow Dose Equivalent (SDE)	20,000	40,000	45,000

The administrative TIER 2 limit is 2,000 mrem.

Step 8.2.2 "Authorization Required for Assignment of Administrative Exposure Tiers" contains the following:

TIER	PREREQUISITE	AUTHORIZATION REQUIRED
1	Current year estimated or actual exposure documented	None (initial limit)
2	Available exposure confirmed	Written approval from an HP Supervisor
3	Available exposure confirmed	Written approval from the Plant Manager
*To exceed Tier 3	Subject to the requirements of a Planned Special Exposure	Hatch Project Vice President

The SRO must know detailed knowledge of the Tables located in the body of 60AC-HPX-001-0, Radiation Exposure Limits, and when approvals are needed to exceed TIER levels.

Also the SRO must know detailed knowledge of 73EP-RAD-001-0, Section 6.0, which states: The **Control Room Shift Supervisor**, normally in consultation with HP Supervision, must have determined it to be prudent to alert plant personnel to an unusual radiological condition.

The "A" distractor is plausible if the applicant does not remember/confuses the Table values located in the procedure or remembers 4,000 as the TIER 2 value. The second part is plausible if the applicant confuses that since the Shift Manager is the minimum level of management (manager) in the Main Control Room and thinks they can declare the Radiological event. Also plausible since the Shift Manager is responsible for declaring Emergencies and the applicant confusing this with declaring a radiological event.

The "B" distractor is plausible if the applicant does not remember/confuses the Table values located in the procedure or remembers 4,000 as the TIER 2 value. The second part is correct.

The "C" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses that since the Shift Manager is the minimum level of management (manager) in the Main Control Room and thinks they can declare the Radiological event. Also plausible since the Shift Manager is responsible for declaring Emergencies and the applicant confusing this with declaring a radiological event.

A. **Incorrect** - See description above.

B. **Incorrect** - See description above.

C. **Incorrect** - See description above.

D. **Correct** - See description above.

References:
NONE

K/A:

2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.
(CFR: 41.12 / 43.4 / 45.10) 3.2 3.7

SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

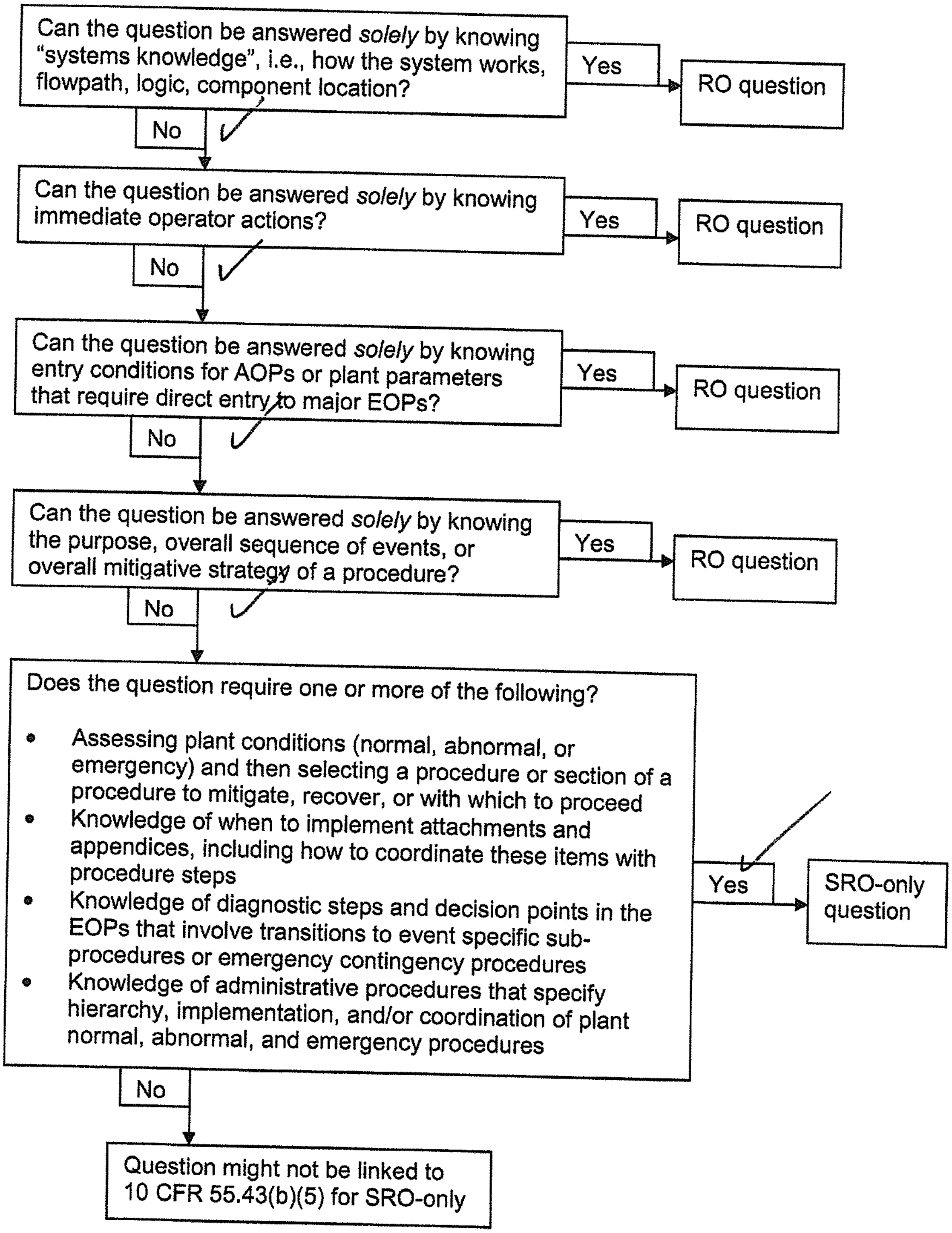
LT-LP-30008, Radiation Control Administration Procedures And Instrumentation,
LO LT-30008.001

References used to develop this question:

60AC-HPX-001-0, Radiation Exposure Limits, **Ver. 10.8**
73EP-RAD-001-0, Radiological Event, **Ver. 2.1**
Item 1: SRO ONLY Guideline
Item 2: 60AC-HPX-001-0, Admin Exposure Control Section, **Ver. 10.8**
Item 3: 73EP-RAD-001-0, Personnel Req. Section, **Ver. 2.1**

Q#98 K/A G2.3.4

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



8.2 ADMINISTRATIVE EXPOSURE CONTROL

This subsection establishes Administrative Exposure Guidelines for the control of occupational radiation exposure. These guidelines are not to be exceeded unless authorized in accordance with 8.2.2, Authorization Required for Assignment of Administrative Exposure Tiers.

8.2.1 Annual Administrative Guidelines

NOTE:	No person under 18 years of age will be permitted to receive occupational radiation exposure at Plant Hatch.

TYPE OF EXPOSURE	ANNUAL ADMINISTRATIVE GUIDELINE (mrem)		
	TIER 1	TIER 2	TIER 3
Total Effective Dose Equivalent (TEDE)	2,000	4,000	4,500
Deep Dose Equivalent + Committed Dose Equivalent (DDE + CDE)	20,000	40,000	45,000
Lens Dose Equivalent (LDE)	6,000	12,000	13,500
Shallow Dose Equivalent (SDE)	20,000	40,000	45,000

8.2.2 Authorization Required for Assignment of Administrative Exposure Tiers

Implicit in the ALARA philosophy is management review of occupational radiation exposure. This subsection lists the required approvals that allow an individual to be assigned an Administrative Exposure Tier.

TIER	PREREQUISITE	AUTHORIZATION REQUIRED
1	Current year estimated or actual exposure documented	None (initial limit)
2	Available exposure confirmed	Written approval from an HP Supervisor
3	Available exposure confirmed	Written approval from the Plant Manager
*To exceed Tier 3	Subject to the requirements of a Planned Special Exposure	Hatch Project Vice President

* up to, but not to exceed, Federal limits specified in 8.1.1

8.2.3 Prenatal Radiation Exposure

4.0 REQUIREMENTS

4.1 PERSONNEL REQUIREMENTS

4.1.1 Health Physics personnel are required to perform radiological monitoring upon receipt of information regarding abnormal radiological conditions existing in the plant.

4.1.2 Control Room Shift Supervisor is the minimum level of qualification necessary to declare a Radiological Event AND will make immediate decisions concerning Emergency Call List notifications.

4.1.3 Operations supervisory personnel will evaluate radiological condition information for possible emergency classification and will ensure appropriate public address announcements are made to advise plant personnel of changing plant radiological conditions.

4.2 MATERIAL / EQUIPMENT REQUIREMENTS

4.2.1 Equipment, as specified in appropriate plant procedures, necessary to perform radiation, contamination and airborne radioactivity surveys.

4.2.2 Dosimetry as deemed appropriate by Health Physics.

4.2.3 Respiratory protection as deemed appropriate by Health Physics.

4.2.4 Protective clothing as deemed appropriate by Health Physics.

4.3 SPECIAL REQUIREMENTS

ONLY an HP & CHEM Department representative OR a Shift Supervisor may authorize entry without an RWP into an area which would normally require an RWP for entry; and ONLY when critical immediate action is required.

99. G2.4.38 001

A **Unit 1** Primary system line break is discharging to the environment and CANNOT be isolated from the Main Control Room.

The Shift Manager has declared a General Emergency due to dose rates exceeding 1,000 mr/hr beyond the site boundary.

An Authorization To Exceed 10CFR20 exposure limits will be needed to rescue an injured operator attempting to isolate the line.

There are NO volunteers to perform the life saving rescue.

The Health Physic Manager has arrived in the TSC.

The OSC & TSC facilities have NOT been activated at this time.

IAW NMP-EP-110, Emergency Classification Determination and Initial Action, the ________ is responsible for authorizing EXCEEDING the 10CFR20 exposure limits.

IAW 73EP-EIP-017-0, Emergency Exposure Control, and with the above conditions, the HIGHEST listed exposure that can be authorized for the rescue is ________ .

A. Shift Manager;
10 REM.

B.✓ Shift Manager;
25 REM.

C. Health Physic Manager;
10 REM.

D. Health Physic Manager;
25 REM

Description:

IAW NMP-EP-110, Emergency Classification Determination and Initial Action, step 5.1.1, 5th bullet, requires the **Shift Manager** (**Emergency Director**) to have the responsibility to authorize plant personnel to exceed 10CFR20 radiation exposure limits.

IAW 73EP-EIP-017-0, Emergency Exposure Control, step 7.4.1, the Exposure Limit is 25 REM for life saving rescue or protection of large populations.

IAW 60AC-HPX-001-0, Radiation Exposure Limits, the HP Manager is in the chain of progression for authorizing exceeding Tier 1 limits.

The SRO must realize what constitues an emergency exposure and then determine who provides the authorization and the value based on a table inside of the procedure. ED responsibilites are above the RO knowledge level.

The "A" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses 10 and 25 Rem and would be correct if asking for protecting equipment.

The "C" distractor is plausible since the Health Physic Manager is responsible for radiation protections at Plant Hatch and will be in the TSC asssiting with radiation decisions, but can NOT authorize exceeding 10CFR20 limits (NMP-EP-110 step 5.1.1). The second part is plausible if the applicant confuses 10 and 25 Rem and would be correct if asking for protecting equipment.

The "D" distractor is plausible since the Health Physic Manager is responsible for radiation protections at Plant Hatch and will be in the TSC asssiting with radiation decisions, but can NOT authorize exceeding 10CFR20 limits (NMP-EP-110 step 5.1.1). The second part is correct.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NONE

K/A:

2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.
(CFR: 41.10 / 43.5 / 45.11) 2.4 4.4

SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

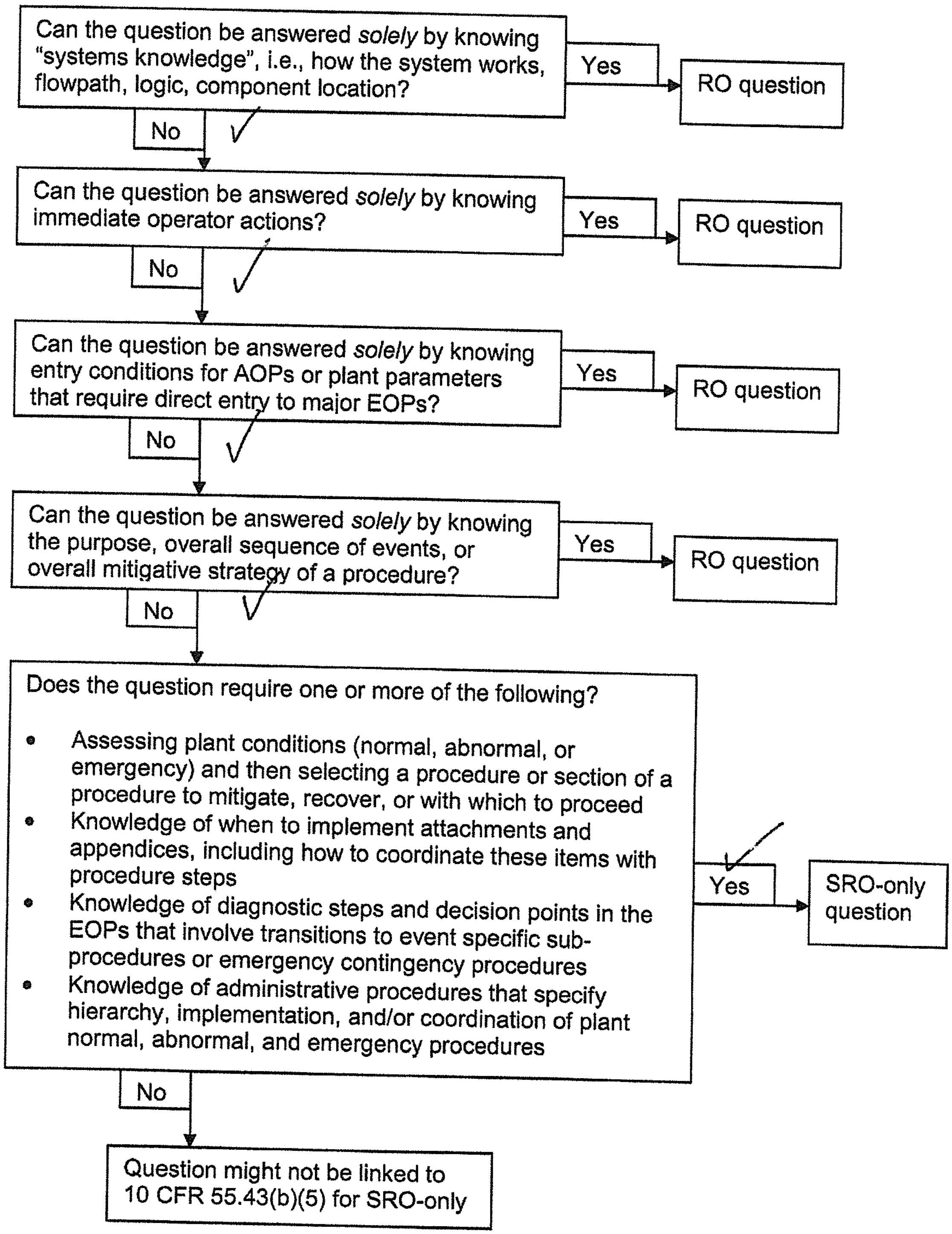
EP-LP-20101, Initial/Terminating Activities, EO 001.017.A.02
EP-LP-20102, Protective Actions, 001.087.A.12

References used to develop this question:

NMP-EP-110, Emergency Classification Determination and Initial Action, **Ver. 5.0**
73EP-EIP-017-0, Emergency Exposure Control, **Ver. 4.0**
2007 HOPE CREEK NRC EXAM Q#99
Item 1: SRO ONLY Guideline
Item 2: NMP-EP-110, , **Ver. 5.0**
Item 3: 73EP-EIP-017-0, , **Ver. 4.0**
Item 4: 2007 HOPE CREEK NRC EXAM Q#99

Q#99 K/A G2.4.38

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



1.0 PURPOSE

The purpose of this procedure is to provide instructions for the classification of off-normal events into one of four emergency classification levels. This procedure establishes the methodology for emergency classification and delineates the initial actions required by the Emergency Director.

2.0 APPLICABILITY

This procedure applies to emergency classification determinations and associated initial responses. This procedure will be utilized for actual emergencies, emergency drills/exercises, or training as required. This procedure is applicable to all SNC sites.

3.0 RESPONSIBILITIES

3.1 Emergency Director (ED)

1. The ED has the following non-delegable responsibilities:

   - The decision to declare, escalate, or terminate emergency classifications.
   - The decision to notify offsite emergency response agencies.
   - The decision to recommend protective actions to offsite authorities.
   - The decision to request federal assistance.
   - Authorization for plant personnel to exceed 10CFR20 radiation exposure limits.
   - Authorization for use of potassium iodide (KI) tablets during a declared emergency.
   - The decision to dismiss nonessential personnel from the site at an ALERT or higher emergency classification.

2. The ED has the following delegable responsibilities:

   - **Maintaining** communications with offsite authorities regarding all aspects of emergency response.
   - **Providing** overall direction for management of procurement of site-needed materials, equipment, and supplies, documentation, accountability, and security function.
   - **Directing** the notification AND activation of the emergency organization; including emergency response facility activation.
   - **Coordinating** AND directing emergency operations.

7.4 EMERGENCY EXPOSURE GUIDELINES

7.4.1 The Emergency Director will establish the exposure limits for the emergency response personnel based on the following Emergency Response Personnel Exposure Guides:

NOTES:	• These guidelines do not establish a rigid upper limit of exposure. The Emergency Director may use his/her judgment in establishing the appropriate limit. • No thyroid limit is specified for lifesaving action since the complete loss of the thyroid may be considered an acceptable risk for saving a life; however, thyroid exposure must be minimized through the use of respiratory protection and/or KI tablets.

EMERGENCY RESPONSE PERSONNEL EXPOSURE GUIDES

Dose Limit* (REM)	*Activity*	*Condition*
5	all	n/a
10	protecting valuable property	lower dose not practicable
25	life saving or protection of large populations	lower dose not practicable
>25	life saving or protection of large populations	only on a voluntary basis to persons fully aware of the risks involved

* This limit is expressed as the sum of the effective dose equivalent (EDE) and the committed effective dose equivalent (CEDE). The lens of the eye will normally be limited to three (3) times the values AND doses to other organs (including skin and extremities) will normally be limited to ten (10) times the listed value.

7.4.2 Review the qualifications of the volunteer emergency response personnel to ascertain which volunteers would have the highest probability of completing the rescue while accumulating the least exposure.

7.4.3 Review the exposure history of the emergency response personnel for current accumulated exposure levels.

Question: 99 Answer: A

1 Pt(s) Given the following:

- A Site Area Emergency was just declared 20 minutes ago due to a primary system line break that is discharging to the environment and it CANNOT be isolated from the control room.
- TSC & EOF personnel are beginning to arrive at their facilities; however, these facilities have NOT been fully activated at this time.
- Operators are developing a plan in the OSC to manually isolate the line in order terminate the release.
- An Emergency Dose Authorization will be needed to isolate the line.
- The Radiological Assessment Coordinator (RAC) has arrived in the TSC.
- The Emergency Duty Officer (EDO) has NOT arrived in the TSC, and the EDO CANNOT be reached by phone.

Who can authorize the Emergency Exposure in the absence of the EDO and what is the Planned Emergency Exposure Limit (PEEL)?

A. The Shift Manager is empowered to authorize up to 25 REM.

B. The Shift Manager is empowered to authorize up to 75 REM.

C. The RAC is empowered to authorize up to 25 REM.

D. The RAC is empowered to authorize up to 75 REM.

Distracter Analysis:

A. Correct: NC.EP-EP.ZZ-0304 states that the Shift Manager has the responsibility to authorize Emergency Exposures until the EDO assumes his or her responsibilities. The Planned Emergency Exposure Limit is 25 REM for accident mitigation and isolating the line is an accident mitigation action.

B. Incorrect: 75 REM is the PEEL limit for saving a life. Isolating the line is NOT a life saving action.

C. Incorrect: The RAC is NOT authorized to grant permission for an Emergency Exposure.

D. Incorrect: The RAC is NOT authorized to grant permission for an Emergency Exposure. 75 REM is the PEEL limit for saving a life. Isolating the line is NOT a life saving action.

Level: SRO Exam
CFR 55.43(b)(4) & (5)

Lesson Plan Objective: ????

Source: New

Level of knowledge: Memory

Reference(s):
NC.EP-EP.ZZ-0304, "Operational Support Center (OSC) Radiation Protection Response"

KA: G2.4.38
2.4.38 Ability to take actions called for in the facility emergency plan / including (if required) supporting or acting as emergency coordinator. (CFR: 43.5 / 45.11) IMPORTANCE RO 2.2 SRO 4.0

Comment / Change Record:
None

100. G2.4.4 001

Unit 1 is at 100% power with the 1B EDG tagged out of service for repairs.

The following sequence of events occurs:

- o 11:30 - Offsite power is lost to Unit 1 ONLY
- o 11:35 - SO reports 1A EDG Lube Oil Pressure of 20 psig, lowering 0.5 psig /min
- o 11:45 - 1C EDG trips on Differential Lockout

Which ONE of the choices below completes the following statement?

The EARLIEST listed time which entry into 34AB-R43-001-1, Diesel Generator Recovery, is REQUIRED is __________ .

At 12:05, the HIGHEST required emergency classification is ______ .

Reference Provided

A. 11:40;
an Alert Emergency

B.✓ 11:40;
a Site Area Emergency;

C. 11:45;
an Alert Emergency

D. 11:45;
a Site Area Emergency

Description:

Unit 1 has lost off-site power. The 1A EDG trips on low lube oil pressure (18 psig) because the trip is active now. There are some of the EDG trips that are only active when the EDG is in the "Test" mode of operation. At 11:37 the 1A EDG will trip and the plant will only have the 1C EDG running.

At 11:45, the 1C EDG will trip but the plant was already in an Alert (SA5).

At 12:05 the plant will have exceeded the Site Area Emergency (SS1) Threshold limit.

The operator will select Alert if they thought the 1A EDG did NOT trip.

The SRO must realize what constitues a Site Area Emergency based on SUT/EDG status. EALs are above the RO knowledge level.

The "A" distractor is plausible since the first part is correct. The second part is plausible if the applicant confuses 1A EDG operation and thinks the trip is not active therefore the EDG would not have tripped at 11:37. At 12:05 an Alert would be the highest EAL.

The "C" distractor is plausible if the applicant confuses the active trips on the EDG and thinks the 1C EDG is the first EDG to trip, thus requiring entry into the abnormal procedure. The second part is plausible if the applicant confuses 1A EDG operation and thinks the trip is not active therefore the EDG would not have tripped at 11:37. At 12:05 an Alert would be the highest EAL.

The "D" distractor is plausible if the applicant confuses the active trips on the EDG and thinks the 1C EDG is the first EDG to trip, thus requiring entry into the abnormal procedure. The second part is plausible if the applicant confuses the EAL and thinks since 1B & 1C are not running, the SS1 EAL is in effect.

A. **Incorrect** - See description above.

B. **Correct** - See description above.

C. **Incorrect** - See description above.

D. **Incorrect** - See description above.

References:
NMP-EP-110-GL02, "Emergency Classification & Initial Actions", Attachment 2 "Hot" Initiating Condition Matrix Evaluation Chart, AC Power Section ONLY.

K/A:

2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.
(CFR: 41.10 / 43.2 / 45.6) 4.5 4.7

SRO only because of link to 10CFR55.43 (5): Assessment of facility conditions and selection of appropriate procedure, recalling the action in the body of procedure and when to take the action.

LESSON PLAN/OBJECTIVE:

EP-LP-20101, Initial/Terminating Activities, TO 001.017.A

References used to develop this question:

34SO-R43-001-1, Diesel Generator Standby AC System, **Ver. 26.0**
NMP-EP-110-GL02 – HNP EALs - ICs, Threshold Values and Basis, **Ver. 2.0**
Item 1: SRO ONLY Guideline
Item 2: 34SO-R43-001-1, Limitations, **Ver. 26.0**
Item 3: NMP-EP-110-GL02 **Ver.2.0**

End of test

Q#100 K/A G2.4.4

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

5.2 LIMITATIONS

5.2.1 Voltage must NOT exceed 4400 volts on any diesel generator phase.

5.2.2 The diesel generator will be run for one (1) hour after cranking for any reason, except WHEN requested to satisfy a vendor OR due to equipment malfunction.

5.2.3 The following conditions will trip a diesel generator:

PARAMETER	SETPOINT
Lube Oil Pressure Low	18 psig
Lube Oil Temperature High	230°F*
Jacket Coolant Temperature High	205°F*
Jacket Coolant Pressure Low	10 psig*
Crankcase Pressure High	0.5 inches H_2O*
Engine Overspeed	1000 rpm
Start Failure	< 250 rpm and <6 psig oil pressure 7 seconds after diesel is started
Differential Current	N/A
Reverse Current	N/A *

* These trips are only applicable in the TEST mode.

5.2.4 Energizing a diesel generator's test relays results in the following:

- Locks out associated diesel generator emergency start
- Prevents AUTO closure of associated Diesel Generator output breaker
- Allows paralleling of associated Diesel Generator AND EITHER its normal OR alternate power supply
- Prevents MANUAL closure of start-up Transformer Supply breakers to 4160V Busses 1A, 1B, 1C, AND 1D. (Auto fast transfer will still occur)
- Prevents AUTO transfer of associated Emergency 4160V bus to its alternate supply
- Arms additional diesel generator trips as stated in 5.2.3

5.2.5 A LOCA OR LOSP signal will deenergize the diesel generator test relays.

5.2.6 Voltage must NOT exceed 605 volts on any phase on 600V Bus 1C AND 1D.

5.2.7 Diesel Generator frequency must be maintained between 59 AND 61 Hertz.

5.2.8 IF the Diesel Generator trips while tied to the grid, the resulting Governor (Diesel Gen Speed) Setting may NOT be at 900 rpm. Following Diesel Gen trip, the Diesel Gen must be run again using a “slow start” procedure (e.g., 34SV-R43-001-1) to ensure the Governor Setting is correct WHEN the Diesel Gen is shutdown.

CA3 - Loss of All Offsite Power **AND** Loss of All Onsite AC Power to Essential Busses. (Pg. 74)

1. a. Loss of power to or from Startup Auxiliary Transformers (SAT) 1/2C and 1/2D resulting in loss of all off-site electrical power to 4160 VAC Emergency Buses 1/2E, 1/2F, and 1/2G.

   AND

   b. Failure of emergency diesel generators to supply power to emergency busses.

   AND

   c. Failure to restore power to at least one emergency bus within 15 minutes from the time of loss of both offsite and onsite AC power.

<u>CU3</u> - Loss of All Offsite Power to Essential Busses for GREATER THAN 15 Minutes (Pg. 80)
1. a. Loss of power to or from Startup Auxiliary Transformers 1/2C and 1/2D resulting in loss of all off-site electrical power to 4160 VAC Emergency Buses 1/2E, 1/2F, and 1/2G. for greater than 15 minutes **<u>AND</u>** b. At least one emergency diesel generator supplying power to 1/2E, 1/2F, or 1/2G..
<u>CU7</u> - UNPLANNED Loss of Required DC Power for Greater than 15 Minutes. (Pg. 83)
1. a. UNPLANNED loss of DC power to 125/250 VDC Bus 1/2R22-S016 & 1/2R22-S017 indicated by bus voltage less than 105/210 VDC **<u>AND</u>** b. Failure to restore power to at least one required DC bus within 15 minutes from the time of loss.