Draft Submittal

BRUNSWICK OCT/NOV 2004

EXAM 50-325, 324/2004-301
OCTOBER 29, 2004 &
NOVEMBER 2 - 10, 2004

1. Reactor Operator Operator Written Exam

Brunswick Nuclear Plant

Initial Examination **DRAFT** RO/SRO Written

Examination Report Nos.

05000325/2004301 - 05000324/204301

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Prep - September 18 - 22, 2004
Exam Weeks - November, 1 - 5 & 8 - 12, 2004

CONFIDENTIAL
TEST MATERIAL

NRC 2004 RO exam references to be provided:

1. AOP-39 Fig. 2 and Attachment 1 – Question 40, Needed to allow operator to determine extent of DC power loss.
2. ASSD Calculation Sheet 1 – Question 44, Needed to aid operator in performing manual calculation of drywell temperature.
3. AOP-32 Appendix 3 and 4 – Question 46, needed to aid operator in determining cool down rate. (Steam Tables can also be provided.)
4. Attachment 5 EOP User's Guide – Question 52, DWSIL and PSP limit curves needed to assess containment limits.

NOTE: No references will aid applicant in answering other questions and none of the above questions result in being a direct lookup when reference is provided.

BRUNSWICK OCT/NOV 2004

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DRAFT RO WRITTEN EXAM

Figure 2, Unit 2 DC Distribution Figure
Rev 18 - Page 14 of 18
Summary + References

INTENTIONALLY OMITTED
PER SISP REVIEW

ATTACHMENT 1
Page 1 of 3
Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
UA-19 6-3	X										
UA-21 6-3	X										
UA-15 2-3/UA-17 2-3	X										
UA-09 4-2		X			X						
UA-09 5-1		X					X				
UA-09 5-3		X					X				
UA-09 6-2		X									
UA-13 5-7		X					X				
UA-13 5-8		X					X				
UA-07 2-3							X				
UA-07 5-1		X									
UA-07 5-2		X					X				
UA-07 5-4		X									
UA-08 2-3							X				
UA-08 5-1		X									
UA-08 5-2		X					X				
UA-08 5-4		X									
UA-10 2-3							X				
UA-10 5-1		X									
UA-10 5-2		X					X				
UA-10 5-4		X									
UA-11 2-3							X				
UA-11 5-1		X									
UA-11 5-2		X					X				
UA-11 5-4		X									

ATTACHMENT 1
Page 2 of 3
Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
A-06 5-8			X						X		
UA-13 6-10			X		X			X	X	X	
A-03 1-3			X								
A-03 2-2			X						X		
A-01 5-5			X								
A-01 6-4			X								
A-01 2-5			X								
A-01 1-6			X								
A-01 2-8			X								
A-02 4-4			X								
UA-09 2-5				X							
A-04 5-1					X						
A-04 1-8					X						
A-05 1-7					X						
A-05 5-6					X					X	
A-05 3-6					X					X	
A-05 2-6					X					X	
A-05 5-3					X						
A-06 1-6					X						
A-06 5-6					X						
UA-13 4-1					X						
UA-13 6-2					X						

ATTACHMENT 1
Page 3 of 3
Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
UA-13 5-10					X						
UA-20 6-3						X					
UA-22 6-3						X					
UA-16 2-3/UA-18 2-3						X					
UA-05 4-2								X			
A-01 6-5									X		
A-03 1-4									X		
A-03 6-5									X		
A-03 1-6									X		
A-03 2-7									X		
A-02 4-8									X		
A-04 2-8										X	
A-04 6-1										X	
A-05 2-7										X	
A-05 5-4										X	
A-06 2-6										X	
A-06 6-6										X	
UA-13 5-2										X	
UA-13 6-3										X	
A-06 6-4											X
A-06 6-5											X

Calculation Sheet 1

Values Obtained From Recorder CAC-TR-778

80' elev

PT No. 1 ______ ______ ______ ______ ______ ______ ______ ______

x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14

A_____ A_____ A_____ A_____ A_____ A_____ A_____ A_____

28' elev

PT No. 3 ______ ______ ______ ______ ______ ______ ______ ______

x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4

B_____ B_____ B_____ B_____ B_____ B_____ B_____ B_____

13' elev

PT No. 4 ______ ______ ______ ______ ______ ______ ______ ______

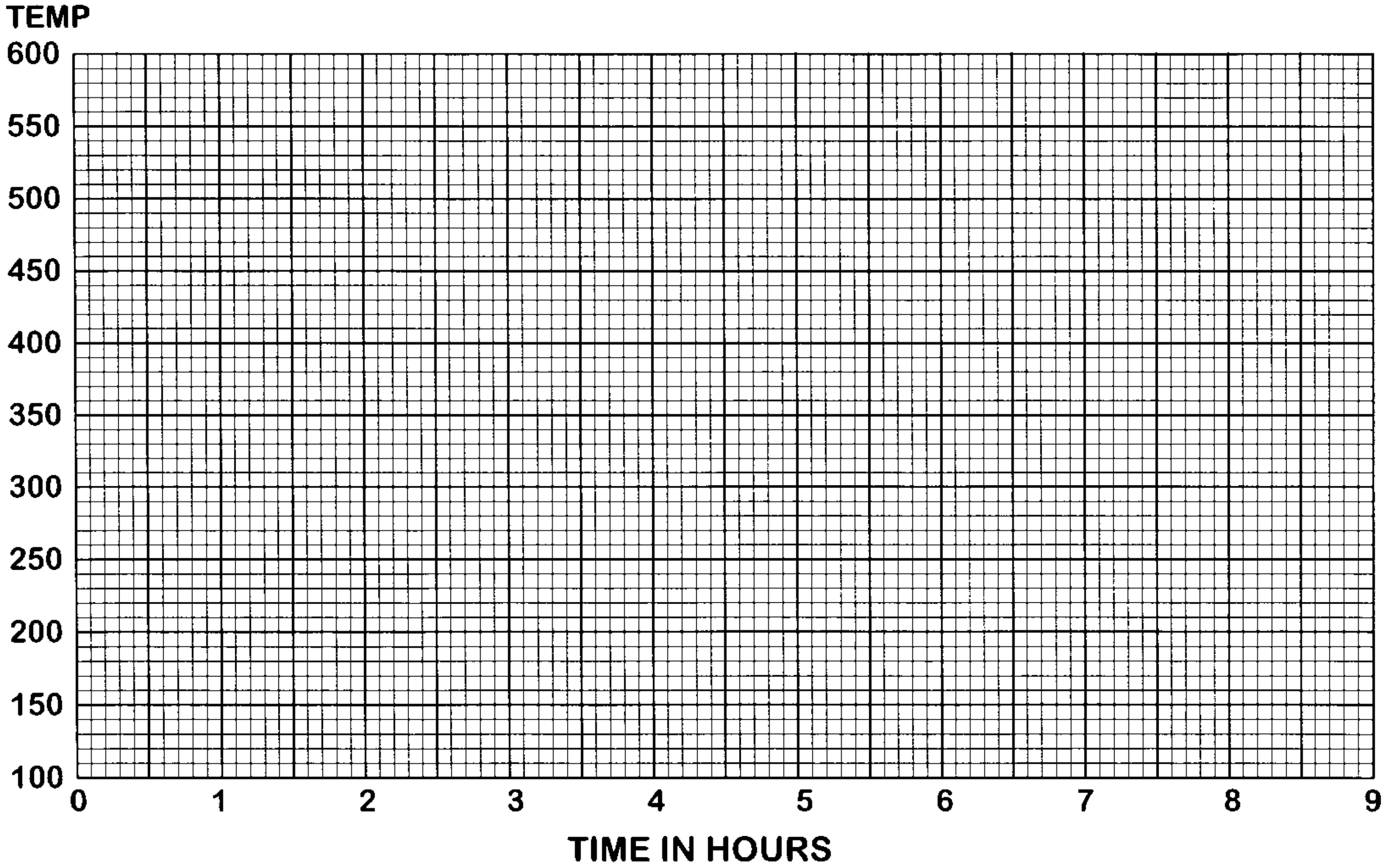
x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46

C_____ C_____ C_____ C_____ C_____ C_____ C_____ C_____

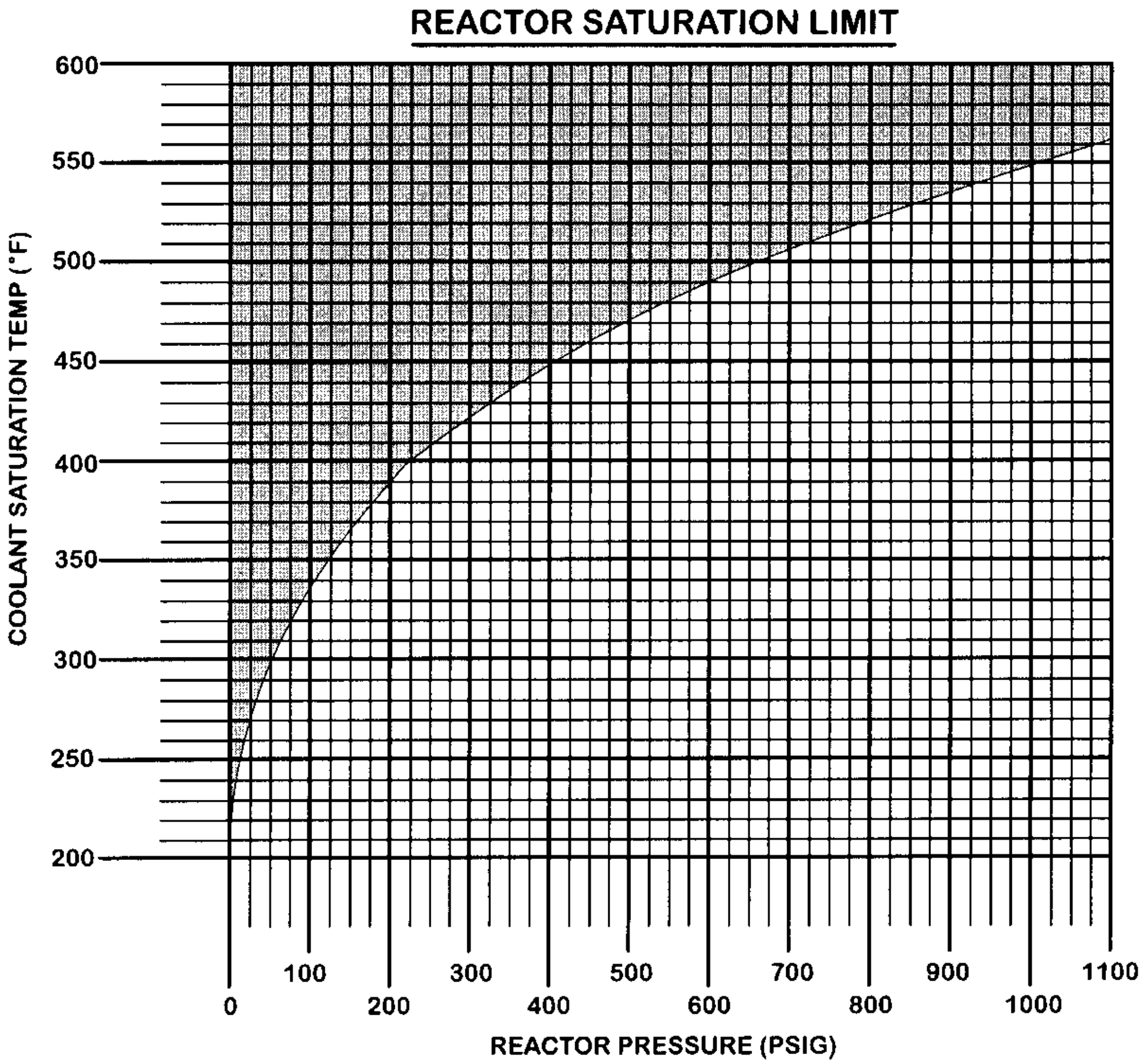
Add the numbers obtained in lines A, B, and C, to obtain average DW temp.

Average
DW Temp ______ ______ ______ ______ ______ ______ ______ ______

ATTACHMENT 13
Page 1 of 1
Cool Down Plot

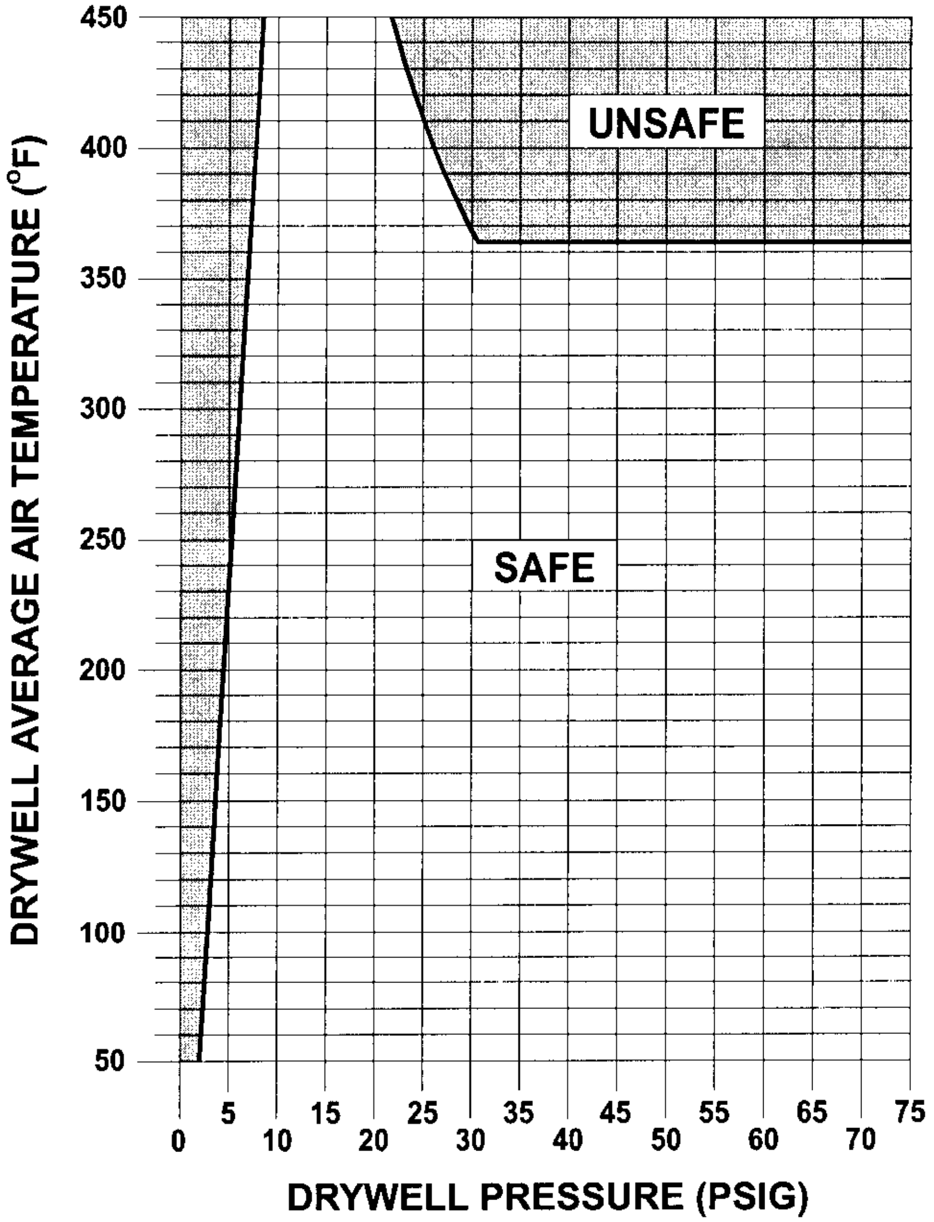


ATTACHMENT 14
Page 1 of 1
Saturation Curve



ATTACHMENT 5 (Cont'd)

FIGURE 1
DRYWELL SPRAY INITIATION LIMIT

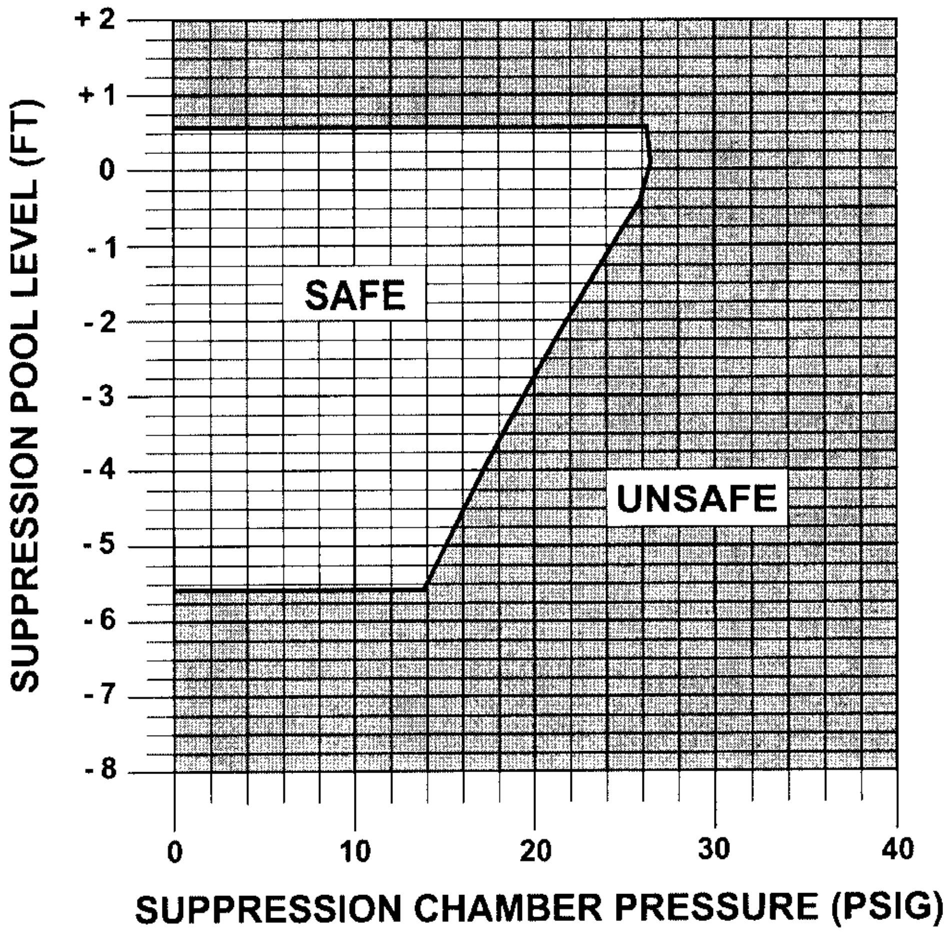


NOTE

DRYWELL AVERAGE AIR TEMPERATURE MAY BE DETERMINED
USING ATTACHMENT 4 OF THE "USER'S GUIDE"

ATTACHMENT 5 (Cont'd)

FIGURE 7
PRESSURE SUPPRESSION PRESSURE



BANK INFORMATION REPORT

for RO NRC2004

Category 5 (Ref Req'd Y or N)	#Items	Title
NO	71	
AOP 32 APP 3 & 4	1	295016AA206 RO Q# 46
Y AOP-39 FIG2 & AT 1	1	295004AA202 RO Q# 40
Y ASSD-02 CALC SH 1	1	295012G2.1.23 RO Q# 44
Y DWSIL & PSPL	1	295024EK301 RO Q# 52

Category 7 (? Cognitive Level)	#Items	Title
C/A	43	
M OR FK	32	

Category 8 (? Source)	#Items	Title
BANK LOI	13	
BANK NRC	2	
MOD. LOI BANK	11	
MOD. NRC	1	
NEW	48	

1. 201003A402 1
Unit Two (2) is at 80% when Control Rod 10-31 is fully withdrawn continuously. From a proper control rod coupling check it is determined that the control rod is uncoupled.

Per the Annunciator Procedure the operator attempts recoupling by a single notch insertion.

Which ONE of the following identifies the expected rod position indication response to a single notch insertion of the uncoupled control rod?

After the single notch insertion is complete the operator should expect the Four Rod Display to indicate:

A. 44

B. 46

C. 48

D. Blank

2. 201006A303 1
Unit Two (2) is in Mode 2 at 5% of rated thermal power with a reactor startup in progress.

The Rod Worth Minimizer (RWM) System, RWM-NUMAC, experiences a hardware failure that results in an inabilty of the RWM to communicate with ERFIS.

Which ONE of the following describes the expected annunciator and alarm signal response from this RWM-NUMAC failure?

The RWM operator display will indicate a:

A. critical self test fault <u>AND</u> annunciator A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE <u>ONLY</u> will actuate.

B. critical self test fault <u>AND</u> annunciators A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE and A-5 (2-2) ROD OUT BLOCK will actuate.

C. non-critical self test fault <u>AND</u> annunciator A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE <u>ONLY</u> will actuate.

D. non-critical self test fault <u>AND</u> annunciators A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE and A-5 (2-2) ROD OUT BLOCK will actuate.

1. 201003A402 001

Unit Two (2) is at 80% when Control Rod 10-31 is fully withdrawn continuously. From a proper control rod coupling check it is determined that the control rod is uncoupled.

Per the Annunciator Procedure the operator attempts recoupling by a single notch insertion.

Which ONE of the following identifies the expected rod position indication response to this single notch insertion of the uncoupled control rod?

After the single notch insertion is complete the operator should expect the Four Rod Display to indicate:

A. 44

B. 46

C.✓ 48

D. Blank

Feedback

Reference SD-08.1 - Rev.1 page 15 - "One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of down-travel is normally provided by the control rod itself as it reaches the backseat position, the index tube can pass this position and actuate the overtravel switch only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at ar time during plant operation." Since Control Rod 10-31 is at the overtravel out position a single notch ir would place the control rod at position 48.

Distractor Analysis

A incorrect - could be selected if candidate did not know position
B and D - incorrect - could be selected if candidate did not und and RPIS information available.

Notes

Removed

SYSTEM: 201003 Control Rod and Drive Mechanism

A4. Ability to manually operate and/or monitor in the contro

(CFR: 41.7 / 45.5 to 45.8)

A4.02 CRD mechanism position: Plant-Specific . . . .

This question matches the k/a in that it measures the RO's knowledge of CRD . operate and monitor CRD position in order to correctly predict the CRD position

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-08.1*08B
Ref Req'd Y or N:	NO	Technical Ref.:	SD-08.1
? Cognitive Level:	C/A	? Source:	NEW

One reed switch is located at each position corresponding to an index tube notch, allowing indication at each latching point. An additional switch is located at each midpoint between the notches, allowing indication of the intermediate positions during drive motion. Thus, indication is provided for each three inches of travel. Also, switches are provided for the full-in and full-out positions.

One additional switch (an overtravel switch) is located at a position below the normal full-out position. Because the limit of down-travel is normally provided by the control rod itself as it reaches the backseat position, the index tube can pass this position and actuate the overtravel switch only if it is uncoupled from its control rod. A convenient means is thus provided to verify that the drive and control rod are coupled after installation of a drive or at any time during plant operation.

Also mounted in the indicator probe at the very top is a Chromel/Alumel thermocouple which is used to monitor drive temperature to assure adequate cooling of the drive seals. The thermocouples are wired out to a temperature recorder (C11/C12-TR-R018) located in the Reactor Building Elev. 20'-0".

2.12 Graphitar Seals

Graphitar is selected for seals and bushings on both the drive piston and the stop piston. The material is inert and has a low friction coefficient when water lubricated. The seals limit the flow of water (leakage) from the CRDM to the reactor vessel and vice versa. Graphitar seals also maintain the differential pressure across the drive piston which is required for drive movement. Since loss of strength is experienced at higher temperatures, the drive is supplied with cooling water to hold temperatures below 250°F. Inadequate venting of the CRDM could also damage the seals by increasing the impact forces applied during movement and in particular, during scrams. The Graphitar is relatively soft, which is advantageous if an occasional particle of foreign matter reaches a seal. The resulting scratches in the seal reduce sealing efficiency until they are worn smooth, but the drive design can tolerate considerable water leakage past the seals into the reactor vessel.

2. 201006A303 001

Unit Two (2) is in Mode 2 at 5% of rated thermal power with a reactor startup in progress.

The Rod Worth Minimizer (RWM) System, RWM-NUMAC, experiences a hardware failure that results in an inabilty of the RWM to communicate with ERFIS.

Which ONE of the following describes the expected annunciator and alarm signal response from this RWM-NUMAC failure?

The RWM operator display will indicate a:

A. critical self test fault and annunciator A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE only will actuate.

B. critical self test fault and annunciators A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE and A-5 (2-2) ROD OUT BLOCK will actuate.

C✓ non-critical self test fault and annunciator A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE only will actuate.

D. non-critical self test fault and annunciators A-5 (5-2) ROD BLOCK RWM/RMCS SYS TROUBLE and A-5 (2-2) ROD OUT BLOCK will actuate.

Feedback

Reference APP A-5 rev. 47 page 67
AUTO ACTIONS
1. Rod movement of selected rod is prohibited for critical RWM SELF-TEST failures.

NOTE: RWM-NUMAC will utilize the annunciator to inform the operator of hardware failures (whether they are critical or noncritical) and the existence of movement errors (if they are not selected).
NOTE: A critical self-test fault is one which renders RWM-NUMAC unable to perform its rod movement/sequence enforcement function.
NOTE: A noncritical self-test fault is one which does not interfere with RWM-NUMAC's rod movement/sequence enforcement function but may inhibit other auxiliary RWM-NUMAC functions (ex. ability to drive the rod group backlighting or ability to communicate with the ERFIS computer).

Distractor analysis
A and B - incorrect, loss of ERFIS communication does not give a critical self test fault. Candidate may select these distractors if he/she assumes that RWM receives RPIS input from ERFIS. Actually, RWM provides output to ERFIS and receives no input from ERFIS as part of the design criteria.
D - incorrect since A-5 (2-2) ROD OUT BLOCK is only received for critical self test faults.

Notes

SYSTEM: 201006 Rod Worth Minimizer System (RWM) (Plant Specific)
A3. Ability to monitor automatic operations of the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) including:
(CFR: 41.7 / 45.7)
A3.03 Annunciator and alarm signals: P-Spec(Not-BWR6) 3.1 3.0

This question matches the k/a in that it measures the RO's knowledge of the impact that a loss of ERFIS communication will have on RWM automatic operation and the associated alarms.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.1	Facility Objective:	CLS-LP-07.1*016
Ref Req'd Y or N:	NO	Technical Ref.:	APP A-5 5-2
? Cognitive Level:	C/A	? Source:	NEW

ROD BLOCK RWM/RMCS SYS TROUBLE

AUTO ACTIONS

1. Rod movement of selected rod is prohibited for critical RWM SELF-TEST failures.

CAUSES

1. Selecting an incorrect rod (will be accompanied by a RWM select error).
2. Withdraw block.
3. Insert block.
4. RWM-NUMAC system integrity check failure.
5. RWM-NUMAC self-test failure (critical or noncritical).
6. Various RMCS failures.

OBSERVATIONS

1. Selected rod will not move (if a rod placement error exists and is not selected).
2. Selected rod will not move (if a critical RWM-NUMAC self-test failure exists).
3. RWM-NUMAC operator display self-test window indicates FAULT and/or blocks window indicates INSERT/WITHDRAW blocks.
4. Power level has been reduced to below the LPAP and the rod pattern is not yet compatible with the RWM-NUMAC's selected sequence. This information message will clear after 30 seconds even if sequence incompatibility still exists.

ACTIONS

1. Utilize the RWM-NUMAC "MESSAGES" and "SELF-TEST" screens to determine the cause of the problem.

NOTE: RWM-NUMAC will utilize the annunciator to inform the operator of hardware failures (whether they are critical or noncritical) and the existence of movement errors (if they are not selected).

NOTE: A critical self-test fault is one which renders RWM-NUMAC unable to perform its rod movement/sequence enforcement function.

NOTE: A noncritical self-test fault is one which does not interfere with RWM-NUMAC's rod movement/sequence enforcement function but may inhibit other auxiliary RWM-NUMAC functions (ex. ability to drive the rod group backlighting or ability to communicate with the ERFIS computer).

ACTIONS (Continued)

a. Press the ETC softkey (Key 4) until MESSAGES is observed over the first softkey.
b. Press the MESSAGES softkey (Key 1).
c. Note latest message and any current reasons for blocks messages
   (1) If SELF-TEST FAILURE is indicated, use SELF-TEST screen on the RWM-NUMAC to diagnose the reason for the failure (Panel P607).
d. Return to the normal RWM-NUMAC screen by pressing the EXIT softkey (Key 4).

2. Contact the Reactor Engineers. Correct any rod movement errors in accordance with the Reactor Engineer's recommendation. Observe annunciator and block indications clear.
3. If RWM-NUMAC remains failed or a circuit malfunction is suspected, refer to Technical Specification 3.1.6 for operation with RWM bypassed.
4. If a failed reed switch has caused the rod block(s), it will be necessary to use the SUBSTITUTE POSITION screen at the operator display to substitute the correct position. RWM-NUMAC will present the inferred position for that rod based on last known position, direction of movement, and odd notches passed for use as the "value to substitute" (refer to OP-07).
5. If RWM Operator displays mode "unknown", ensure a WR/JO is prepared (GE SIL 540 describes a mode "unknown" condition that occurs following a power interruption of longer than 20 milliseconds but shorter than 5 seconds).

DEVICE/SETPOINTS

N/A

POSSIBLE PLANT EFFECTS

1. If the RWM-NUMAC is bypassed, a technical specification LCO may result.
2. This annunciator is required to be operable to support RWM-NUMAC System operability; annunciator inoperability will result in a LCO.

REFERENCES

1. LL-9364 - 75
2. Technical Specification 3.1.6
3. OP-07, RMCS Operating Procedure
4. EER 88-0259, Using the RWM-NUMAC Deferred Position Capability
5. GE SIL 540, NUMAC Mode "Unknown" Condition

3. 202002G2.1.2 001

A single RO is monitoring Unit One (1) and the SRO has stepped out after turning over with the Unit Two (2) SRO. The RO determines that a minor recirculation flow adjustment, "bump down", to maintain 100% reactor power is required.

Which ONE of the following describes the responsibilities of the RO during a situation as described above, per Operations Performance Standards listed in Attachment 19 of 0OI-01.02, Shift Routine and Operating Practices?

In order to perform this minor adjustment to recirculation flow the RO:

A.✓ must obtain permission from the SRO and Concurrent Verification is required.

B. must obtain permission from the SRO but Concurrent Verification is not required.

C. does not need to obtain permission from the SRO but Concurrent Verification is required.

D. does not need to obtain permission from the SRO and Concurrent Verification is not required.

Feedback

Reference 0OI-01.02 Rev. 35 page 91

Distractor analysis

A - CORRECT - SRO PERMISSION and concurrent verification are required for recirc speed manipulation.

B - INCORRECT - Concurrent verification is required even though flow bump is minor.

C and D - INCORRECT -SRO PERMISSION and concurrent verification are required for recirc speed manipulation even though flow bump is minor.

Notes

SYSTEM: 202002 Recirculation Flow Control System

2.1.2 Knowledge of operator responsibilities during all modes of plant operation.

(CFR: 41.10 / 45.13)

IMPORTANCE RO 3.0 SRO 4.0

This question matches the k/a in that it measures the RO's knowledge of generic responsibilities listed in 0OI-01.02 for operating the reactor recirculation flow control system.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.0	Facility Objective:	CLS-LP-201-D*24B
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-01.02
? Cognitive Level:	M OR FK	? Source:	NEW

C. Control Rod and Recirc Manipulations (Reactivity Adjustments) (Continued)

- Communicates the control rod selected, starting position, movement to be taken, limitations on movement, and final position to the Concurrent Verifier.
- Obtains concurrence of Concurrent Verifier before beginning the move.
- Observes plant response to movement commands.
- When moving a control rod four notches or more, stops one notch short, then single notches to final position unless control rod is going to position "48" (full out).
- When moving a control rod three notch or less, performs separate single notch moves.
- Verifies settle light is extinguished before relinquishing controls.
- Does **NOT** select another control rod until three seconds after control rod settle light has extinguished.
- Periodically checks position on applicable Power to Flow map.
- Ensures Concurrent Verifier performs duties per OPS-NGGC-1306.
- Updates and signs documentation as steps are completed.

- For Recirculation Speed Manipulations:
  - Manipulates recirculation speed only with SRO permission.
  - Ensures Concurrent Verifier is available, prepared, and in place.
  - Communicates to Concurrent Verifier direction of movement, component to be manipulated, amount of adjustment, and stopping point.
  - Obtains concurrence of Concurrent Verifier before commencing adjustments.
  - Operates recirculation speed controls one at a time, allows time for plant and machine response.
  - Observes redundant plant indications while performing adjustments.
  - Notifies SRO when recirculation speed adjustments are completed.
  - Periodically checks position on "Power to Flow" map.

4. 203000A304 001

Unit Two (2) was at 100% RTP when a small break LOCA concurrent with a loss of all high pressure makeup capability to the reactor occurred. The Reactor is being depressurized using the SRVs due to inability to maintain RPV level above TAF. All RHR and Core Spray pumps have started as required with normal indications. Reactor pressure is approximately 400 psig at this time.

Concerning the "A" RHR system only.

Which ONE of the following describes the expected sytem parameters and configuration?

A.✓ RHR "A" system flow indicates 0 gpm, “A” and "C" RHR Pump discharge pressures are approximately 200 psig. RHR pump "A" and "C" minimum flow valve is OPEN.

B. RHR “A” system flow indicates slightly less than 2300 gpm, “A” and "C" RHR Pump discharge pressures are approximately 200 psig. RHR pump "A" and "C" minimum flow valve is OPEN.

C. RHR “A” system flow indicates slightly greater than 2300 gpm, “A” and "C" RHR Pump discharge pressures are approximately 400 psig. RHR pump "A" and "C" minimum flow valve is CLOSED.

D. RHR “A” system flow indicates approximately 17,000 gpm, “A” and "C" RHR Pump discharge pressures are approximately 400 psig. RHR pump "A" and "C" minimum flow valve is CLOSED.

Feedback

References: SD-17, Residual Heat Removal System, Rev. 6, pg.11 & 12

Original Question

Unit Two (2) was at 100% RTP when a small break LOCA concurrent with a loss of all high pressure makeup capability to the reactor occured. The Reactor is being depressurized using the SRVs due to level not being able to be maintained above TAF. All RHR and Core Spray pumps have started as required with normal indications. Reactor pressure is approximately 400 psig at this time.

Concerning the "A" RHR system only, which ONE of the following describes the expected sytem parameters and configuration?

A. Both LPCI “A” injection valves OPEN, “A” system flow indicates 0 gpm, “A” and "C" RHR Pump discharge pressures are approximately 200 psig.

B. Both LPCI “A” injection valves CLOSED, “A” system flow indicates 0 gpm, “A” and "C" RHR Pump discharge pressures are approximately 200 psig.

C. Both LPCI “A” injection valves OPEN, “A” system flow indicates 12,000 gpm, “A” and "C" Pump discharge pressures are approximately 400 psig.

D. Both LPCI “A” injection valves OPEN, “A” system flow indicates 2300 gpm, “A” and "C" Pump discharge pressures are approximately 200 psig.

Distractor Analysis

A is correct as RHR pumps would be running on minimum flow which is not indicated on the system flow indicator. Discharge shutoff head is ~ 200 psig.

B is incorrect as no flow would be indicated on the system flow indicator. Candidate may select this as > 2300 gpm is where the min. flow valves close.

C is incorrect, candidate may read this as RHR system flow is just starting to inject if he did not know that Shutoff head of RHR pumps is 200 psig vs. 400 psig.

D. is incorrect and this describes normal injection which would not occur with RPV at 400 psig.

Notes

SYSTEM: 203000 RHR/LPCI: Injection Mode (Plant Specific)
A3. Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including:
(CFR: 41.7 / 45.7)
A3.04 System flow 3.8 3.7

This question matches the k/a in that it measures the RO's knowledge of an event that requires LPCI initiation. The operator must be able to verify LPCI response by monitoring for proper system flow.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.8	Facility Objective:	CLS-LP-017*007
Ref Req'd Y or N:	NO	Technical Ref.:	SD-17
? Cognitive Level:	C/A	? Source:	MOD. NRC

Once the core is flooded to at least two-thirds core height, one RHR or Core Spray Pump is normally required to make up for Jet Pump throat to diffuser slip joint leakage. The other pumps are stopped so that the emergency power (if there is a loss of offsite power) that was being used by these pumps may be shifted to other plant loads including RHR Service Water Booster Pumps. One RHR Pump and Heat Exchanger are typically placed in the Containment Cooling Mode after a line break in the drywell as per the Emergency Operating Procedures (EOPs).

Upon automatic initiation of the A(B) Loop of LPCI, the associated pumps should start 10 seconds after the initiation signal is received. If a loss of power has occurred resulting in the Emergency Diesel Generators powering the pumps, the pumps will start 10 seconds after the associated diesel ties onto the emergency bus.

The A(B) and C(D) RHR Pumps take a suction from the Suppression Pool through the normally open motor operated Suppression Pool Suction Valve, E11-F020A(B), and the associated motor-operated RHR Pump Suppression Pool Suction Valve, E11-F004A(B) or E11-F004C(D), and discharge through the Pump Discharge Check Valve, E11-F031A(B) or E11-F031C(D). The discharge check valves are designed to prevent backflow through the pump and to maintain a water leg in the discharge piping.

With the RHR and CS Pumps running, indicated pump discharge pressure on CS should increase to approximately 305 psig and RHR should be about 200 psig. As reactor pressure decreases to approximately 410 psig, the LPCI Inboard Injection Valve, E11-F015A(B), should automatically open. As reactor pressure continues to decrease, the discharge of the RHR Pumps should overcome reactor pressure at approximately 200 psig, allowing the flowpath to continue from the RHR Pumps' discharge check valve directly into the Reactor Vessel through the normally open LPCI Outboard Injection Valve, E11-F017A(B), the LPCI Inboard Injection Valve, E11-F015A(B), the LPCI Injection Line Check Valve, E11-F050A(B), the locked open LPCI Manual Injection Valve, E11-F060A(B), and into the Reactor Recirculation System discharge lines. Once reactor pressure is reduced to approximately 20 psig, RHR flow should reach approximately 17,000 gpm per operating loop with two pumps.

The LPCI Outboard Injection Valve, F017A(B), is a throttle valve which may be adjusted to control flow into the vessel, whereas the Inboard Injection Valve, E11-F015A(B), is designed for either full open or full close service. E11-F017A(B) is normally open, but with an automatic open signal present, this valve cannot be closed or throttled for 5 minutes to ensure a discharge path exists from the pumps to the vessel. E11-F015A(B), cannot be closed as long as the LPCI initiation signal is present. In addition, the RHR heat exchanger is automatically bypassed via the RHR Heat Exchanger Bypass Valve, E11-F048A(B), for the first three minutes to ensure that flow gets to the reactor through the most direct route. During the interval of time when the RHR pumps are operating to restore the reactor vessel level, heat removal is not necessary.

Each LPCI loop is provided with a minimum flow line to the Suppression Pool to protect the pumps from damage due to overheating as a result of low or no-flow operation. This feature allows the pumps to operate with a closed discharge valve, without overheating, by recirculating Suppression Pool water through the minimum flow bypass line.

After the core has been flooded to at least two-thirds core height, only one Core Spray or one RHR pump is required to maintain this level.

1.3.3 Shutdown Cooling (Figure 17-4)

The Shutdown Cooling Mode removes the residual heat from the Reactor to maintain the Reactor in a cold shutdown condition for refueling and maintenance. The system is placed into operation during a reactor shutdown, after a normal cooldown which uses the main condenser, when the reactor vessel pressure reaches 130.8 psig (350°F) or less. The Shutdown Cooling Mode has the capability of completing cooldown to 125°F in less than 21 hours after the control rods have been inserted with the river temperature at 95°F and is capable of maintaining the nuclear system temperature at or below 125°F in order to refuel and service the Reactor.

5. 203000K413 001

Which ONE of the following describes the design feature that provides for prevention of leakage to the environment through RHR heat exchangers?

A.✓ Tube side pressure is maintained a minimum of 15 psig above shell side pressure regardless of cooling water flow rate.

B. Shell side pressure is maintained a minimum of 15 psig above tube side pressure regardless of cooling water flow rate.

C. Tube side pressure is maintained a minimum of 15 psig above shell side pressure ONLY if cooling water flow rate is maintained above 2000 gpm.

D. Shell side pressure is maintained a minimum of 15 psig above tube side pressure ONLY if cooling water flow rate is maintained above 2000 gpm.

Feedback

Reference SD-17 Rev. 6 page 24 and 25.
The pressure of the RHR Service water (tube side), regardless of cooling water flow rate, is maintained 15 psig above the shell side (Reactor water side) pressure. This design ensures that internal leakage will be from the tube side into the shell side, thus preventing the discharge of Reactor water to the environment.
Distractor Analysis
A - correct
B- RHRSW is tube side and is 15 psig higher than shell side at all flow rates
C- RHRSW flow less than 2000 gpm per OP-43 may cause piping water hammer in SW discharge piping. used as a distractor, candidate must know bases of precaution to eliminate as possible correct answer.

Notes

SYSTEM: 203000 RHR/LPCI: Injection Mode (Plant Specific)
K4. Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following:
(CFR: 41.7)
K4.13 The prevention of leakage to the environment through LPCI/RHR heat exchanger: Plant-Specific 3.4 3.7

This question matches the k/a in that it measures the RO's knowledge of design basis of how leakage to the environment is prevented via LPCI/RHR heat exchanger.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-017*011
Ref Req'd Y or N:	NO	Technical Ref.:	SD-17
? Cognitive Level:	M OR FK	? Source:	NEW

The pumps were originally sized as 33 1/3% capacity pumps until the LPCI FIX MOD where the E11-F010 Valve was locked closed, now the pumps are 100% capacity per the SAFER-GESTR-LOCA Report.

The power supply for the pumps, along with the associated diesel generator and power supply division, is listed below (also see Figure 17-2B):

RHR Pump	1A/2A	1B/2B	1C/2C	1D/2D
Power Source	E3	E4	E1	E2
Diesel	#3	#4	#1	#2
Division	I	II	I	II

Pump start sequencing is provided by the load sequencing logic that is automatically provided during auto initiation conditions. No manual operation is required for emergency pump starts.

NOTE: The pumps are qualified for only short periods of time without the minimum flow valve open. Long term operation without heat rejection can damage pumps.

2.2 RHR Heat Exchangers

Two heat exchangers, one for Loop "A" and the other for Loop "B" are located in separate areas of the Reactor Building above the associated pumps at the 9' to 30' elevation. The heat exchangers are designed to remove the heat from the Suppression Pool water, the Reactor water, or, the Fuel Pool Cooling System water. Process water enters the carbon steel shell side and makes one pass through the heat exchanger, while RHR Service Water passes through copper-nickel "U" tubes to cool the water entering the shell side.

The heat exchangers' shell and tube sides are designed to withstand 450 psig, the maximum pressure which could occur in the Shutdown Cooling Mode. The original peak design temperature of the heat exchangers was based on the Steam Condensing Mode, which is no longer functional. The design operating temperature range is 32°F to 400°F and the design flow rating is 11,550 gpm for the shell side and 8000 gpm for the RHR Service Water (tube) side.

The pressure of the RHR Service Water (tube side), regardless of cooling water flowrate, is maintained a minimum of 15 psig above the shell side (Reactor water side) pressure. This design ensures internal leakage will be from the tube side into the shell side, thus preventing the discharge of Reactor water to the environment.

2.3 RHR Service Water Booster Pumps

Four, single stage, horizontally mounted centrifugal RHR Service Water (RHRSW) Booster Pumps, located on the 50' elevation of the Reactor Building, have a capacity of 4000 gpm each. These pumps raise the pressure of the Service Water to the RHR heat exchangers to supply sufficient cooling water flow and to maintain service water pressure approximately 15 psi higher than primary system pressure. Service Water pressure is maintained slightly higher than that of the Reactor water to prevent contamination of the Service Water, and thus the environment, in the event of a heat exchanger leak. Pump motor cooling is provided from a tap off the suction of the booster pump.

These pumps are not required for LPCI operation; therefore, they receive a trip signal on a LOCA initiation. However, they may be started with a LOCA signal present by utilizing an associated RHRSW LOCA override switch (discussed later).

Each pump is powered from its own unit's 4160 VAC emergency bus, as follows:

Component	Power Supply
2(1)A RHRSW Booster Pump	4160 Bus E3
2(1)B RHRSW Booster Pump	4160 Bus E4
2(1)C RHRSW Booster Pump	4160 Bus E1
2(1)D RHRSW Booster Pump	4160 Bus E2

Further details of the RHR Service Water Booster Pumps and Flowpaths may be obtained from the Service Water System Description, SD-43 and Figure 17-6.

6. 204000A109 001

Unit One (1) is operating at rated power placing the second Reactor Water Cleanup (RWCU) System filter demineralizer (F/D "B") in service. While the AO is increasing flow through the F/D, the RO observes a sudden marked increase in F/D "B" effluent conductivity on RWCU Conductivity Recorder, G31-CRS-R601.

Which ONE of the following identifies the correct course of action to take in response to this observed increase in F/D "B" effluent conductivity?

The RO should:

A. isolate the RWCU System.

B.✓ direct the AO to shutdown F/D "B".

C. direct the AO to continue placing F/D "B" in service.

D. close the Return to Vessel Valve, G31-F042, and establish a reject path to Radwaste.

Feedback

Reference - 1OP-14 Rev. 68 Page 54

Distractor Analysis
A - INCORRECT - RWCU system isolation is not appropriate as F/D "A" is OK and available to help cleanup vessel if resin intrusion in fact occurs.
B - CORRECT - Procedure directs immediate shutdown of F/D to prevent resin intrusion.
C- INCORRECT - resin break through causes conductivity indication to increase, this is not normal and measures to stop vessel intrusion are required.
D - INCORRECT - Rinsing in condensate deep bed demineralizers is a common practice but not allowed for RWCU resin as they are powdex resin not beads and rinse in will not help.

Notes

SYSTEM: 204000 Reactor Water Cleanup System
A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER CLEANUP SYSTEM controls including:
(CFR: 41.5 / 45.5)
A1.09 Reactor water conductivity 3.0 3.2

This question matches the k/a in that it measures the RO's ability to monitor reactor water conductivity while placing a F/D in service. The RO must prove that he understands the implications of a conductivity excursion during this operation by selecting the appropriate action required.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.0	Facility Objective:	CLS-LP-014*10A
Ref Req'd Y or N:	NO	Technical Ref.:	OP-14
? Cognitive Level:	C/A	? Source:	NEW

5.5 Placing the Second F/D On Line

C
Continuous
Use

5.5.1 Initial Conditions

1. All applicable Precautions and Limitations as listed in Section 3.0 have been reviewed. ☐

2. **IF** any loss of power, or large voltage perturbation has occurred to RWCU filter logic, **THEN** the controller has been reprogrammed, **AND** dummy backwashes have been performed. ☐

3. F/D to be placed in service has been precoated in accordance with Section 5.3 or 8.5, and is in *HOLD*. ☐

4. RWCU is in service in accordance with Section 5.4 **AND** is capable of providing at least 140 gpm total F/D flow *(G31-FI-R605 A/B)* to support two-filter operation without pump runout. ☐

> **NOTE:** **IF** RWCU is **NOT** capable of providing a total F/D flow of at least 140 gpm because of restricted reject operation or other factors, **THEN** the on-line F/D must be taken off line in accordance with Section 7.2 and the other F/D brought on line in accordance with Section 5.4.

5.5.2 Procedural Steps

CAUTION

IF Process Computer Point B074 or B075, or NUMAC LDM *B21-XY-5949B* is **NOT** available, the PPC heat balance may be up to 3 CMWT lower than actual reactor power because computer points only get input from RWCU filter demin flow. Raising reactor power to rated will exceed 2923 CMWT if either of the following exist:

- RWCU is in service with *G31-F044* open.
- RWCU filter demin(s) is in service with RWCU reject flow in progress.

1. **ENSURE** off-line F/D flow controller is in *MAN*. ☐
2. **ENSURE** manual thumbwheel on off-line flow controller is set at zero. ☐
3. **OPEN** off-line *F/D A(B) EFFLUENT STRAINER ISOLATION VALVE, G31-Z002-AO-41A(B).* ☐
4. **SLOWLY OPEN** off-line *F/D MANUAL PRESSURIZATION VALVE, G31-V89(V87)*, to obtain a filter pressure within approximately 200 psig of reactor pressure. ☐
5. **OPEN** F/D A(B) *EFFLUENT ISOLATION VALVE, G31-Z002-AO-31A(B)* **AND** *INFLUENT ISOLATION VALVE, G31-Z002-AO-32A(B)*, by placing *INFLUENT & EFFLUENT ISOLATE/OPEN* switch, on Panel G31-Z002-26, in *OPEN*. ☐

5.5.2 Procedural Steps

> **CAUTION**
>
> **WHEN** placing a filter in operation, it is possible to initiate an RWCU System isolation from hi-delta flow instrumentation. Flow perturbations should be closely monitored.

R15

6. **CLOSE** off-line *F/D MANUAL PRESSURIZATION VALVE, G31-V89(V87).* ☐

> **CAUTION**
>
> Do **NOT** press *FILTER START* push button on an unpressurized filter.

7. **DEPRESS** off-line *FILTER START* push button. ☐

8. **ENSURE** the following actions occur:

   a. F/D A(B) *FILTER* light is illuminated. ☐

   b. *FILTER INFLUENT VALVE, G31-Z002-AO-6A(B)*, opens. ☐

9. **IF** necessary, **THEN THROTTLE** *RETURN TO VESSEL VLV, G31-F042*, and *RWCU REJECT FLOW CONTROL VLV, G31-F033*, to limit pump flow. ☐

> **NOTE:** Communications should be established between the local panel and the Control Room prior to performing the next step.

10. **SLOWLY OPEN** *F/D A(B) EFFLUENT FLOW CONTROL VALVE, G31-Z002-66A(B)*, using the manual thumbwheel on *F/D A(B) FLOW CONTROLLER, G31-Z002-FC-74A(B)*, until a filter flow rate of 70-107 gpm, as indicated on *G31-FI-R605A(B)*, is achieved. ☐

5.5.2 **Procedural Steps**

> **NOTE:** **WHEN** operating two F/Ds simultaneously, either or both F/D flow controllers may be operated in *AUTO*.

11. **IF** it is desired to place the second F/D flow controller in *AUTO*, **THEN PERFORM** the following:

    a. **ADJUST** automatic set knob on *F/D A(B) FLOW CONTROLLER, G31-Z002-FC-74A(B)*, so that the setpoint (black indicator) matches the process point (red indicator). ☐

    b. **TRANSFER** *G31-Z002-FC-74A(B)* to automatic by placing the black lever in *AUTO*. ☐

12. **OBSERVE** vessel conductivity to ensure conductivity spikes are **NOT** occurring as a result of bringing the filter on line. ☐

R9, 11

13. **IF** a marked increase in conductivity is observed on *RWCU CONDUCTIVITY, G31-CRS-R601*, as a result of bringing F/D A(B) on line, **THEN PERFORM** the following:

    a. **IMMEDIATELY SHUT DOWN** F/D A(B) in accordance with Section 7.1. ☐

    b. **WHEN** F/D A(B) is shut down, **THEN CONTINUE** in this procedure. ☐

    c. **NOTIFY** E&RC chemistry to sample the reactor coolant for impact to 0AI-81 limits. ☐

> **NOTE:** Reject flow should be minimized to 90 to 105 gpm, as indicated on *G31-FI-R602*, during reactor power operation to prevent damage to the heat exchangers.
>
> **NOTE:** *ORIFICE BYPASS VLV, G31-F031*, may be opened at low pressure in order to establish the desired reject flow.

5.5.2 Procedural Steps

CAUTION

During reactor reject operations, the nonregenerative heat exchanger outlet temperature should be observed. **IF** temperature reaches 130°F, limit reject by returning some water through the regenerative heat exchangers for cooling.

CAUTION

RWCU System pressure should be equalized with reactor pressure and F/D operation should be stable prior to opening *RETURN TO VESSEL VLV, G31-F042.*

CAUTION

WHEN reactor temperature is greater than 212°F, the regenerative heat exchangers should be cooled down gradually after a major portion of the RWCU flow has been rejected. The following guidelines should be followed when establishing flow back to the reactor vessel:

	G31-FI-5954	*G31-FI-R602*	
	Return Flow	Reject Flow	Flow Time
One Filter Operation	35	100	15 min.
	65	65	15 min.
	100	35	15 min.
	130	0	
Two Filter Operation	65	195*	15 min.
	130	130	15 min.
	195	65	15 min.
	260	0	

**G31-FI-R602* range is 0-150 gpm. Reject flowrate above 150 gpm can be calculated as the difference between pump flow (*G31-FI-R609*) and return flow (*G31-FI-5954*).

5.5.2 Procedural Steps

14. **IF** shutdown of the applicable F/D is **NOT** required by Step 5.5.2.13, **THEN PERFORM** the following to partially reject system flow, if desired, while maintaining the proper pump maximum flow limits and regenerative heat exchanger cooldown guidelines:

    a. **IF** necessary, **THEN ADJUST** each *F/D A(B) FLOW CONTROLLER, G31-Z002-FC-74A(B)*, to obtain individual flow rates of 70-107 gpm, as indicated on *G31-FI-R605A(B)*, with total reject flow **NOT** exceeding 90-105 gpm as indicated on *G31-FI-R602.* ☐

    b. **THROTTLE** *RETURN TO VESSEL VLV, G31-F042*, and *RWCU REJECT FLOW CONTROL VLV, G31-F033*, as necessary. ☐

    c. **IF** necessary, **THEN THROTTLE OPEN** *ORIFICE BYPASS VLV, G31-F031.* ☐

15. **IF** shutdown of the applicable F/D is **NOT** required by Step 5.5.2.13, **THEN PERFORM** the following to return total flow to the reactor, while maintaining the proper regenerative heat exchanger cooldown guidelines:

    a. **IF** necessary, **THEN ADJUST** each *F/D A(B) FLOW CONTROLLER, G31-Z002-FC-74A(B)*, to obtain individual flow rates of 70-107 gpm, as indicated on *G31-FI-R605A(B).* ☐

    b. **ENSURE** *RETURN TO VESSEL VLV, G31-F042*, is open. ☐

    c. **ENSURE** *RWCU REJECT FLOW CONTROL VLV, G31-F033*, is closed. ☐

    d. **ENSURE** *ORIFICE BYPASS VLV, G31-F031*, is closed. ☐

    e. **ENSURE** RWCU *REJECT TO CNDSR VLV, G31-F034*, is closed. ☐

    f. **ENSURE** RWCU *REJECT TO RADWASTE VLV, G31-F035*, is closed. ☐

7. 205000K202 001

Which ONE of the following describes the proper power supply alignment for 1-E11-F009, RHR Shutdown Cooling Inboard Suction Throttle Valve motor breakers with Unit One (1) at rated power?

The Normal Feed is from:

A.✓ 1XA and the ASSD Feed is from 1XD. Both are normally OFF.

B. 1XD and the ASSD Feed is from 1XA. Both are normally OFF.

C. 1XA and it is normally ON while the ASSD Feed from 1XD is normally OFF.

D. 1XD and it is normally ON while the ASSD Feed from 1XA is normally OFF.

Feedback

Reference 1OP-17 Electrical Lineup
On Unit One the Inboard valve E11-F009 is the suction throttle vlv. It is 480Vac powered and normally Div I 1XA with ASSD feed available from 1XD. This valve's breakers are designated as prefire rackout breakers in 1OP-17 and must both be left OFF unless SDC is in service. Distractor B is not correct as it has Norm/ASSD power suppllies reversed. C and D do not reflect the prefire rackout staus for the valve's normal power supply and therefore are incorrect.

Notes

SYSTEM: 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)
K2. Knowledge of electrical power supplies to the following:
(CFR: 41.7)
K2.02 Motor operated valves 2.5* 2.7*

This question matches the k/a and because it measures the RO's knowledge of MOV power supplies for SDC mode of RHR including normal line-up of the breakers.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.5	Facility Objective:	LOI-CLS-LP-017*17B
Ref Req'd Y or N:	NO	Technical Ref.:	SD-17 & 1-OP-17
? Cognitive Level:	M OR FK	? Source:	NEW

Residual Heat Removal System Common Electrical Lineup

Number	Description	Position/ Indication	Checked	Verified
Common - Control Bldg. - Emergency 120V AC Dist. Pnl. 1C - El. 23' Cable Spread Area				
16	Control Power For Excess Flow Check Valves B32-F039A, B32-F058A, B32-F006A, B32-F005A, And For Solenoid Valve CAC-SV-1215E	ON		
Common - Control Bldg. - Emergency 120V AC Dist. Pnl. 1D - El. 23' Cable Spread Area				
17	Control Power For Excess Flow Check Valves B32-F039B, B32-F058B, B32-F006B, B32-F005B, And B32-F039D (RIP-CS-1232)	ON		
Common - Reactor Bldg. - 250V MCC 1XDB - El. 20' - South				
B50 Row - A4	RHR Shutdown Cooling Outboard Isolation Valve, 1-E11-F008 Normal Feed	ON*		
B50 Row - A4	Valve E11-F008 Normal-Local Control Switch	NORMAL		
B50 Row - A4	Valve 1-E11-F008 Selector Switch: Open/Off/Close	OFF		
B0C Row - A3	Valve E11-F009 Interlock with RHR Pumps B & D Normal-Isolate Switch	NORMAL		
Common - Reactor Bldg. - 480V MCC 1XA (Rear) - El. 20' - North				
DH3 Row - C1	RHR Shutdown Cooling Inbd Suction Throttle Valve 1-E11-F009, Normal Feed	Note 1 OFF**		
DH3 Row - C1	Valve 1-E11-F009 Normal-Local Keylock Switch	NORMAL		

*This component has an alternate power feed on *MCC 1XDA*, Compartment *B26*. This power feed has ABT circuitry between normal and alternate feeds.

Note 1 - Refer to Section 3.0, Precautions and Limitations, regarding pre-fire rackout requirements.

**This component has an alternate power feed on *MCC 1XD*, Compartment *DX5*. This power feed has ABT circuitry between normal and alternate feeds.

Residual Heat Removal System Common Electrical Lineup

Number	Description	Position/ Indication	Checked	Verified
Common - Reactor Bldg. - 480V MCC 1XA (Rear) - El. 20' - North				
DH3 Row - C1	Valve 1-E11-F009 Selector Switch: Open/Off/Close	OFF		
Common - Reactor Bldg. - 480V MCC 1XD (Front) El. 20' - Southeast				
DX5	RHR Shutdown Cooling Inbd Suction Throttle Valve, 1-E11-F009, ASSD Feed	Note 1 OFF*		
DX5	Valve 1-E11-F009 ASSD Feed Keylock Switch: Close/Off/Open	OFF		
Common - Reactor Bldg. - 250 VDC MCC - 1XDA - El. 20'				
B26 Row - I1	RHR Shutdown Cooling Otbd Isolation Valve, 1-E11-F008, Alternate Feed	Note 1 OFF**		
B26 Row - I1	Valve 1-E11-F008, Keylock Switch: Close/Off/Open	OFF		

*This component has a normal power feed on *MCC 1XA*, Compartment *DH3*. This power feed has ABT circuitry between normal and alternate feeds.

Note 1 - Refer to Section 3.0, Precautions and Limitations, regarding pre-fire rackout requirements.

**This component has a normal power feed on *MCC 1XDB*, Compartment *B50*. This power feed has ABT circuitry between normal and alternate feeds.

3.7.2 The following valve motor breakers are to be left in the prefire rackout position:

Valve No./Switch	MCC/Area	Compt/Node	Breaker Position
E11-F006A	1XA	DE9	OFF
E11-F006B	1XB	DL1	OFF
E11-F006C	1XA	DF0	OFF
E11-F006D	1XB	DL2	OFF
E11-F008	1XDA	B26	OFF
E11-F009	1XA	DH3	OFF
E11-F009	1XD	DX5	OFF
E11-F011B	1XB	DL6	OFF
E11-F026B	1XB	DM3	OFF
E11-V32	1XA	DG3	OFF
E11-V33	1XB	DM9	OFF
E11-F049	1XA	DH4	OFF
E11-F073 App R Local Bkr/Discon	Cable Spread	Node L1G	OFF
E11-F103B	1XB	DN4	OFF

3.7.3 Manual operators for the above valves and *E11-F010* that are outside the primary containment, except *E11-F073*, shall have a label on the operator indicating, "Manual operation of this valve could cause an uncontrolled or unrecoverable loss of primary coolant during a fire. Reference 1OP-17, General Precautions and Limitations."

3.7.4 Manual operator for *E11-F073* shall have a label on the operator indicating, "Manual operation of this valve could cause service water injection into the suppression pool or reactor vessel during a fire. Reference 1OP-17, General Precautions and Limitations."

8. 206000A405 001

The Unit Two (2) High Pressure Coolant Injection (HPCI) System is operating to maintain reactor water level between the low level intiation setpoint and the high level trip setpoint.

Which ONE of the following identifies the reactor water level instruments that are available in the control room to monitor reactor water level over the entire range of HPCI operation?

A.✓ N026A and B

B. N027A and B

C. N036 and N037

D. N004A, B, and C

Feedback

Reference SD-01.2 RPV Instrumentation Rev. 3 page 66 and U2 TS Amm. 233 pages 3.3-42&43

DISTRACTOR ANALYSIS
A. CORRECT - HPCI initiates at LL2 - 105" and trips at 206" N026 Range 0" - 210"
B. INCORRECT - N027 Range 150" - 550"
C. INCORRECT - N004 Range 150" - 210"
D. INCORRECT - N036 Range -150" - + 150"

Notes

SYSTEM: 206000 High Pressure Coolant Injection System
A4. Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)
A4.05 Reactor water level: BWR-2,3,4 . . . . 4.4* 4.4*

This question matches the k/a in that it measures the RO's ability to determine the appropriate level instrument used while operating HPCI over it's entire designed range of RPV level.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 4.4	Facility Objective:	CLS-LP-01.2*03D
Ref Req'd Y or N:	NO	Technical Ref.:	SD-01.2 & SD-19
? Cognitive Level:	C/A	? Source:	NEW

FIGURE 01.2-1
Reactor Water Level Instrument Ranges

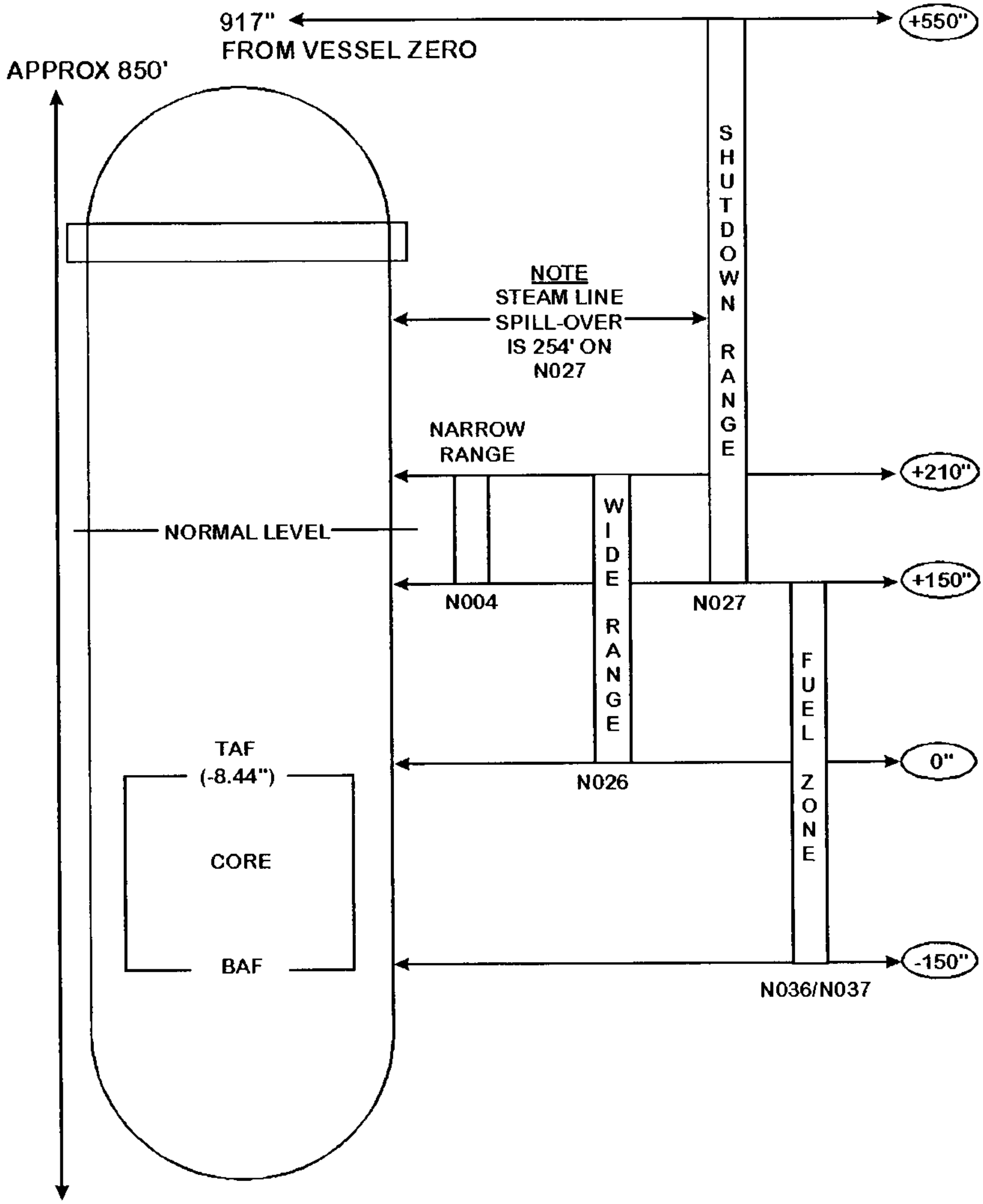


Table 3.3.5.1-1 (page 2 of 4)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
c. Reactor Steam Dome Pressure—Low	1,2,3	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
	4[a], 5[a]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 402 psig and ≤ 425 psig
d. Reactor Steam Dome Pressure—Low (Recirculation Pump Discharge Valve Permissive)	1[b], 2[b], 3[b]	4	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 302 psig
e. Reactor Vessel Shroud Level	1,2,3	2	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ -50 inches
f. RHR Pump Start—Time Delay Relay	1,2,3, 4[a], 5[a]	4 1 per pump	C	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 9 seconds and ≤ 11 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level—Low Level 2	1, 2[c], 3[c]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 101 inches
b. Drywell Pressure—High	1, 2[c], 3[c]	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 1.8 psig

(continued)

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

(b) With associated recirculation pump discharge valve or recirculation pump discharge bypass valve open.

(c) With reactor steam dome pressure > 150 psig

Table 3.3.5.1-1 (page 3 of 4)
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
3. HPCI System (continued)					
c. Reactor Vessel Water Level—High	1, 2(c), 3(c)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≤ 207 inches
d. Condensate Storage Tank Level—Low	1, 2(c), 3(c)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 23 feet 4 inches
e. Suppression Chamber Water Level—High	1, 2(c), 3(c)	2	D	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≤-2 feet
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level—Low Level 3	1, 2(c), 3(c)	2	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 13 inches
b. ADS Timer	1, 2(c), 3(c)	1	F	SR 3.3.5.1.4 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 108 seconds
c. Reactor Vessel Water Level—Low Level 1	1, 2(c), 3(c)	1	E	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 153 inches
d. Core Spray Pump Discharge Pressure—High	1, 2(c), 3(c)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig
e. RHR (LPCI Mode) Pump Discharge Pressure—High	1, 2(c), 3(c)	4 2 per pump	F	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.5	≥ 102 psig and ≤ 130 psig

(continued)

(c) With reactor steam dome pressure > 150 psig

9. 206000K602 001

Unit Two (2) HPCI has automatically initiated on a valid initiation signal. The operator observes the following indications:

Steam Supply Pressure	0 psig
Turbine Exhaust Pressure	0 psig
Pump Discharge Pressure	0 psig
Turbine Speed	1000 RPM, lowering
HPCI Turb Trip Sol Energ	NOT Alarming

Which ONE of the following would explain the above indications?

A. The turbine has tripped on overspeed.

B. Isolation signal due to steam leak detection.

C. The Auxiliary Oil Pump has tripped on magnetics.

D.✓ Loss of 125 VDC to the 24 and 52.5 VDC power supplies.

Feedback

Reference SD-19 rev. page 64
Distractor Analysis
A - incorrect - turbine overspeed would auto reset prior to 1000 rpm
B - incorrect - isolation would energize the turbine trip solenoid
C- incorrect - loss of oil pressure would not cause loss of steam supply pressure
D - Correct - loss of 24/52.5 VDC power supply would de-energize flow controller and indications except for turbine speed.

Notes

SYSTEM: 206000 High Pressure Coolant Injection System
K6. Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM :
(CFR: 41.7 / 45.7)
K6.02 D.C. power: BWR-2,3,4 . . . . 3.3 3.7*

This question matches the k/a because it measures the ROs knowledge of DC power loss on HPCI operation.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.3	Facility Objective:	LOI-CLS-LP-019*03S
Ref Req'd Y or N:	NO	Technical Ref.:	SD-19
? Cognitive Level:	C/A	? Source:	BANK LOI

4.3.11 DC Electrical Distribution

The majority of the HPCI System components, including all of the valves and control circuitry required for automatic initiation, are powered from Division I 125 VDC Electrical Distribution via Distribution Panel 3(4)A and MCC 1(2)XDA. The HPCI relay logic (including initiation, trip and Bus A isolation) is powered from Distribution Panel 3(4)A. The turbine speed control, both the 24 VDC power supply which powers the flow controller and the 52.5 VDC power supply which powers the instrumentation, is powered from Distribution Panel 3(4)A.

The solenoid for AOVs E41-F026, E41-F053 and E41-F054 are powered from Division I 125 Vdc via Distribution Panel 3(4)A.

The solenoid AOV E41-F025 and Isolation Logic Bus B are powered from Division II 125 Vdc via Distribution Panel 3(4)B.

4.3.12 AC Electrical Distribution

Three of the HPCI System MOVs are 480 VAC powered; E41-F002, E41-F075, and E41-F079. E41-F075 is powered from Division I Emergency Distribution via MCC 1(2)XA. E41-F002 and E41-F079 are normally powered from Division II Emergency Distribution via MCCs 1(2)XD and 1(2)XB, respectively. Both of these valves (E41-F002 and E41-F079) have an alternate power source (ASSD) from Division I via 1(2)XC. Loss of power to these valves should not affect automatic initiation of the HPCI System since these valves are normally aligned in the position required for automatic operation.

The Turbine Test Power Supply receives its power from Emergency 120 VAC Distribution Panel 1A(2A).

If a Leak Detection System NUMAC device loses power, it will not automatically isolate HPCI on a high temperature condition. If only one division AC is lost, the other division NUMAC should still isolate HPCI on an actual condition. NUMAC power is from Div. I 120 VAC Pnl. 31A(32A) and Div. II 120 VAC Pnl. 31B(32B).

10. 209001A413 001

Following a valid automatic initiation of Core Spray on Unit Two (2) the RO verifies proper system response without taking any manual operator action. Subsequently, the following alarms are received:

(A-03 6-6) Core Spray Pump 2B Overload
(UA-18 6-1) BUS E4 4KV MOTOR OVLD

Assume that both alarms are sealed in and that no operator action has been taken.

Which ONE of the following describes the expected status of P603 control board lights for Core Spray Pump 2B?

	RED Light	GREEN Light	WHITE Light
A.✓	ON	OFF	OFF
B.	OFF	ON	OFF
C.	ON	OFF	ON
D.	OFF	ON	ON

Feedback

Reference APPs A-03 6-6 and UA-18 6-1
With alarms shown sealed in operator should be able to determine that the pump is still running with an overcurrent condition on the B phase. The APP allows continued operation with an overload condition as necessary under accident conditions. When pump breaker trips open (UA-18 6-1) BUS E4 4KV MOTOR OVLD clears and the pump breaker will indicate OPEN with green light lit. WHITE light is only lit if the pump has been overridden off with control switch while an initiation signal is present.

Distractor Analysis
A - correct - pump motor energized red light lit
B - not correct - pump motor is energized red light lit
C - not correct - WHITE light indicates overridden off
D - not correct - pump motor is energized red light lit, WHITE light indicates overridden off

Notes

SYSTEM: 209001 Low Pressure Core Spray System
A4 Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)
A4.13 Lights and alarms 3.4 3.4

This question matches the k/a in that it measures the RO's ability to understand alarms and to predict Core Spray system status. Also RO must describe the status by understanding indicating lights.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	LOI-CLS-LP-018*8&12
Ref Req'd Y or N:	NO	Technical Ref.:	SD-18 A-03 6-6
? Cognitive Level:	C/A	? Source:	NEW

CORE SPRAY PUMP 2B OVERLOAD

AUTO ACTIONS

NONE

CAUSE

1. Time overcurrent on Phase B.
2. Circuit malfunction.

OBSERVATIONS

1. BUS E4 4KV MOTOR OVLD (UA-18 6-1) alarm.
2. Excessive Core Spray Pump 2B current as read on the local compartment ammeter (Compartment AK5 on 4160V Emergency Bus E4).

ACTIONS

1. If in an accident status, continue running the pump as required.
2. If Core Spray Pump 2B is being tested, trip the pump.
3. Perform a visual inspection of Compartment AK5 on 4160V Emergency Bus E4 to determine the cause.
4. Perform a visual inspection of Core Spray Pump 2B to verify that the pump is free of mechanical trouble.
5. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Overcurrent Alarm Relay 74 OC (actuated from Time Overcurrent Relay 51 on ØB)	Energized

POSSIBLE PLANT EFFECTS

1. Damage to Core Spray Pump 2B.
2. If Core Spray Pump 2B is inoperable a Technical Specification LCO may result.

REFERENCES

1. LL-9364 - 52
2. Technical Specification 3.5.1, 3.5.2
3. APP UA-18 6-1, BUS E4 4KV MOTOR OVLD

BUS E4 4KV MOTOR OVLD

AUTO ACTIONS

NONE

CAUSES

1. Motor overload in one of the following motors:
   a. RHR Pumps 1B and/or 2B (Breakers AL0 and AK3)
   b. RHR SW Pumps 1B and/or 2B (Breakers AK9 and AK4)
   c. Core Spray Pump 2B (Breaker AK5)
   d. CRD Pump 2B (Breaker AK8)
   e. Conventional Service Water Pumps 1A and 2B (Breakers AK6 and AL2)
   f. Nuclear Service Water Pump 2B (Breaker AL1)
2. Circuit, electrical, or mechanical malfunction

OBSERVATIONS

1. Annunciator CORE SPRAY PUMP 2B OVERLOAD (A-03 6-6)
2. Unit 1 Annunciator RHR SW PUMP 1B OVERLOAD (A-03 2-8)
3. Unit 1 Annunciator RHR PUMP 1B OVERLOAD (A-03 4-7)
4. Annunciator RHR SW PUMP 2B OVERLOAD (A-03 2-8)
5. Annunciator RHR PUMP 2B OVERLOAD (A-03 4-7)

ACTIONS

1. Direct Auxiliary Operator to Bus E4 switchgear to confirm which pump has an overload condition.
2. If in other than an emergency condition, shift to alternate equipment, if feasible, per applicable operating procedure.
3. If applicable, refer to appropriate annunciator and operating procedure for further action.

DEVICE/SETPOINTS

51	Reference respective equipment annunciators

POSSIBLE PLANT EFFECTS

1. Loss of essential equipment during an emergency condition.
2. Severe damage to motors of applicable pumps.
3. Loss of safety equipment may result in a Technical Specification LCO.

REFERENCES

1. LL-9359 - 8
2. LL-9047 - 53
3. 1APP A-03
4. 2APP A-03

11. 211000A403 001

During an ATWS on Unit One (1) the operator has initiated SLC by placing the SLC PUMPS A & B control switch in the "Pump A & B Run" position and notes the following SLC indications and parameters:

SLC A Squib Valve Continuity Light	OFF
SLC B Squib Valve Continuity Light	ON
SLC Pump A	ON
SLC Pump B	ON
Reactor Pressure (800 - 1000 psig)	Currently 950 psig
Reactor Water Level (60 - 90 inches)	Currently 90 inches

Which ONE of the following describes the SLC system response and the expected operator response to these indications?

The operator should recognize that squib valve:

A. A failed to fire. The RO should leave the pump control switch in the "Pump A & B RUN" position.

B✓ B failed to fire. The RO should leave the pump control switch in the "Pump A & B RUN" position.

C. A failed to fire. The RO should place the pump control switch in the "SLC Pump A" or "SLC Pump B" position.

D. B failed to fire. The RO should place the pump control switch in the "SLC Pump A" or "SLC Pump B" position.

Feedback

Reference 1OP-05 rev. 39 Sect 5.2 pages 10&11
The SLC mod (EC 50516) was installed on Unit One during HLC 2004. The mod. allows running 2 SLC pumps with one squib valve and the precaution to only run the not associated with a squib valve that failed to fire was removed. Questions similar to this existed in the LOI Systems Bank but now the answer is different due to the mod.
Distractor Analysis
A - incorrect since continuity off for A indicated Squib A fired.
B - correct
C - incorrect since continuity off for A indicated Squib A fired.
D - incorrect, 1OP-05 allows 2 pump ops with one squib provided pressure is <1184 psig.

Notes

SYSTEM: 211000 Standby Liquid Control System
A4. Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)
A4.03 Explosive valves firing circuit status 4.1 4.1

This question matches the k/a because measures the ROs ability to understand squib valve status and the RO must demonstrate the ability to manually operate the SLC system with the squib valve status diagnosed.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 4.1	Facility Objective:	CLS-LP-005*006
Ref Req'd Y or N:	NO	Technical Ref.:	1OP-05
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

5.2 Manual Initiation of SLC

5.2.1 Initial Conditions

1. Standby Liquid Control System in standby in accordance with Section 5.1. ☐

2. Reactor can **NOT** be shut down by normal control rod drive or Scram functions. ☐

3. Manual initiation of SLC has been directed by Unit SRO. ☐

5.2.2 Procedural Steps

CAUTION

WHEN the Standby Liquid Control System is manually initiated for a plant emergency, **THEN** the entire quantity of sodium pentaborate solution should be injected into the reactor before the SLC Pumps are stopped to ensure the reactor is subcritical in all conditions.

1. **UNLOCK AND PLACE** *SLC PUMPS A & B, C41-CS-S1*, in the *PUMP A & B RUN* position. ☐

   a. **OBSERVE** *SQUIB VALVE CONTINUITY LOSS* annunciator on Annunciator Panel A-4 on. ☐

   b. **OBSERVE** *SLC A/B SQUIB VALVE CONTINUITY* indicating lights off at Panel P603. ☐

   c. **OBSERVE** *SLC PUMP A* **AND** *SLC PUMP B* red indicating lights on. ☐

5.2.2 Procedural Steps

NOTE:	The SLC pump discharge relief valve should not actuate with two pumps operating and only one squib valve open unless reactor pressure exceeds 1184 psig, which is possible during a ATWS even with 10 SRVs open.

2. **IF** *SLC A SQUIB VALVE CONTINUITY* **OR** *SLC B SQUIB VALVE CONTINUITY* indicating light on Panel P603 remains on **AND** reactor pressure is greater than or equal to 1184 psig, **THEN PLACE** *SLC PUMP A & B* Control Switch, *C41-CS-S1,* to the *SLC PUMP A* **OR** *SLC PUMP B* position. ☐

   a. **ENSURE** the selected SLC Pump red indicating light on. ☐

3. **ENSURE** *RWCU OUTBOARD ISOL VLV, G31-F004,* closes. ☐

NOTE:	It will take approximately 29 to 40 minutes to completely empty the SLC Storage Tank with two-pump operation and 58 to 81 minutes with one-pump operation.

4. **ENSURE** SLC injection as follows:

   a. *SLC STORAGE TANK LEVEL, C41-LI-R601,* indicates level decreasing. ☐

   b. *SLC PUMP DISCHARGE PRESSURE, C41-PI-R600,* is greater than reactor vessel pressure. ☐

CAUTION

WHEN the Standby Liquid Control System is manually initiated for a plant emergency, failure to stop the SLC Pumps when the liquid supply is exhausted may damage the pumps.

5. **WHEN** all the solution has been injected into the vessel as indicated by reading zero percent on level indicator *C41-LI-R601*, **OR** the Unit SCO has directed the SLC System be shutdown, **THEN PLACE** *SLC PUMP A & B* Control Switch, *C41-CS-S1*, in the *STOP* position **AND LOCK.** ☐

12. 212000G2.2.22 002

Which ONE of the following Average Power Range Monitor (APRM) RPS functions is ONLY applicable with the reactor in MODE 2?

A. INOP

B. OPRM Upscale

C✓ Neutron Flux - High (Setdown)

D. Simulated Thermal Power - High

Feedback

Reference TS 3.3.1.1 Ammendment 243

At BNP we do not typically expect ROs to have TS completely memorized but this question can be figured out by understanding the purpose of each RPS function listed. It is operationally applicable as all ROs are expected to know when LCOs are not met.

Distractor Analysis

A - incorrect - INOP applicable in MODE 1 and 2

B - incorrect - OPRM Upscale applicable >20% RTP

C - correct

D- incorrect - STP - High applicable in MODE 1 ONLY

Notes

SYSTEM: 212000 Reactor Protection System

2.2.22 Knowledge of limiting conditions for operations and safety limits.

(CFR: 43.2 / 45.2)

IMPORTANCE RO 3.4 SRO 4.1

This question matches the k/a because it measures the RO's knowledge of TS for RPS settings.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-003*027
Ref Req'd Y or N:	NO	Technical Ref.:	TS AND SD-03
? Cognitive Level:	M OR FK	? Source:	NEW

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 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]	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 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	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3[c]	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 22.7% RTP
b. Simulated Thermal Power—High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	≤ 0.55W + 62.6% RTP[b] and ≤ 117.1% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) ≤ [0.55 (W - ΔW) + 62.6% RTP] when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.
(c) Each APRM channel provides inputs to both trip systems.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	3[c]	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3[c]	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 20% RTP	3[c]	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18 SR 3.3.1.1.19	NA[d]
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

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

(d) See COLR for OPRM period based detection algorithm (PBDA) setpoint limits.

13. 215001A202 001

Unit Two (2) is operating at rated power and Reactor Engineering has just completed running TIP Traces with TIP Channel D. The TIP detector has been left at the index position to allow for sufficient decay. Switch positions for TIP Channel D are as follow:

MAN VALVE CONTROL switch	CLOSED
MODE switch	OFF

Subsequently, a loss of drywell cooling occurs and drywell pressure rises to 2.3 psig.

Which ONE of the following describes response of the TIP system and what operator action is required to mitigate the consequences of the high drywell pressure condition?

A. The TIP detector automatically retracts to the shield chamber and the ball valve will automatically isolate. The RO only needs to verify that the ball valve is CLOSED.

B. The TIP system will automatically fire the shear valve since the TIP detector was left in the index position. Once the shear valve fires the RO only needs to verify that the shear valve is CLOSED.

C. The TIP detector cannot be retracted with an isolation signal present and the ball valve will not CLOSE with the detector in the index position. The RO must manually fire the shear valve and verify that the shear valve is CLOSED.

D✓ The containment isolation logic has been defeated. The RO must place the MODE switch in MAN/AUTO and then verify that the TIP detector automatically retracts to the shield chamber and that the ball valve automatically isolates and is CLOSED.

Feedback

Reference 2OP-09.1 rev. 27 page 6 and SD-09.5 page 31 and 32
2OP-09.1 CAUTION
The *MODE* switch should **NOT** be placed in *OFF* while the probe is inserted past the Ball Valve to ensure the isolation logic is **NOT** defeated.
Distractor Analysis
A - incorrect - MODE switch in OFF defeats isolation logic
B - incorrect - shear valves are only manually actuated w/ keylock switch.
C - incorrect - TIP detector can be retracted w/ isolation signal and shear vlv. is not necessary
D- Correct - placing MODE switch in MAN/AUTO will enable isoaltion logic

Notes

System: 215001 Traversing In-Core Probe
A2. Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; 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 High primary containment pressure: Mark-l&II(Not-BWR1) 2.9 3.0

This question matches the k/a in that it measures the RO's knowledge of the TIP response to a high drywell pressure condition when in a certain lineup. The lineup is abnormal condition that is described in a precaution in the operating procedure and the RO must demonstrtae his knowledge of the system in order to formulate a course of action to take in response.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 2.9	Facility Objective:	CLS-LP-09.5*05B
Ref Req'd Y or N:	NO	Technical Ref.:	OP-09.1 SD-09.1
? Cognitive Level:	C/A	? Source:	NEW

R
Reference
Use

5.2 Post TIP Machine Operation

5.2.1 Initial Conditions

1. All detector runs for the applicable TIP Machine are completed. ☐

5.2.2 Procedural Steps

1. **PERFORM** the following for the TIP Machine operated:

   a. **ENSURE** probe is at the index position. ☐

   b. **ENSURE** *MAN VALVE CONTROL* switch is in *CLOSED*. ☐

> **CAUTION**
>
> The *MODE* switch should **NOT** be placed in *OFF* while the probe is inserted past the Ball Valve to ensure the isolation logic is **NOT** defeated.

   c. **ENSURE** *MODE* switch is in *MAN* **OR** *AUTO*. ☐

> **CAUTION**
>
> Placing *MODE* switch in *OFF* when the detector is outside the shield chamber will defeat the containment isolation logic and may cause damage to the detector and cable.

2. **IF** the probe is to be left at the index position **AND** the control station will **NOT** be manned, **THEN COMPLETE** Attachment 4. ☐

4.1.3 Manual Drive

Used to probe guide tube runs to determine core-top and bottom limits during axial alignment and during maintenance to remove or wind cable onto the takeup reels. This is an entirely manual operation for TIP Ball Valve and hand crank operation and will also require detachment of the drive chain from TIP Drive Mechanism gear head to mechanical slip clutch.

Manual drive employs a hand crank at the TIP Drive Mechanisms and can also be used if a detector encounters excessive resistance during a traverse for LPRM calibration. Manual manipulation of the detector for short distances does not require the drive chain to be disconnected; however, enough force is applied to allow the clutch to slip.

4.1.4 TIP Purge Operation

Unit 1 TIP purge uses nitrogen from the Containment Atmosphere Control System (CAC) and Unit 2 TIP purge uses nitrogen from the Pneumatic Nitrogen System (PNS). Nitrogen is normally supplied to the Indexers to maintain a dry, sealed, positive pressure inside the Indexers to help preclude moisture buildup and component oxidation.

4.2 Abnormal Operation

4.2.1 TIP Operation with a Group 2 Isolation Signal Present

Upon receipt of a Group 2 Isolation signal from the PCIS System:

- Low Reactor Water Level
- Hi Drywell Pressure

On this signal, any TIP not in the Shield Chamber is automatically transferred to the manual reverse mode of operation (Result of relay logic in Drive Control Unit). The detector will be retracted from the core at fast speed. When the detector is In-Shield as indicated by the limit switch, the TIP Ball Valve is closed.

4.2.2 TIP Fails To Isolate With an Isolation Signal Present

CAUTION
1. The MODE Switch should **NOT** be placed OFF while the TIP Probe is inserted past the TIP Ball Valve to ensure that the PCIS Isolation Logic is not defeated. 2. The MODE Switch should **NOT** be placed in OFF until the TIP Ball Valve CLOSED position indicating light is ON. 3. IF the TIP Probe becomes stuck beyond the shield, the Unit SCO must be notified that the Primary Containment Isolation Logic is defeated for the associated TIP Ball Valve. 4. The TIP Ball Valve will **NOT** CLOSE and the TIP Probe will **NOT** STOP if the shield proximity switch fails to actuate while retracting the TIP Probe from the Indexer to the in-shield position. Should the proximity switch fail, using the MANUAL mode, the TIP Probe must be placed at the in-shield position and the Unit SCO informed immediately to determine the TIP Ball Valve operability (Primary Containment Isolation Valve TECH SPEC 3.6.1.3).

Several conditions can cause this situation:

- Ball valve will not close
- TIP Detector will not retract (stuck)

Given the need to isolate the guide tube, the Shear Valve is capable of being closed by operating the key lock switch (S-1) at the Valve Control Monitor. The Shear Valve itself is not a PCIS Valve.

Operators need to be aware that a Technical Specification LCO needs to be initiated if either of the following conditions occurs:

a. The TIP Detector is inserted beyond the TIP Ball Valve and the associated TIP Machine power is turned off. The TIP logic is defeated in this condition and a Group 2 isolation signal will not occur on this TIP probe.

b. A TIP Detector becomes stuck beyond the TIP Ball Valve.

14. 215003A105 001

Unit One (1) is commencing a startup with IRM C reading as follows:

IRM C 21/125 on Range 2

Which ONE of the following describes the expected change in IRM C indication associated with taking the Range Switch for IRM C from Range 2 to Range 1?

IRM C indication increases and:

A. remains on scale ONLY. No other warnings or actuations occur.

B.✓ starts to flash indicating that IRM C be should ranged up. No rod out block or half scram occurs.

C. starts to flash indicating that IRM C be should ranged up. A rod out block will occur but no half scram occurs.

D. starts to flash indicating that IRM C be should ranged up. A rod out block and a half scram will occur.

Feedback

Reference SD-09.1
Ranging IRM C to range 1 would result in a reading of 21/40 or ~65/125. Rod block setpoint is 70/125 and scram setpoint is 117/125. Meter indication would definetly increase and recent plant mod. has installed new recorders that flash to warn operators to range up at 50/125.

Distractor Analysis
A - incorrect - meter indication would increase to 65/125 and flash
B - correct
C - incorrect - not high enough for rod block 70/125
D - incorrect - not high enough for a half scram

Notes

SYSTEM: 215003 Intermediate Range Monitor (IRM) System
A1. Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including:
(CFR: 41.5 / 45.5)
A1.05 SCRAM and rod block trip setpoints 3.9 3.9

This question matches the k/a in that it measures the RO's ability to predict the IRM indication resulting from operating the range switch and then test his knowledge of scram and rod block set points.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.9	Facility Objective:	CLS-LP-09.1*3A&C
Ref Req'd Y or N:	NO	Technical Ref.:	SD-09.1
? Cognitive Level:	C/A	? Source:	NEW

- ***Trend message:*** Print one of 36 preset messages to the Trend Screen and the Event/Alarm Log.

- ***Display Previous:*** Display the previous value for all points that can be reset.

- On the IRM paperless recorders when ranging IRMs (maintaining IRM ranging between 15 and 50), if an IRM reaches 50 the pen value will flash indicating that the IRM needs to be ranged (if the bottom block is ON).

- On the IRM recorders, the 0-40 BLOCK VALUES still indicate the 0-125-scale value, so it is important to use the graphical value when on the 0-40 scale.

- If bar graphs are displayed, then the trend capability is not displayed.

- Screen savers on the recorders are set at 30 minutes.

2.6 Trip Units

SRM/IRM trip units provide an alarm and, in some cases, a protective interlock or action (i.e., rod block, scram) when a signal exceeds a preset limit.

Each trip unit has two inputs: the reference input and the signal input. If the input exceeds the reference, an output is generated. Two types of trip units are used: the upscale trip and the downscale trip. An upscale trip is where the unit produces an output when the signal input rises above the reference input. A downscale trip is where the unit produces an output when the signal input drops below the reference input.

Each trip unit produces two types of outputs: seal-in and auto reset. Once a seal-in output is produced, it must be manually reset by operator. An example of a seal-in output is one that drives the local indicator lights on the chassis front. The auto reset output allows the trip signal to reset as soon as the monitored parameter returns within reference limits. This output drives indicators on panel P603 and Reactor Protection or Reactor Manual Control System logic.

TABLE 09.1- 1 (Cont'd)
INSTRUMENT AND CONTROL SETPOINTS
STARTUP RANGE NEUTRON MONITORING SYSTEM

INSTRUMENT DESIGNATION AND TRIP FUNCTION	TRIP SETPOINT	FUNCTION, ADDITIONAL CONDITIONS AND COMMENTS
IRM Inop Trip C51-IRM-K601 (A-H)$^{TS \& TRM}$ C72-K14 (A-H)^{TS} Annunciator "IRM A(B) UPSCALE/INOP" (A-05 3-4 & 4-4)	• 80 ± 10 Vdc • Switch not in OPERATE • IRM module unplugged	Initiates a **rod block** and **half scram** if the following conditions are met: •Reactor MODE SWITCH is not in RUN •Associated IRM is not bypassed
IRM Downscale C51-IRM-K601 (A-H)^{TRM} Annunciator "IRM A(B) DOWNSCALE" (A-05 1-4)	5/125 ± 1.5	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is not in RUN •Associated IRM is not bypassed •Above Range 1
IRM Upscale Alarm C51-IRM-K601 (A-H)^{TRM} Annunciator "IRM A(B) UPSCALE" (A-05 2-4)	70/125 ± 2.5	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is not in RUN •Associated IRM is not bypassed
IRM UpscaleTrip C51-IRM-K601 (A-H$^{TS \& TRM}$ C72-K14 (A-H)^{TS} Annunciator "IRM A(B) UPSCALE/INOP" (A-05 3-4 & 4-4)	117/125 ± 2.5	Initiates a **rod block** and **half scram** if the following conditions are met: •Reactor MODE SWITCH is not in RUN •Associated IRM is not bypassed
Detector Not Full In C51-IRM-K601 (A-H)^{TS}	N/A	Initiates a **rod block** if the following conditions are met: •Reactor MODE SWITCH is not in RUN •IRM detector not full in Note: Interlock does not prevent detector movement

^{TS} Technical Specification related ^{TRM} Technical Requirement Manual related

NOTE: With the shorting links removed, any single SRM Upscale Trip, IRM or APRM Upscale or INOP signal will cause a full scram.

15. 215004A302 001

During a reactor startup on Unit One (1) the RO performs a continuous withdrawal of a control rod prior to reaching three doublings of initial count rate. SRM Period A-05 (3-3) annunciator alarms and the RO observes that the amber period light for SRM C is lit. The RO then observes that SRM Period A-05 (3-3) annunciator is now clear. SRM indications are as follows:

SRM Channel	SRM Counts	SRM Period
A	8.5×10^1	+200 sec.
B	2.5×10^2	+150 sec.
C	2.0×10^2	+90 sec.
D	1.0×10^2	+120 sec.

Which ONE of the following describes the current expected condition of the SRM C amber period lights?

The SRM C amber period light at P603 will be:

A. ON and in the back panels the SRM C amber period light will be ON.

B. ON and in the back panels the SRM C amber period light will be OFF.

C✓ OFF and in the back panels the SRM C amber period light will be ON.

D. OFF and in the back panels the SRM C amber period light will be OFF.

Feedback

Reference APP A-05 (3-3) and SD-09.1- The amber period lights are lit from the SRM Period alarm output channel. The P603 lights automatically clear with the alarm but the back panel lights must be reset.
Distractor analysis
A - incorrect - P603 amber light will automatically reset
B - incorrect - P603 amber light will automatically reset and back panel light will not.
C - correct
D - incorrect - back panel light will not automatically reset

Notes

SYSTEM: 215004 Source Range Monitor (SRM) System
A3. Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including:
(CFR: 41.7 / 45.7)
A3.02 Annunciator and alarm signals 3.4 3.3

This question matches the k/a in that it measures the RO's knowledge of SRM alarm signals in response to an automatic occurrence of a intermittent short period experienced while moving a control rod.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-09.1*017
Ref Req'd Y or N:	NO	Technical Ref.:	SD-09.1 APP-A-05 3-3
? Cognitive Level:	C/A	? Source:	NEW

- ***Trend message:*** Print one of 36 preset messages to the Trend Screen and the Event/Alarm Log.

- ***Display Previous:*** Display the previous value for all points that can be reset.

- On the IRM paperless recorders when ranging IRMs (maintaining IRM ranging between 15 and 50), if an IRM reaches 50 the pen value will flash indicating that the IRM needs to be ranged (if the bottom block is ON).

- On the IRM recorders, the 0-40 BLOCK VALUES still indicate the 0-125-scale value, so it is important to use the graphical value when on the 0-40 scale.

- If bar graphs are displayed, then the trend capability is not displayed.

- Screen savers on the recorders are set at 30 minutes.

2.6 Trip Units

SRM/IRM trip units provide an alarm and, in some cases, a protective interlock or action (i.e., rod block, scram) when a signal exceeds a preset limit.

Each trip unit has two inputs: the reference input and the signal input. If the input exceeds the reference, an output is generated. Two types of trip units are used: the upscale trip and the downscale trip. An upscale trip is where the unit produces an output when the signal input rises above the reference input. A downscale trip is where the unit produces an output when the signal input drops below the reference input.

Each trip unit produces two types of outputs: seal-in and auto reset. Once a seal-in output is produced, it must be manually reset by operator. An example of a seal-in output is one that drives the local indicator lights on the chassis front. The auto reset output allows the trip signal to reset as soon as the monitored parameter returns within reference limits. This output drives indicators on panel P603 and Reactor Protection or Reactor Manual Control System logic.

16. 215005K202 001

Unit One (1) is at 100% power. RPS MG-Set "A" trips.

Which ONE of the following identifies how the APRM NUMACs at P608 and APRM Operator Display Assemblies (ODAs) at P603 are affected?

A. All APRM NUMACs and APRM ODAs remain energized from RPS "B".

B.✓ All APRM NUMACs remain energized from RPS "B" and all APRM ODAs remain energized from UPS.

C. APRM NUMACs 1 and 3 are deenergized and APRM ODAs remain energized from RPS "B".

D. APRM NUMACs 1 and 3 are deenergized and APRM ODAs remain energized from UPS.

Feedback

Reference SD-09.6 rev.0 page 44 and 45 - APRM NUMACS are redundantly powered from RPS A and B and ODAs are powered from UPS V7A.

Distractor Analysis

A - incorrect ODAs P/S UPS

B- correct

C - incorrect - RPS B will provide redundant power to NUMACs 1 and 3. ODAs P/S UPS

D- incorrect - APRM 1 and 3 redundant P/S RPS B

Notes

SYSTEM: 215005 Average Power Range Monitor/Local Power Range Monitor System

K2. Knowledge of electrical power supplies to the following:

(CFR: 41.7)

K2 02 APRM channels 2.6 2.8

This question matches the k/a in that it measures the ROs knowledge of power supplies for APRM NUMACS and ODAs.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.6	Facility Objective:	CLS-LP-09.6*07A
Ref Req'd Y or N:	NO	Technical Ref.:	SD-09.6
? Cognitive Level:	M OR FK	? Source:	NEW

each lose power causing their outputs to change state and all their indications to be off. A half scram has already occurred because of the actual RPS bus power loss.

If a loss of RPS bus "B" occurs, there is a loss of one of two redundant power supplies to each APRM and each RBM channel. APRM channels and RBM channel A remain operable. RBM channel B is rendered inoperable because of a loss of power to the RBM B Interface Panel. This inhibits the ability of RBM channel B to receive signals from the Reactor Manual Control System (RMCS) causing a critical fault and INOP trip. VOTER 2 and VOTER 4 each lose power causing their outputs to change state and all their indications to be off. A half scram has already occurred because of the actual RPS bus power loss.

A loss of 120VAC Bus V7A (Circuit #2) results in a loss of power to the IRM recorders, APRM/RBM recorders, and the APRM and RBM Operator Display Assemblies at Panel P603.

A loss of Instrument Bus 1AB (Circuit #9) results in a loss of power to the P603 apron section indicating lights and the recirculation flow recorder.

4.3 Interrelationships With Other Systems

4.3.1 Reactor Protection System

APRM channels provide signals to open contacts in the scram trip logic of the RPS System under various conditions discussed previously.

The RPS System provides power to each of the four APRM instruments, which in turn provide power to all subsystems driven from the APRM instruments or NUMAC. Both RPS busses, A and B, provide power to each APRM instrument, as well as, each RBM. Therefore, a loss of one RPS bus will not affect operation of the PRNMS.

The reactor mode switch provides input to each APRM instrument to determine when to enforce the fixed or flow biased scram trip and rod block settings. OPRM circuitry is enabled only when power/flow conditions are met and the mode switch in RUN.

Loss of any APRM channel will result in a trip signal being sent to the respective 2/4 voter logic which in turn is sent to the RPS Trip

Subsystem. Loss of the RPS System will affect the APRM System dependent upon the type of failure.

4.3.2 UPS

The UPS Distribution System provides power to the IRM recorders, the combined APRM/RBM paper-less recorders, the APRM Operator Display Assemblies, and the RBM Operator Display Assemblies at Panel P603 through Panel V7A (Circuit #2). Loss of power from Panel V7A (Circuit #2) will require monitoring the APRM channels from the APRM NUMAC at P608.

4.3.3 120 VAC Distribution System

The 120 VAC distribution system supplies power to the APRM and RBM panel indicating lights and the recirculation flow recorder at Panel P603 through Panel 1AB (Circuit #9). Loss of the power from Panel 1AB (Circuit #9) causes the APRM and RBM indicating lights inoperable. Recirculation flow can still be determined using the Operator Display Assemblies and flow indicators at Panel P603 and the APRM NUMAC at P608.

4.3.4 125 VDC Distribution System

The 125 VDC Distribution System supplies power to the LPRM annunciators. Failure of the LPRM's has no effect on 125VDC Distribution system, however the loss of 125 VDC sources [3(4) A/B] to P603 annunciators will render the LPRM annunciators inoperable.

4.3.5 Reactor Manual Control System

APRM channels provide signals to open contacts in the RMCS rod withdraw block circuitry to inhibit outward rod motion under the following conditions:

- STP Upscale Alarm
- Neutron Flux Downscale
- APRM Inop Trip
- Operating LPRM inputs at any level < 3, or LPRM total < 17
- OPRM operating cells at any level <2, or cell total <18
- Recirc Flow Upscale/ Off-Normal

4.3.6 Traversing In-Core Probe (TIP) System

The TIP System provides for periodic calibration of the LPRM detectors during power operation. Loss of the TIP System will have

17. 215005K504 001

Which ONE of the following describes the operational implications of LPRM detector assignments and core symmetry as they apply to the Power Range Neutron Monitoring (APRM/LPRM) System?

An APRM must have at least 17 of 31 assigned LPRMs OPERABLE with at least:

A. 2 LPRM inputs from each of the 4 axial levels OPERABLE; this ensures that the APRM will provide an accurate representation of core wide average power.

B✓ 3 LPRM inputs from each of the 4 axial levels OPERABLE; this ensures that the APRM will provide an accurate representation of core wide average power.

C. 2 LPRM inputs from each of the 4 axial levels OPERABLE; this ensures that the APRM will provide a valid simulated thermal power input to the RBM.

D. 3 LPRM inputs from each of the 4 axial levels OPERABLE; this ensures that the APRM will provide a valid simulated thermal power input to the RBM.

Feedback

Reference APP A-06 (3-7) rev. 38 page 40
Less than 17 operating LPRMs in flux average or less than 3 per reactor level actuates APRM TROUBLE to signify that the APRM no longer has sufficient symmetric inputs to accurately determine core average power. APRM must be declared INOP per TS.
TS Bases 3.3.1.1 Rev no. 30 pg. B 3.3.1.1-7 - ... to provide adequate coverage of the entire core

Distractor Analysis
A- incorrect - 3 per level required, 2 per level was required before PRNMS mod.
B - correct
C - incorrect - per level required, 2 per level was required before PRNMS mod., Too few inputs generates a Rod Block due to innaccuracies in core wide average power. Bases is too few has nothing to do with RBM. An INOP APRM can be bypassed and RBM will still have a valid APRM reference from an alternat APRM.
D - incorrect - Too few inputs generates a Rod Block due to innaccuracies in core wide average power. Bases is too few has nothing to do with RBM. An INOP APRM can be bypassed and RBM will still have a valid APRM reference from an alternat APRM.

Notes

SYSTEM: 215005 Average Power Range Monitor/Local Power Range Monitor System
K5. Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM :
(CFR: 41.5 / 45.3)
K5.04 LPRM detector location and core symmetry 2.9 3.2

This question matches the k/a in that it measures the RO's knowledge operational implications regarding LPRM axial location and symetry requirements required to ensure an accurate representation of core wide average power.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.9	Facility Objective:	CLS-LP-09.6*017
Ref Req'd Y or N:	NO	Technical Ref.:	SD-09.6
? Cognitive Level:	M OR FK	? Source:	NEW

APRM TROUBLE

AUTO ACTIONS

1. Rod Withdrawal Block if alarm initiated by too few LPRM detectors per level or too few LPRM detectors in flux average.

CAUSE

1. The quantity of operating LPRM detectors at any given reactor level is less than three.
2. The quantity of operating LPRM detectors in the flux average is less than 17.
3. Any self-test fault.

OBSERVATIONS

1. ROD OUT BLOCK (A-05 2-2) alarm.
2. The Rod Withdrawal Permissive indicating light will be off.
3. On APRM BARGRAPH display at P608 and PPC Displays 882-885, LPRMs in average is less than 17, if this condition caused the alarm.

ACTIONS

> **NOTE:** If cause of the alarm is due to too few LPRMs in the average or too few LPRMs per axial level, the APRM is inoperable in accordance with Tech Spec Basis 3.3.1.1. However, no trip is automatically sent to the Voters.

1. If necessary to determine which APRM initiated the alarm, perform the following at each APRM ODA:
   a. Press ETC soft key to obtain TRIP STATUS soft key.
   b. Press TRIP STATUS soft key.
   c. Observe an asterisk in inverse video in the Trouble Alarm column, indicating this APRM initiated the alarm.
   d. Press INOP STATUS soft key to determine cause of the alarm.
2. Refer to Tech Spec Section 3.3.1.1 for required actions.
3. If the APRM cannot be returned to operable status, then if possible, place the affected APRM in Bypass.
4. When plant conditions allow, return LPRMs to service and remove the affected APRM from Bypass.
5. If self-test fault initiated the alarm, then contact I&C.

ACTIONS

APRM Channels 1 through 4	Less than 17 LPRM detector inputs to flux average
	Less than 3 LPRM detectors per axial level.
	Self-test fault.

POSSIBLE PLANT EFFECTS

1. APRM inoperable.
2. If an APRM channel is inoperable or bypassed, a Tech Spec LCO or TRM Compensatory Measure may result.

REFERENCES

1. LL-09364 - 94
2. 2-FP-05851
3. Tech Spec 3.3.1.1, B3.3.1.1, TRMS 3.3
4. APP A-05 2-2, ROD OUT BLOCK

18. 217000K304 001

Unit One (1) has been operating at rated power for the last 100 days and the High Pressure Coolant Injection (HPCI) System is INOP. A sudden loss of condenser vacuum causes a Group 1 isolation and all rods insert on the reactor scram.

Reactor Core Isolation Cooling (RCIC) system has initiated and currently reactor water level is 120" and rising slowly. Subsequently, the RCIC TURB OIL PRESS LO A-03 (5-4) annunciator alarms.

Which ONE of the following describes the effect that the low RCIC turbine oil pressure condition will have on adequate core cooling under these conditions?

A. RCIC will trip immediately and adequate core cooling requires initiation of Low Pressure ECCS Systems.

B. RCIC will trip after a time delay and adequate core cooling requires initiation of Low Pressure ECCS Systems.

C✓ RCIC will continue to run. Operation of RCIC will be sufficient to provide adequate core cooling without initiation of Low Pressure ECCS systems and operation should be continued to maintain reactor water level.

D. RCIC will continue to run. However, RCIC does not have adequate capacity to provide adequate core cooling without initiation of Low Pressure ECCS systems and should be tripped to preclude bearing damage.

Feedback

Reference TS LCO BASES Rev. 30 B 3.5.3-2 Rev. 30 and APP A-03 (5-4)
RCIC is designed to provide adequate makeup for ACC in an isolation event. RCIC does not trip on low turb oil press but the APP states that if RCIC is not required to maintain reactor water level to trip it to preclude bearing damage. RCIC is required in this case to preclude plant blowdown and actuation of LP ECCS.

Distractor Analysis
A & B- incorrect - RCIC does not trip on low oil pressure.
C - correct
D - RCIC is designed to provide ACC in an isolation event

Notes

SYSTEM: 217000 Reactor Core Isolation Cooling System (RCIC)
K3. Knowledge of the effect that a loss or malfunction of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) will have on following:
(CFR: 41.7 / 45.4)
K3.04 Adequate core cooling 3.6 3.6

This question matches the k/a in that it measures the RO's knowledge of the effect that a malfunction has on RCIC and the importance of RCIC on ACC under specific case of loss of high pressure feed scenario.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.6	Facility Objective:	CLS-LP-016*001&12B
Ref Req'd Y or N:	NO	Technical Ref.:	TS AND APP-03 (5-4)
? Cognitive Level:	C/A	? Source:	NEW

RCIC TURB OIL PRESS LO

AUTO ACTIONS

NONE

CAUSE

1. Defective oil pump.
2. Abnormal oil tank level.
3. Clogged oil filter.
4. Leak in the oil system.
5. Circuit malfunction.

OBSERVATIONS

1. RCIC turbine oil pressure less than 4 psig (local).
2. Oil filter differential pressure greater than 6.4 psid (local).
3. RCIC OIL FILTER ΔP HI (A-02 2-5) alarm.

ACTIONS

1. If in an accident status, utilize the HPCI System per OP-19 to maintain reactor vessel level.
2. If the RCIC System is in operation and is not required to maintain reactor vessel level, trip the RCIC turbine before the bearings are damaged.
3. If the oil filter differential pressure is high, refer to APP A-02 2-5, RCIC OIL FILTER ΔP HI.
4. If a circuit malfunction is suspected, oil leaks exist, or the filter is dirty, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Turbine Oil Pressure Switch E51-PS-1	2-4 psig
Time Delay Relay E51-K9 (E51-F045 not full closed)	15 seconds

POSSIBLE PLANT EFFECTS

1. If the RCIC System is inoperable, a Technical Specification LCO may result.

REFERENCES

1. LL-93064 - 49
2. Technical Specification 3.5.3
3. APP A-02 2-5, RCIC OIL FILTER ΔP HI
4. OP-19, HPCI System

BASES

BACKGROUND (continued)	The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge piping is maintained full of water using a "keep fill" system.
APPLICABLE SAFETY ANALYSES	The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, however, the system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 3) and is therefore included in the Technical Specifications.
LCO	The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the low pressure ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity for maintaining RPV inventory during an isolation event.
APPLICABILITY	The RCIC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV.
ACTIONS	<u>A.1 and A.2</u> If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI System is verified immediately to be OPERABLE, the RCIC System must be restored to

(continued)

19. 218000K401 001

Unit Two (2) is mitigating a Loss of Coolant Accident (LOCA) per EOP-01-RVCP Reactor Vessel Control Procedure. Conditions are as follows:

Reactor Water Level	40" and lowering slowly
Reactor Pressure	850 psig lowering slowly
AUTO DEPRESS RELAYS ENERGIZED - A-03 (3-2)	In Alarm
AUTO DEPRESS TIMERS INITIATED - A-03 (5-1)	In Alarm
ADS valve position indication	All Green ON, All Red OFF

The SRO has determined that reactor water level can be restored and maintained above the Top of Active Fuel (TAF) with available injection sources and directs the RO to place the ADS inhibit switches to "INHIBIT".

Which ONE of the following describes the effect that placing the ADS inhibit switches to "INHIBIT" has on the ADS?

Placing the ADS inhibit switches to "INHIBIT" will deenergize ADS:

A. valve solenoids. This will be indicated by both red and green position indication lights being OFF for all ADS valves.

B✓ logic relays and timer relays. This will be indicated by annunciators A-03 (3-2) and A-03 (5-1) clearing.

C. logic relays. This will be indicated by annunciator A-03 (3-2) clearing but timer relays will remain energized and will be indicated by annunciator A-03 (5-1) remaining in alarm.

D. timer relays. This will be indicated by annunciator A-03 (5-1) clearing but logic relays will remain energized and will be indicated by annunciator A-03 (3-2) remaining in alarm.

Feedback

Reference SD-20 rev. 1 page 56 and 57 K5 and K8 are deenergized by INHIBIT switches. **APP A-03 3-2** - K8 energizes this alarm indicating logic relays energized. **APP A-03 5-1**-K5 timer relay energizes this alarm. Normal operator indications used in question so as to determine if operator understands exactly what is disabled when inhibit switches are placed in inhibit without having operator recall specific relay nomenclature without reference.

Distractor Analysis

A - incorrect - ADS valve solenoids remain energized for manual ops.
B - correct - inhibit switches deenergize both sides of ADS logic as indicated by annunciators
C & D - incorrect-inhibit switches deenergize both sides of ADS logic as indicated by annunciators.

Notes

SYSTEM: 218000 Automatic Depressurization System
K4. Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following:
(CFR: 41.7)
K4.01 Prevent inadvertent initiatior of ADS logic 3.7 3.9

This question matches the k/a in that it measures the RO's ability to verify that ADS logic has been inhibited through the use of normal RTGB indications.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.7	Facility Objective:	CLS-LP-020*012
Ref Req'd Y or N:	NO	Technical Ref.:	SD-20
? Cognitive Level:	C/A	? Source:	NEW

AUTO DEPRESS TIMERS INITIATED

AUTO ACTIONS

1. If a core spray pump or both RHR pumps in one loop are running, the timer will initiate an auto-depressurization after 83 seconds.

CAUSE

1. Reactor low level one (166 inches) and three (45 inches).
2. Circuit malfunction.

OBSERVATIONS

1. Reactor water level less than 45 inches and decreasing (multiple RTGB indications).
2. Emergency diesel generators operating.
3. RHR and Core Spray Systems operating.
4. CORE SPRAY OR RHR PUMPS RUNNING (A-03 2-1) alarm.
5. REACTOR ADS LO WATER LEVEL (A-03 4-2) alarm.
6. REACTOR LOW WTR LEVEL INITIATION (A-03 6-9) alarm.
7. AUTO DEPRESS RELAYS ENERGIZED (A-03 3-2) alarm.

ACTIONS

1. For a leak inside the drywell, refer to 0AOP-14.0.
2. For an abnormal reactor water level, refer to 0AOP-23.0.
3. For a loss of primary containment integrity, refer to 0AOP-14.0.
4. Verify primary containment group isolations.
5. Verify proper operation of the other ECCS subsystems.
6. For an automatic initiation of ADS, refer to 1OP-20.

CAUTION

After an automatic initiation, an ECCS subsystem or RCIC System shall not be shut down or placed in manual until at least two independent indications are verified for one of the following conditions:

1. **Adequate core cooling is ensured.**
2. **The initiation signal was not valid.**
3. **The system is not functioning properly in the automatic mode.**

7. If the Automatic Depressurization System was inadvertently started due to faulty instrumentation, when the reactor water level has been verified normal, shut down the Automatic Depressurization System and restore it to a standby configuration per 1OP-20.
8. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

ADS Time Delay Relay B21-K5A (actuated from B21-LTM-N042B-1 AND B21-LTS-N031B-3)	Energized
ADS Time Delay Relay B21-K5B (actuated from B21-LTM-N042A-1 AND B21-LTS-N031A-3)	Energized
Level Transmitter Master Trip Unit B21-LTM-N042A-1 and B-1	166 inches
Level Transmitter Slave Trip Unit B21-LTS-N031A-3 and B-3	45 inches

POSSIBLE PLANT EFFECTS

1. Inoperable equipment may result in a Technical Specification LCO.

REFERENCES

1. LL-93064 - 43
2. Technical Specification 3.3.5.1
3. 1APP-A-03 2-1, CORE SPRAY OR RHR PUMPS RUNNING
4. 1APP-A-03 4-2, REACTOR ADS LO WATER LEVEL
5. 1APP-A-03 6-9, REACTOR LOW WTR LEVEL INITIATION
6. 1APP-A-03 3-2, AUTO DEPRESS RELAYS ENERGIZED
7. 0AOP-14.0, Abnormal Primary Containment Conditions
8. 0AOP-23.0, Condensate/Feedwater System Failure
9. 1OP-20, Automatic Depressurization System Operating Procedure

AUTO DEPRESS RELAYS ENERGIZED

AUTO ACTIONS

1. Energizes one half of the ADS valve logic to allow opening of the seven auto relief valves when the ADS timer elapses at 83 seconds.

CAUSE

1. Reactor low level three (45 inches) in conjunction with Core Spray Pump A (115 psig) or RHR Pump A running (115 psig).
2. Reactor low level three (45 inches) in conjunction with Core Spray Pump B (115 psig) or RHR Pump B running (115 psig).
3. Circuit malfunction.

OBSERVATIONS

1. Reactor water level less than 45 inches and decreasing (multiple RTGB indications).
2. Emergency diesel generators operating.
3. RHR and Core Spray Systems operating in the minimum flow condition.
4. CORE SPRAY OR RHR PUMPS RUNNING (A-03 2-1) alarm.
5. REACTOR ADS LO WATER LEVEL (A-03 4-2) alarm.
6. REACTOR LOW WTR LEVEL INITIATION (A-03 6-9) alarm.
7. AUTO DEPRESS TIMERS INITIATED (A-03 5-1) alarm.

ACTIONS

1. For an abnormal reactor water level, refer to 0AOP-23.0.
2. For a loss of primary containment integrity, refer to 0AOP-14.0.
3. Verify primary containment group isolations.
4. Verify proper operation of the other ECCS subsystems.
5. For an automatic initiation of ADS, refer to 1OP-20.

CAUTION

After an automatic initiation, an ECCS subsystem or RCIC System shall not be shut down or placed in manual until at least two independent indications are verified for one of the following conditions:

1. **Adequate core cooling is ensured.**
2. **The initiation signal was not valid.**
3. **The system is not functioning properly in the automatic mode.**

6. If the Automatic Depressurization System was inadvertently started due to faulty instrumentation, when the reactor water level has been verified normal, shut down the Automatic Depressurization System and restore it to a standby configuration per 1OP-20.
7. If a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

ADS Auxiliary Relay B21-K8A (actuated from B21-LTS-N031D-3 AND Relay B21-K9A actuated from E21-PS-N008B or E11-PS-N016B or E11-PS-N020B)	Energized
ADS Auxiliary Relay B21-K8B (actuated from B21-LTS-N031C-3 AND Relay B21-K9B actuated from E21-PS-N008A or E11-PS-N016A or E11-PS-N020A)	Energized
Level Transmitter Slave Trip Unit B21-LTS-N031C-3 and D-3	45 inches
Pressure Switch E21-PS-N008A and N008B	114 psig (-1 psig head correction)
Pressure Switch E11-PS-N016A, N016B, N020A, N020B	117 psig (+2 psig head correction)

POSSIBLE PLANT EFFECTS

1. Inoperable equipment may result in a Technical Specification LCO.

REFERENCES

1. LL-93064 - 46
2. Technical Specification 3.3.5.1
3. 1APP-A-03 2-1, CORE SPRAY OR RHR PUMPS RUNNING
4. 1APP-A-03 4-2, REACTOR ADS LO WATER LEVEL
5. 1APP-A-03 6-9, REACTOR LOW WTR LEVEL INITIATION
6. 1APP-A-03 5-1, AUTO DEPRESS TIMERS INITIATED
7. 0AOP-14.0, Abnormal Primary Containment Conditions
8. 0AOP-23.0, Condensate/Feedwater System Failure
9. 1OP-20, Automatic Depressurization System Operating Procedure

FIGURE 20-10

ADS Logic Channel "A"

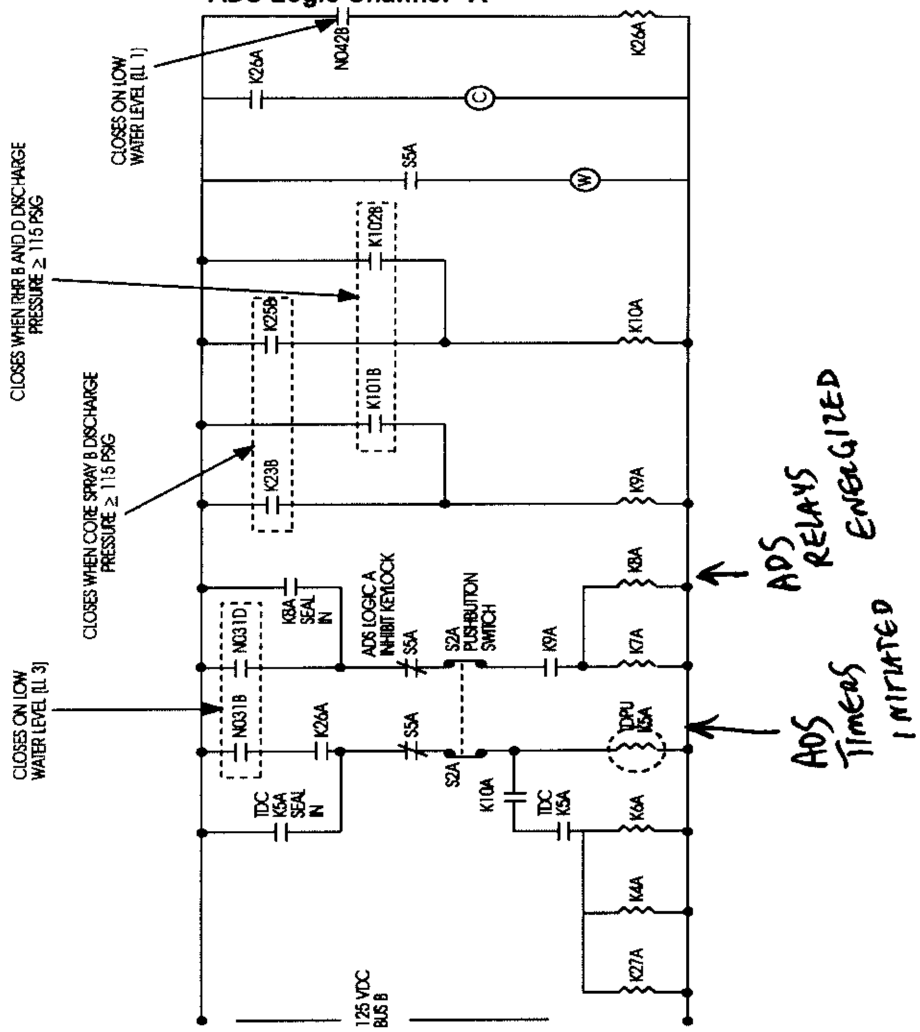
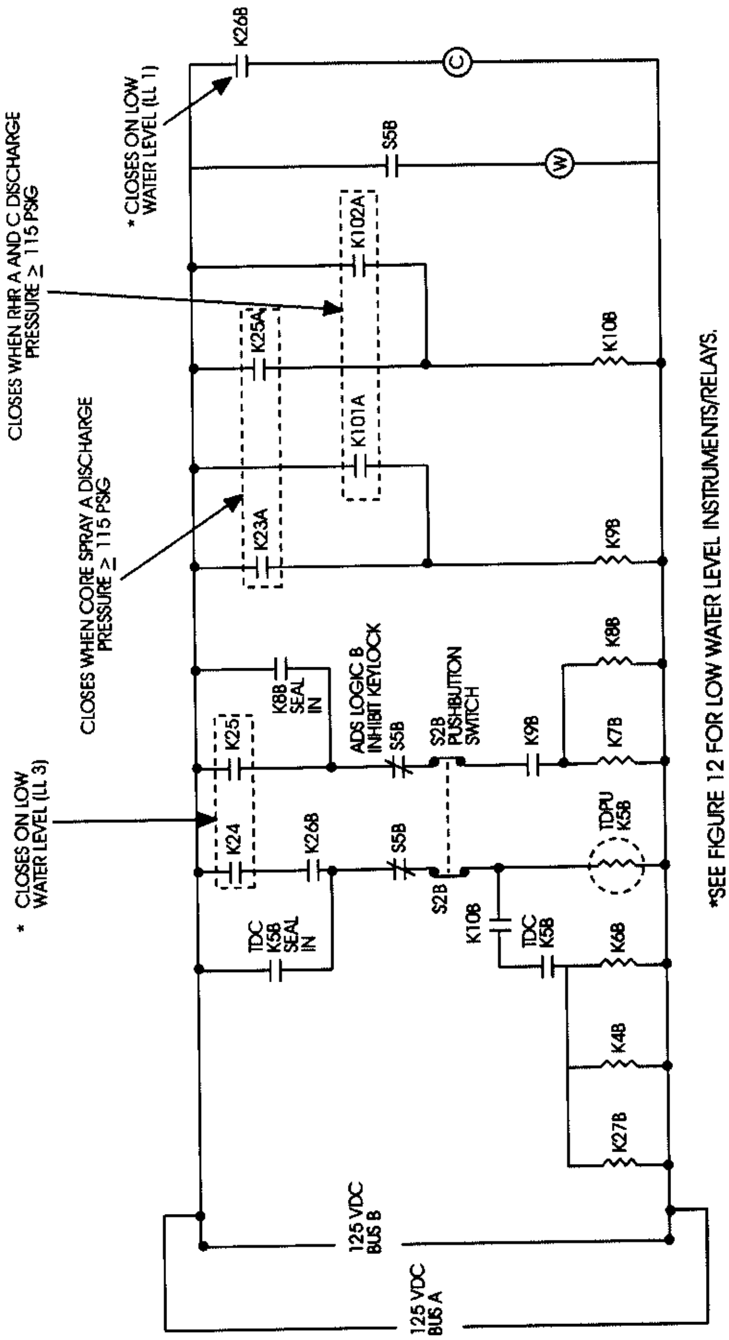


FIGURE 20-11
ADS Logic Channel "B"



20. 223001K406 001

Which ONE of the following describes the design feature that maintains proper secondary containment to primary containment differential pressure?

The Reactor Building to Suppression Chamber Vacuum Breaker isolation valves (CAC-V16 & CAC-V17) automatically open when their control switches are in Auto, and Reactor Building pressure exceeds:

A. Drywell pressure by 0.5 psid. These valves receive pneumatics from RNA or PNS.

B.✓ Drywell pressure by 0.5 psid. These valves receive pneumatics from RNA or backup nitrogen.

C. Suppression Chamber pressure by 0.5 psid. These valves receive pneumatics from RNA or PNS.

D. Suppression Chamber pressure by 0.5 psid. These valves receive pneumatics from RNA or backup nitrogen.

Feedback

Randomly selected from LOI Systems bank - LOI-CLS-LP-004-A*009 001
Reference SD-04 rev. 2 page 15 and 43
Distractor Analysis
A. - incorrect - these valves do not align to PNS but receive pneumatics from B/U Nitrogen to satisfy design function.
B - correct
C - incorrect - pressure switch used to monitor D/P actually senses DW pressure. Also B/U N2 not PNS.
D - incorrect - pressure switch used to monitor D/P actually senses DW pressure.

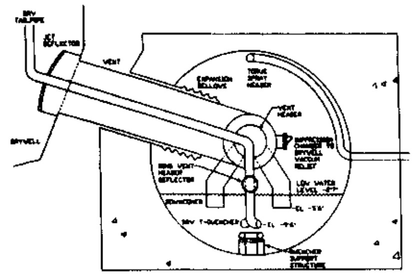
Notes

SYSTEM: 223001 Primary Containment System and Auxiliaries
K4. Knowledge of PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES design feature(s) and/or interlocks which provide for the following:
(CFR: 41.7)
K4.06 Maintains proper containment/secondary containment to drywell differential pressure 3.1 3.3

This question matches the k/a in that it measures the RO's knowledge of the design features that maintain proper primary to secondary containment D/P.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.1	Facility Objective:	CLS-LP-004*009
Ref Req'd Y or N:	NO	Technical Ref.:	SD-04
? Cognitive Level:	M OR FK	? Source:	BANK LOI



The steam discharged from the safety relief valves (SRV) is condensed by the suppression pool water. The SRV tailpipes are routed to the suppression pool through the vent pipes. The SRV tailpipe continues downward and terminates in a manifold (T-Quencher) approximately seven feet below normal water level.

Each T-Quencher is approximately twenty feet long and has holes along both sides of its length. The sparging effect of the T-Quencher provides even heat distribution in the suppression pool and reduced dynamic forces on the suppression chamber upon SRV actuation. The T-Quenchers are restrained (along the length) by gussets fixed to the Quencher Support Structure; the Quencher Support Structure is mounted to the bottom of the suppression chamber.

2.3 Vacuum Reliefs

The Reactor Building to Suppression Chamber vacuum relief valves (Figure 4-4A) protect the drywell and torus from exceeding their design external pressure of 2 psig. Two flow paths are provided from the RHR Pump Room air space to the air space of the torus.

Each flow path contains two valves in series. The first valve outboard of the torus, is an air-operated, normally closed butterfly valve. It is set to automatically open when the drywell is more negative than 0.5 psig with respect to the secondary containment. The second, is a self-actuating swing check, type vacuum relief. The swing check valves are designed to completely open within 1 second after a 0.5 psid is applied across the valve seat. An interlock of the Containment Spray System closes the containment spray valves on a 2.7 psig (decreasing) signal. This allows the vacuum relief valves time to deliver sufficient air such that the theoretical maximum external containment pressure is 1.6 psig.

The vacuum relief valves are testable using local lever arms. The Isolation Valves, CAC-V16 and CAC-V17 are controlled by switches on Panel XU-51. The switches are three position (CLOSE-AUTO-OPEN) maintained contact type. In the AUTO position, the valve automatically opens when the Reactor Building pressure is 0.5 psig greater than drywell pressure. In the OPEN position, the valve opens.

"Primary Containment Inspection and Closeout" procedure, 0AI-127.

Drywell cooling dampers are aligned per previous marked proper position (red paint designation). If dampers are found in wrong position, Operations should reposition to painted position during drywell closeout.

If any work has been performed beneath the reactor vessel, then the CRD equipment platform should be verified locked in place with a ball lock pin on the extreme north end of the platform and all grating removed and stored on the under-vessel floor. The platform hoist power supply should be turned "OFF". Performance of the above actions during Drywell Closeout will allow unobstructed movement of delicate under-vessel monitoring instrumentation and reduce/eliminate problems from damaged monitoring instrumentation, cables and connectors.

0PT-16.2, Drywell Volumetric Average Temperature, determines the operability of the Drywell per the requirements of Technical Specifications (average air temperature ≤ 150°F). The test involves collecting and averaging designated temperatures within the Drywell.

These temperatures may be obtained from the CAC-TY-4426-1(2) microprocessor, CAC-TR-4426-1(2), ERFIS Validation menus, designated process computer points or I&C manually measuring temperature from SPTMS RTDs.

0PT-2.3.2, Reactor Building To Suppression Chamber Vacuum Breaker Operability Test, determines operability of the reactor building to suppression chamber vacuum breakers, isolation valves and their associated check valves. Testing includes manually exercising the vacuum breakers via the external operating lever to the full open position and back to the closed position. As much as ten gallons of water has been documented as spillage when stroking a vacuum breaker. This requires proper radiological precautions to be taken prior to vacuum breaker opening.

Check valves associated with CAC-V16 and V17 are exercised to the forward and reverse positions by demonstrating the ability of CAC-V16 and V17 to be exercised via the Backup N_2 supply while the non-interruptible air supply header is vented to the atmosphere.

21. 223002A104 001

Unit Two (2) is operating at rated power when Main Steam Line Flow - Steam Line A instrument B21-PDT-N006A, which is trip channel A1 for Steam Line A, fails high.

Which ONE of the following describes the expected status of the MSIV Inboard and Outboard AC and DC solenoid indicating white lights on P601 following the failure of the Main Steam Line Flow - Steam Line A instrument ?

	Inboard DC	Inboard AC	Outboard DC	Outboard AC
A.	OFF	ON	ON	ON
B.	ON	OFF	ON	ON
C✓	ON	OFF	OFF	ON
D.	OFF	ON	ON	OFF

Feedback

Reference OI-18 rev. 51 page 18 - Trip Channel A1 is in PCIS Trip System A which deenergizes the two inner white lights (IN/AC and OUT/DC). No MSIV isolation occurs on a single instrument failure
Reference SD-12 rev. 3 page 14 and 18 - IN/AC and OUT/DC solenoids are deenergized by trip system A
Distractor Analysis - Candidate must be familiar with P601 indicating light layout in order to answer the question correctly.
A - incorrect - Chan A1 trips trip system A which deenergizes two inner lights, IN/DC light is denergized by trip system B.
B - incorrect - Chan A1 will deenergize both IN/AC and OUT/DC solenoids
C - correct
D - IN/DC and OUT/AC are deenergized by Trip system B

Notes

SYSTEM: 223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off
A1. Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLYSHUT-OFF controls including:
(CFR: 41.5 / 45.5)
A1.01 System indicating lights and alarms 3.5 3.5

This question matches the k/a in that it requires the RO to predict changes in MSIV PCIS logic light indication when instrument changes status.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-012*007
Ref Req'd Y or N:	NO	Technical Ref.:	SD-12
? Cognitive Level:	M OR FK	? Source:	NEW

Containment pressure will exceed this setpoint in less than one second after a design basis loss-of-coolant accident (LOCA). A break size of one square inch would cause 2 psig to be exceeded in less than a minute and a half, assuming an initial pressure of 0 psig. The containment is usually maintained between 0.15 and 1.25 psig during normal operation in order to meet the technical specification limits and operational considerations.

The power for the sensors, channels, and logics of the PCIS and NSSS system is supplied from either the reactor protection system (RPS) motor-generator sets, 120 Vac Distribution E bus power, 24/48 Vdc and 125/250 Vdc distribution. Isolation valves receive their power from the emergency AC buses, or 125/250 Vdc station distribution. The power for two required automatic isolation valves in a line is normally fed from different sources.

PCIS isolation functions are sometimes contained within the specific system logic (i.e., HPCI and RCIC). Other isolation functions are initiated by the Nuclear Steam Supply Shutoff (NSSS) system logic.

PCIS logic is separated into two Trip Systems, A and B. For NSSS logic, each trip system normally contains two trip channels for a total of four. Trip System A is comprised of Trip Channels A1 and A2; Trip System B is comprised of Trip Channels B1 and B2. You will sometimes see the trip channels referred to as A (A1),B (B1), C (A2) and D (B2). Also, isolation valves and logic are discussed frequently in terms of Inboard (usually Division I) and Outboard (usually Division II). Specific trip system and trip channel definitions are found in OI-18.

NSSS logic is usually arranged in a two-out-of-two logic scheme. Closure of Inboard isolation valves requires a signal from <u>both</u> trip systems (Trip Channels A1 <u>and</u> B1). Closure of Outboard requires signals from Trip Channels A2 <u>and</u> B2. MSIV closure uses a one-out-of-two logic scheme (A1 or A2, <u>and</u> B1 or B2), the same as RPS scram logic. Specific isolation trip logic arrangements are found in OI-18.

2.4 Isolations

2.4.1 **Group 1 Isolation** (Figures 12-1, 12-2, 12-16, 12-19, 12-20, 12-21)

A Group 1 isolation isolates the main steam lines, main steam drain lines, and Rx water sample lines to minimize the following:

- escape of fission products in the event of fuel failure,
- loss of reactor coolant in the event of a steam line break and,
- the potential for over pressurization of the main condenser.

The following valves are Group 1 Isolation valves:

1) Inboard MSIV's (B21-F022 A, B, C, D)
2) Outboard MSIV's (B21-F028 A, B, C, D)
3) Steam Line Drain Valves (B21-F016 and F019)
4) Reactor Recirc sample valves (B32-F019 and F020)*

*LL#3 or LO vacuum only.

Group 1 isolation logic is part of NSSS. Inputs to the Group 1 logic are from the RPS analog trip cabinets, or from steam leak detection. The isolation logic is powered from the RPS buses and de-energizes to cause isolation.

The Group 1 isolation logic <u>For MSIVs</u> (Figures 12-16, 12-19) is arranged in a one-out-of-two twice scheme. Each MSIV actuator has an AC and a DC solenoid. Each solenoid is controlled by a different trip system. Both solenoids must be de-energized for MSIV closure. The Inboard AC and Outboard DC solenoids are de-energized by Trip System A. The Inboard DC and Outboard AC solenoids are de-energized by Trip System B. Four white indicating lights on the P601 panel indicate status of power for each group of solenoids (i.e., Inboard MSIV AC solenoids). The power supply to the solenoids is:

- Inboard AC - RPS A
- Outboard AC - RPS B
- Inboard DC - DC Panel 3A (4A)
- Outboard DC - DC Panel 3B (4B)

INSTRUMENT NUMBER: B21-PDT-N006A, B, C, D

INSTRUMENT NAME: Main Steam Line High Flow - Steam Line A

TS REFERENCE: 3.3.6.1; TRM Table 3.3.6.1-1.1c, -1.1f (Unit 2 only)

TRIP CHANNEL: A1-N006A B1-N006B
A2-N006C B2-N006D

TRIP SYSTEM: A-A1 and A2
B-B1 and B2

TRIP LOGIC: A1 or A2 and B1 or B2 - Closes all MSIVs
A1 and B1 - Closes B21-F016
A2 and B2 - Closes B21-F019

Place channel in tripped condition by: Pull fuse

CHANNEL	INSTRUMENT NUMBER	TRIP UNIT	ACTION	PANEL	FUNCTION
A1	B21-PDT-N006A	B21-PDTM-N006A-1	A71B-F3A	H12-P609	Closes MSIVs, B21-F016
		B21-PDTS-N006A-2*			
A2	B21-PDT-N006C	B21-PDTM-N006C-1	A71B-F3C	H12-P609	Closes MSIVs, B21-F019
B1	B21-PDT-N006B	B21-PDTM-N006B-1	A71B-F3B	H12-P611	Closes MSIVs, B21-F016
B2	B21-PDT-N006D	B21-PDTM-N006D-1	A71B-F3D	H12-P611	Closes MSIVs, B21-F019

COMMENTS: If both channels in a trip system are inop, both channels must be tripped to assure all required functions occur.
*Slave trip unit for Unit 2 while in Modes 2 and 3. Trip logic satisfied from N006A-2 or N008C-2 **AND** N007B-2 or N009D-2.

REFERENCE DRAWINGS: 1-FP-55109, 2-FP-50056

22. 234000A3.02 001

Core offload is in progress. All control rods are inserted and the Mode Switch is in REFUEL.

A fuel assembly has just been released in the fuel pool and the main hoist raised to a safe elevation to pass through the cattle chute (BELOW Normal-Up position). The bridge is now being moved toward the core.

Which ONE of the following identifies when a Control Rod Block will FIRST be initiated?

A.✓ As soon as the bridge moves near or over the core.

B. When the bridge is over the core and the main hoist is lowered into the reactor vessel.

C. When the bridge is over the core, the main hoist is lowered, a fuel assembly is latched and the main hoist is loaded by raising the main hoist.

D. When the bridge is over the core, the main hoist is lowered and a fuel assembly is latched by closing the grapple hooks.

Feedback

LOI Systems bank question LOI-CLS-LP-305-A*011
Reference FH Big Notes BN 305.0.01
Distractor Analysis
A - correct - With main hoist not full up, rod block generated as soon as bridge is near/over the core. If main hoist is full up, B would be correct
C and D - incorrect for reason stated above

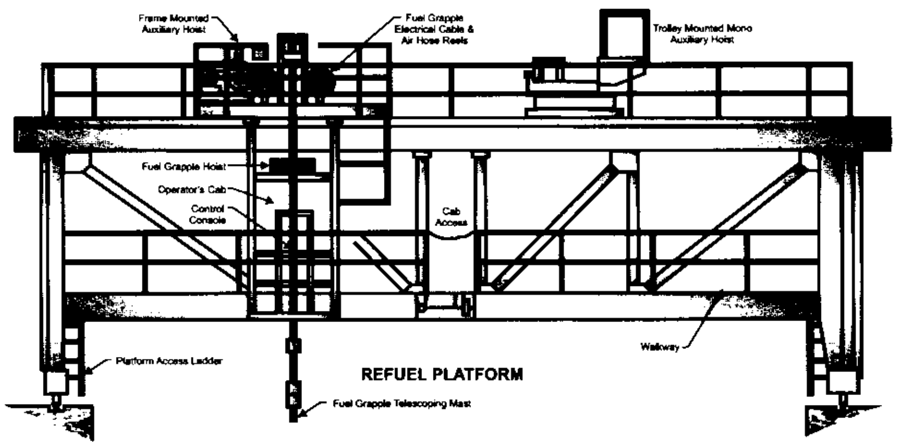
Notes

SYSTEM: 234000 Fuel Handling Equipment
A3. Ability to monitor automatic operations of the FUEL HANDLING EQUIPMENT including:
(CFR: 41.7 / 45.7)
A3.02 †Interlock operation 3.1 3.7

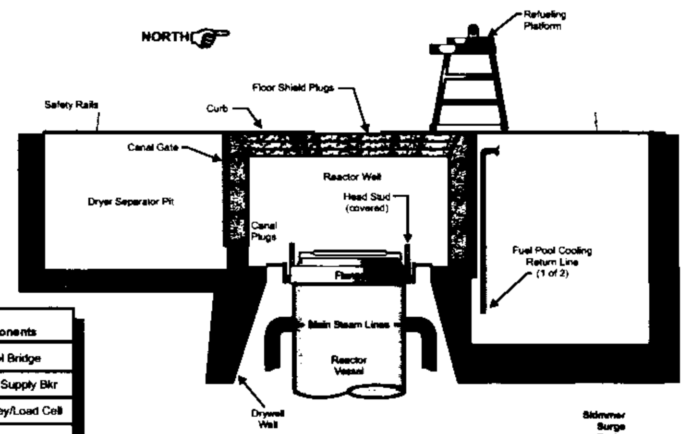
This question matches the k/a in that it measures the RO's knowledge of refueling interlocks when conditions to cuase automatic operation are met.

Categories

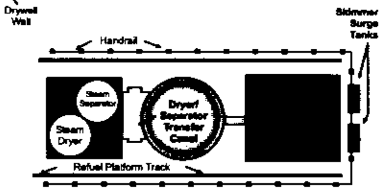
Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.1	Facility Objective:	CLS-LP-305-A*011
Ref Req'd Y or N:	NO	Technical Ref.:	BN 305.0.01
? Cognitive Level:	C/A	? Source:	BANK LOI

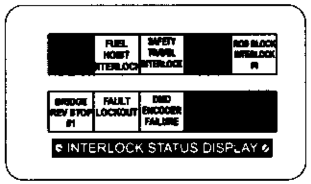


REFUEL PLATFORM



Power Supply	Components
1(2) XG	480V to Refuel Bridge
CB 50	Refuel Bridge Supply Bkr
CB 3	Monorail/Trolley/Load Cell
CB 4	Frame Hoist
CB 8	Fault Lockout electronics





MAIN HOIST NOT UP → BRIDGE NEAR OR OVER CORE (LS1)

ANY HOIST LOADED → BRIDGE NEAR OR OVER CORE (LS1)

BRIDGE NEAR OR OVER CORE (LS1) → ROD BLOCK

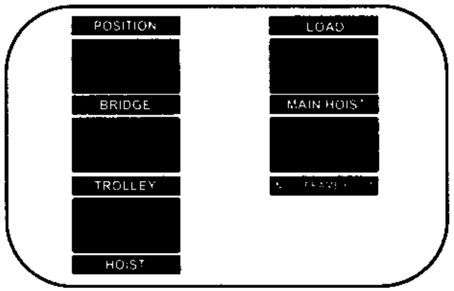
BRIDGE NEAR OR OVER CORE (LS1) → RODS NOT FULL IN / Rod Select Power ON / A Rod Selected / 137 GREEN Backlights → REV STOP #1

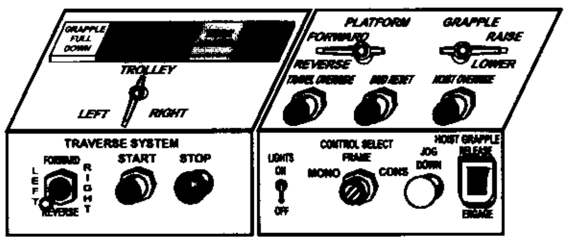
RODS NOT FULL IN / Rod Select Power ON / A Rod Selected / 137 GREEN Backlights → BRIDGE NEAR OR OVER CORE (LS1) → Fuel Hoist Interlock

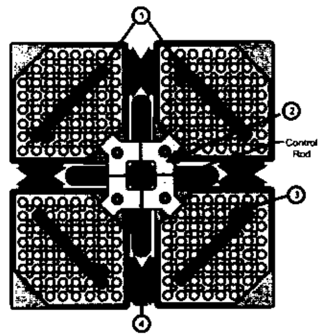
BRIDGE NEAR OR OVER CORE (LS1) → AUXILIARY HOIST LOADED → Auxiliary Hoist Interlock

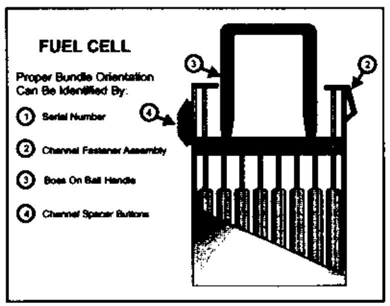
BRIDGE NEAR OR OVER CORE (LS2) → ROD BLOCK

BRIDGE NEAR OR OVER CORE (LS2) → REV STOP #1









Technical Specifications

Section 3.9.1 Refueling Equipment Interlocks
Section 3.9.2 Refueling Position One-rod-Out Interlock
Section 3.9.3 Control Rod Position
Section 3.9.4 Control Rod Position Indication
Section 3.9.5 Control Rod Operability-Refueling
Section 3.9.6 Reactor Pressure Vessel (RPV) Water level
Section 3.9.7 Residual Heat Removal (RHR)-High Water Level
Section 3.9.8 Residual Heat Removal (RHR)-Low Water Level
Section 4.2 Reactor Core
Section 4.3 Fuel Storage

BN-305.0.01 Refueling Operations	
Date: 02/26/02	Rev 01
P&ID:	Approved: MAP

23. 239002K201 001

Which ONE of the following describes the normal electrical power supply for Safety Relief Valve (SRV) solenoids on Unit Two (2)?

The solenoid power supply for all SRV solenoids is normally from:

A. Div I 125V DC

B. Div I 120V AC

C✓ Div II 125V DC

D. Div II 120 V AC

Feedback

Reference SD-20 rev. 1 page 20 - The solenoid power for all SRVs is normally 125V DC Panel 3(4)B. In the event Panel 3(4)B is lost, the SRVs will be automatically supplied by Panel 3(4)A.

Distractor analysis

A,B, and D - incorrect - normal P/S 125Vdc panel 4B which is Div II DC

C - correct

Notes

SYSTEM: 239002 Relief/Safety Valves

K2. Knowledge of electrical power supplies to the following:

(CFR: 41.7)

K2.01 SRV solenoids 2.8* 3.2*

This question matches the k/a in that it measures the ROs knowledge of SRV solenoid power.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.8	Facility Objective:	CLS-LP-020*14B
Ref Req'd Y or N:	NO	Technical Ref.:	SD-20
? Cognitive Level:	M OR FK	? Source:	NEW

to overcome the main valve preload spring tension or the valve will not open. The open SRV will then close once Primary Coolant system pressure decreases to approximately 50 psid above primary containment.

3.2.3 **Remote Shutdown Panel** (Figure 20-13)

The Remote Shutdown Panel (RSDP, RS-4) located on the 20' elevation of the Reactor Building, has two-position NORMAL/LOCAL SRV selector switches for each of the three manually operated SRVs (B21-F013B, E, and G). Once the selector switch is placed to the LOCAL position, the associated three-position CLOSE/OFF/OPEN control switch, also located on the RSDP, may then be positioned to the OPEN position to manually open the valve.

3.2.4 **Fluid Flow Detector Cabinet**

The FFD Cabinet, located in the electronic equipment room, contains the THRESHOLD/INPUT maintained position toggle switch for each SRV. As previously mentioned, each of these switches is normally left in the INPUT position and the FFD Cabinet display for the associated sensor will display the actual signal level being generated by that sensor. If the switch is moved to the THRESHOLD position, the FFD Cabinet display will change to read the trip setpoint.

TEST/RESET spring-loaded return to NEUTRAL toggle switches are also provided for each SRV at the FFD Cabinet. To de-energize both of the amber memory lights, the TEST/RESET switch on the FFD must be operated to the RESET position. When this switch is operated to the TEST position, a signal is generated to test the associated lights. After a test, the amber memory light must again be reset.

Power Supplies (Figures 20-10, 11, 12)

There are two logic circuits, Logic A and Logic B, either of which when energized will open all of the ADS valves. Logic A receives power from 125V DC Panel 3(4)B. Logic B also receives power from the 125V DC Panel 3(4)B. In the event normal power from Panel 3(4)B is lost, the Logic B circuit will be automatically supplied by Panel 3(4)A.

The solenoid power for all SRVs is normally 125V DC Panel 3(4)B. In the event Panel 3(4)B is lost, the SRVs will be automatically supplied by Panel 3(4)A. Panel 3(4)A also supplies power to the ADS Logic B low level relays which do NOT have an alternate power source.

24. 245000K302 001

While attempting to synchronize the main generator to the grid on Unit Two (2) at 23% rated thermal power a generator primary lockout occurred while the operator was raising generator output voltage towards rated terminal voltage.

Assume all systems operated as expected.

Which ONE of the following describes the effect that the generator lockout will have on reactor pressure?

Reactor pressure will initially:

A. increase rapidly until controlled by bypass valves. After conditions stabilize reactor pressure will be slightly higher than the steady state pressure prior to the generator lockout.

B✓ increase rapidly until controlled by bypass valves. After conditions stabilize reactor pressure will be approximately the same as the steady state pressure prior to the generator lockout.

C. decrease rapidly until controlled by bypass valves. After conditions stabilize reactor pressure will be slightly lower than the steady state pressure prior to the generator lockout.

D. decrease rapidly until controlled by bypass valves. After conditions stabilize reactor pressure will be approximately the same as the steady state pressure prior to the generator lockout.

Feedback

Reference CLS-LP-110-A page 18 discusses Reactor Pressure transients such as for a turbine trip - Pressure is proportional to the amount of steam leaving the reactor in a saturated system. Due to EHC characteristics, pressure is also affected by EHC pressure set and load. Finally, pressure in the steam dome is a function of steam flow, since pressure is controlled at a point significantly downstream from the reactor. A generator trip at this power results in a turbine trip which causes a rapid reactor pressure increase as steam flow is suddenly stopped. After conditions stabilize the loss of extraction steam will cause inlet subcooling to increase thus reactor power increases but reactor pressure remains the same as EHC maintains pressure constant.This could be confusing since steady state to state the reactor power will increase since it will take more energy to make the same steam flow to maintain EHC PAM pressure constant.

Distractor Analysis

A - incorrect - although reactor power increases due to increased subcooling reactor pressure will be maintained constant by EHC or the same with turbine tripped as compared to before turbine trip

B- correct

C and D - incorrect - initial pressure change is a rapid pressure increase due to collapsing voids

Notes

SYSTEM: 245000 Main Turbine Generator and Auxiliary Systems
K3. Knowledge of the effect that a loss or malfunction of the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS will have on following:
(CFR: 41.7 / 45.4)
K3.02 Reactor pressure 3.9 4.0

This question matches the k/a since it measures the RO's knowledge of cause and effect between turbine trip and ractor pressure at low power.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.9	Facility Objective:	CLS-LP-110A*4B
Ref Req'd Y or N:	NO	Technical Ref.:	CLS-LP-110A
? Cognitive Level:	C/A	? Source:	NEW

E. PLANT PARAMETERS

> **NOTE:** The parameters listed in this material for Unit 1 and Unit 2 do not necessarily illustrate current BNP parameters in every case. The figures attached to this material were extracted from a nuclear engineering software program that is no longer being maintained however the figures themselves are still suited for learning the mechanics of Anticipated Operational Occurrences. Actual current plant parameters and transient response data may be found in Technical Specifications, UFSAR, COLR, and/or other controlled plant documents as applicable.

1. Reactor Pressure

Pressure is proportional to the energy of the steam leaving the reactor in a saturated system. Due to EHC characteristics, pressure is also affected by the EHC pressure setpoint and load. Finally, pressure in the steam dome is a function of steam flow, since pressure is controlled at a point significantly downstream from the reactor.

Any event, which causes an increase in core power without a corresponding immediate increase in steam flow, will result in a pressure rise. Conversely, any event, which causes a decrease in steam flow without a corresponding immediate decrease in core power, will cause an increase in vessel pressure. The reverse of the two situations above also applies. Examples of this principle would include main steam isolation valve (MSIV) closure (pressure rise) and safety/relief valve (SRV) lift (pressure drop).

The EHC System controls turbine control valve (TCV) position based on a pressure signal sensed at the transducer header between the MSIVs and TCVs. The system is set to achieve 0% to 100% power over a sensed 0 to 30 psi rise above the pressure setpoint. As a result, pressure and thus power, as measured by transducer header pressure, controls TCV position and, consequently, generator output.

As an example, if the EHC pressure setpoint is 945 psig and reactor power is 50%, actual transducer pressure would be 960 psig. An increase in reactor power to 75% would cause a pressure rise. TCVs would open to pass 75% steam flow to the turbine (neglecting non-main turbine flow), and header pressure would stabilize at 367.5 psig. It can be seen from the above that two EHC characteristics affect reactor vessel pressure. As the pressure setpoint is changed, EHC will attempt to maintain the new pressure by adjustment of TCVs, and as power rises, the EHC droop characteristic by itself, would cause a small rise in EHC transducer header pressure.

25. 256000K201 001

Unit Two (2) is in operation with the following Condensate System alignment:

Condensate pumps	2A, 2B & 2C running
Condensate pump 2C	Unit trip load shed enabled
Condensate Booster pumps	2B & 2C running
Condensate Booster pump	2A in standby with unit trip load shed enabled

A reactor scram is initiated by a turbine trip. The main generator backup lockout relay trips. BOP Bus 2C fails to transfer to the SAT.

Which ONE of the following identifies how many Condensate and Condensate Booster pumps, if any, are immediately available to provide makeup to the Reactor Vessel?

A. One Condensate pump but no Condensate Booster pump.

B✓ One Condensate pump and one Condensate Booster pump.

C. Two Condensate pumps and one Condensate Booster pump.

D. Two Condensate pumps and two Condensate Booster pumps.

Feedback

LOI Systems Bank question - LOI-CLS-LP-032-A*06A
Reference SD-32 - CP A-D B-C C-D AND CBP A-C B-D C-C; Power to Condensate Pumps is Bus D for A/C and Bus C for B, Booster Pumps is Bus C for A/C and Bus D for B. Cond pump B and Booster pumps A/C have now power. Cond pump C and booster pump A trip from unit trip load shed.
Distractor Analysis
A, C and D incorrect - Condensate pump A and Booster pump B will be running.
B - correct

Notes

SYSTEM: 256000 Reactor Condensate System
K2. Knowledge of electrical power supplies to the following:
(CFR: 41.7)
K2.01 System pumps 2.7* 2.8

This question matches the k/a in that it measures the ROs knowledge of power supplies to condensate system pumps after an electric plant alignment change and load shed. This question takes fundamental knowledge question to the analysis level since Condensate system power supplies are subject to change when electric plant realigns (turbine trip) and may be tripped under certain load shed conditions.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 2.7	Facility Objective:	CLS-LP-032*06A
Ref Req'd Y or N:	NO	Technical Ref.:	SD-32
? Cognitive Level:	C/A	? Source:	BANK LOI

TABLE 32-2
Page 1 of 4
Power Supplies

CONDENSATE AND FEEDWATER SYSTEM

COMPONENT	POWER SUPPLY (BUS OR MCC)
Condensate Pump 1A (2A)	4 KV Bus 1D (2D)
Condensate Pump 1B (2B)	4 KV Bus 1C (2C)
Condensate Pump 1C (2C)	4 KV Bus 1D (2D)
Condensate Booster Pump 1A (2A)	4 KV Bus 1C (2C)
Condensate Booster Pump 1B (2B)	4 KV Bus 1D (2D)
Condensate Booster Pump 1C (2C)	4 KV Bus 1C (2C)
Hotwell Isolation Valve 1(2)-CO-V1	MCC 1TB (2TB)
Hotwell Isolation Valve 1(2)-CO-V2	MCC 1TD (2TD)
Hotwell Isolation Valve 1(2)-CO-V3	MCC 1TE (2TE)
Hotwell Isolation Valve 1(2)-CO-V4	MCC 1TE (2TE)
Condensate Pump 1(2)A Discharge Valve 1(2)-CO-V11	MCC 1TB (2TB)
Condensate Pump 1(2)B Discharge Valve 1(2)-CO-V12	MCC 1TE (2TE)
Condensate Pump 1(2)C Discharge Valve 1(2)-CO-V13	MCC 1TB (2TB)
SJAE A-Train Intercondenser Outlet Valve 1(2)-CO-V15	MCC 1TB (2TB)
SPE A-Train Inlet Valve 1(2)-CO-V14	MCC 1TB (2TB)
SJAE B-Train Intercondenser Outlet Valve 1(2)-CO-V17	MCC 1TE (2TE)
SPE B-Train Inlet Valve 1(2)-CO-16	MCC 1TE (2TE)
Turbine Exhaust Hood Spray Valve 1(2)-CO-V19	MCC 1TN (2TN)
Condenser A-N SJAE Recirc. 1(2)-CO-V111	MCC 1TN (2TN)
Condenser A-S SJAE Recirc. 1(2)-CO-V112	MCC 1TG (2TG)

26. 259001K104 001

Implementation of Final Feedwater Temperature Reduction (FFWTR) is in progress on Unit One (1) per 0GP-13. The RO will be isolating extraction steam to 4A and 4B Feedwater Heaters (FWH).

Which ONE of the following describes the effect on the reactor feedwater system when extraction steam is isolated to 4A and 4B feedwater heaters?

Gravity drainage from 4A and 4B FWHs to:

A.✓ the deaerator may be reduced which could in turn result in a higher level in the East/West MSR drain tanks which may result in a main turbine trip .

B. the deaerator may be reduced which could in turn result in 4A and 4B emergency dump valves to the condenser opening to control level in the FWHs.

C. 3A and 3B FWHs may be reduced which could in turn result in 5A and 5B FWHs having a higher temperature gradient and steam velocity.

D. 3A and 3B FWHs may be reduced which could in turn result in a HDD level change, which may result in a HDP trip.

Feedback

Reference SD-34 rev. 8 page 52 If any feedwater heating is bypassed the next higher heater will have a higher temperature gradient and steam velocity. If extraction steam is isolated to any FW heater, HDD level may change, which may result in a HDP trip. **Reference CAUTION GP-13 rev. 19 page 19 -** If level in the MSR Drain Tank is allowed to increase while bypassing the 4A and 4B FW heaters, a turbine trip may result. Procedure requires that FWH shell to HDD D/P does lower below 10 psid which would reduce gravity drainage from #4 FWHs to HDD.

Distractor Analysis

A - correct

B - incorrect - #4FWHs do not have emergency dump valves to the condenser.

C and D incorrect - #4FWHs do not drain to #3FWHs.

Notes

SYSTEM: 259001 Reactor Feedwater System

K1. Knowledge of the physical connections and/or causeeffect relationships between REACTOR FEEDWATER SYSTEMand the following:

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.04 Extraction steam system 2.8 2.9

This question matches the k/a since it measures the ROs knowledge of extraction steam/ feedwater interrelationship. Scenario described allows RO to demonstrate knowledge of this interrelationship.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 2.8	Facility Objective:	CLS-LP-034*5,6,&7
Ref Req'd Y or N:	NO	Technical Ref.:	SD-34 GP-13
? Cognitive Level:	C/A	? Source:	NEW

FIGURE 34-4
Extraction Steam to High Pressure Feedwater Heaters

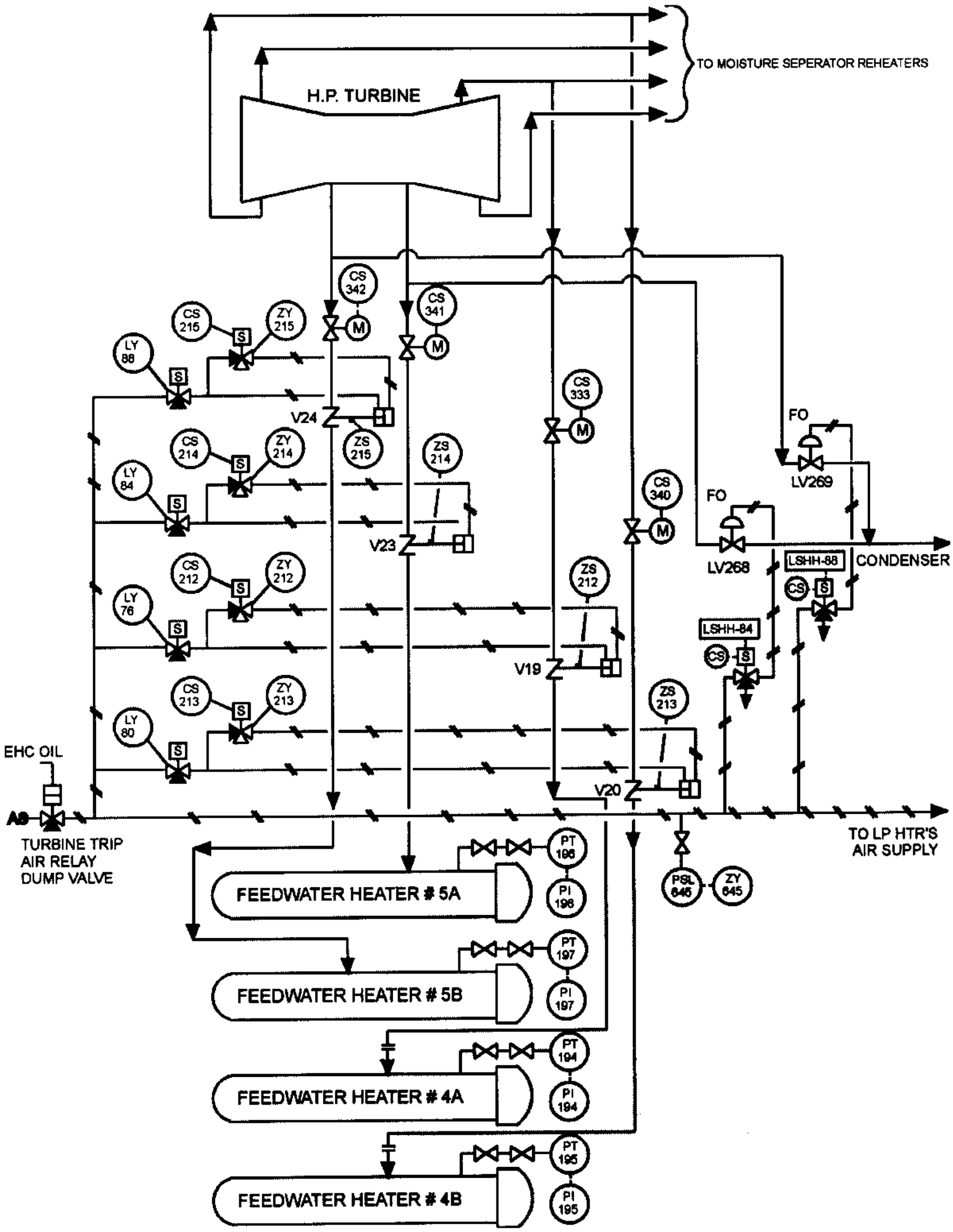
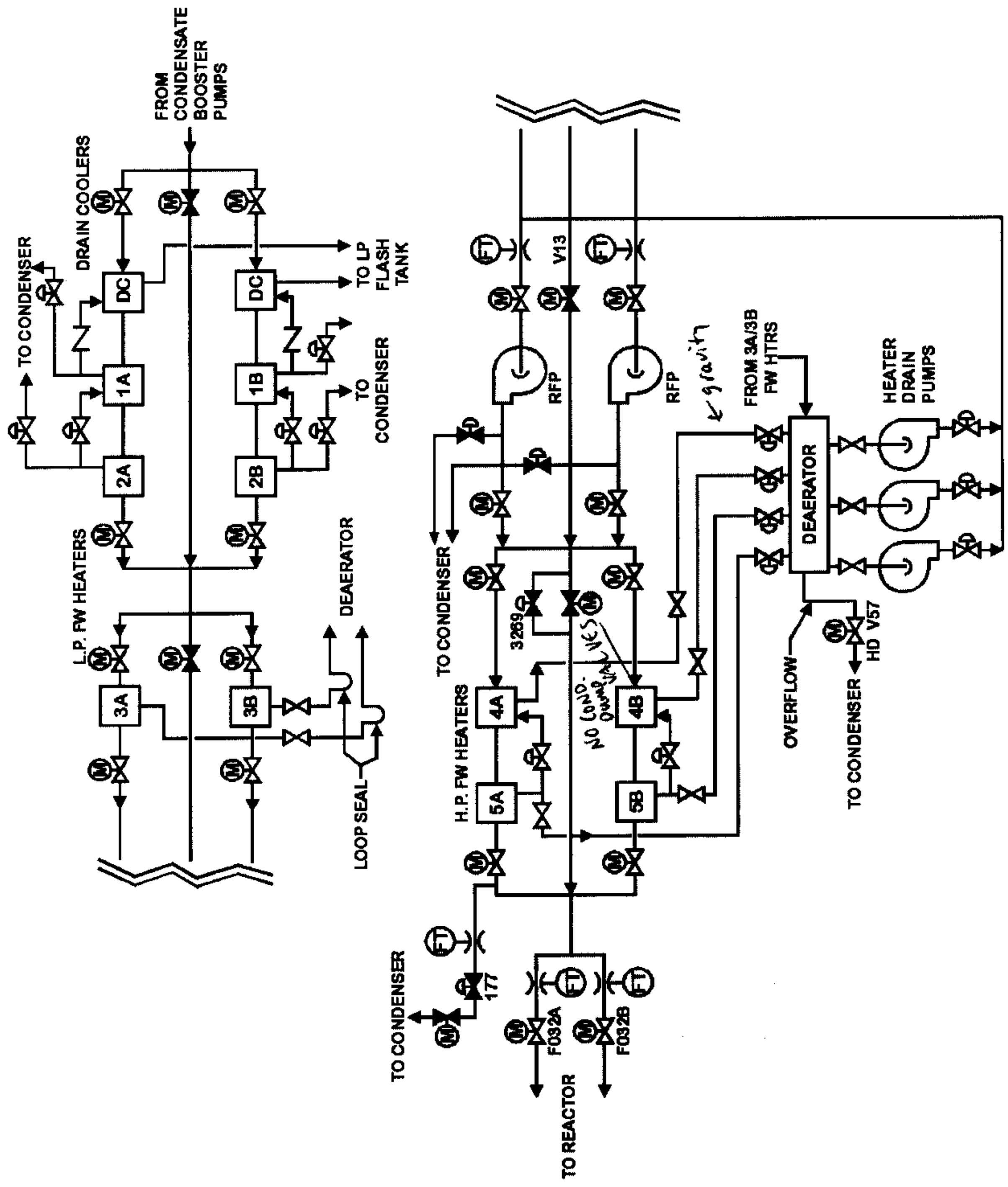


FIGURE 34-1
Heater Drains, Vents, and Level Control System
Feedwater Heating Simplified Flowpath



During restoration from the postmaintenance run-in alignment the procedure precautions that the air isolation to the heater drain pump discharge deaerator level control valve must be opened slowly to prevent:

- An air header pressure transient which would cycle the running heater drain pump level control valve and cause a feedwater transient.

- Unnecesssary rapid cycling of the selected heater drain pump level control valve.

The Feedwater and Condensate Infrequent Operating Procedures OP-32 Section 8.0 includes:

- Section 8.7 Bypassing Feedwater Heaters.

- Cautions:

If any feedwater heating is bypassed

- Reactor power will increase and thermal limits may be approached or exceeded.

- The next higher heater will have a higher temperature gradient and steam velocity.

- If both feedwater heater strings are bypassed, then feed flow should be limited to 5 million lbm/hr.

- If extraction steam is isolated to any FW heater, HDD level may change, which may result in a HDP trip.

- If level in the MSR Drain Tank is allowed to increase while bypassing the 4A and 5A (4B and 5B) FW heaters, a turbine trip may result.

- Generator Load Reduction recommendations for removal of Feedwater Heaters should be reviewed.

5.0 PROCEDURAL STEPS

CAUTION

WHEN Extraction Steam is isolated, reactor power will rise.

CAUTION

The time Extraction Steam is unbalanced from the turbine should be minimized.

5.3.7 **ADJUST** FEEDWATER HEATER 4A level controller to the low end of the normal operating level band in accordance with Attachment 2. ____

5.3.8 **CLOSE** *EXTRACTION STEAM ISOLATION VALVE TO FEEDWATER HEATER 4A, EX-V17* to remove Extraction Steam from Feedwater Heater 4A. ____

CAUTION

IF the MSR Drain Tank level is allowed to rise, a Turbine trip may result.

CAUTION

Feedwater Heater 4A Shell to Heater Drain Deaerator differential pressure should **NOT** be reduced below 10 psid.

5.3.9 **MONITOR** the parameters listed on Attachments 1 and 2. ____

5.3.10 **IF** Feedwater Heater 4A Shell to Heater Drain Deaerator differential pressure lowers below 10 psid, **THEN OPEN** *EXTRACTION STEAM ISOLATION VALVE TO FEEDWATER HEATER 4A, EX-V17* **AND RETURN** to Step 5.1.13. ____

27. 259002A206 001

Unit Two (2) is operating at power. Reactor Feed Pump (RFP) 2A is operating in automatic DFCS control at 4500 RPM when the following alarm is received:

UA-13 (6-5) RFP A CONTROL TROUBLE

The RO observes that RFPT 2A DFCS CTRL light on XU-1 is NOT illuminated and 2-RFA-SI-7325 for RFP 2A on P603 has the following indications:

DFCS STPT	1704 RPM
SPEED STPT	5201 RPM
ACT SPEED	5203 RPM

Which ONE of the following describes how RFP 2A will respond and what actions are required to control RFP 2A under this condition?

RFP 2A will:

A✓ remain at the current speed until changed by the RO manually by using the LOWER/RAISE Speed Control Switch on XU-1.

B. remain at the current speed and can ONLY be lowered or raised from the local Operator Control Panel in the Breezeway.

C. remain at the current speed until changed by the RO manually by using the LOWER/RAISE pushbuttons at RFP 2A panel display station on P603 after placing it in manual.

D. automatically lower to 1704 RPM unless the RO takes manual control of RFP 2A by placing RFP 2A panel display station at P603 in manual.

Feedback

Reference APP UA-13 (6-5) rev. 29 page 89- indications are consisitent with a loss of DFCS control signal which will require operator action to change RFPT speed by using the LOWER/RAISE speed control switch at XU-1.

Distractor Analysis

A - correct

B - incorrect local operator control panel is not the ONLY location to lower or raise speed

C - incorrect - DFCS signal has failed and affects both Manual and Auto signals so panel display stations at P603 will not function.

D - loss of DFCS the RFP will automatically operate at speed stpt and will ignore DFCS signal.

Notes

SYSTEM: 259002 Reactor Water Level Control System

A2. Ability to (a) predict the impacts of the following on the REACTOR WATER LEVEL CONTROL 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)

A2.06 Loss of controller signal output 3.3 3.4

This question matches the k/a in that it measures the ROs ability to diagnose a signal failure for feedwater controllers and requires RO to demonstrate proper mitigating actions.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.3	Facility Objective:	CLS-LP-32.2*08J
Ref Req'd Y or N:	NO	Technical Ref.:	SD-32.3 APP UA-13
? Cognitive Level:	C/A	? Source:	NEW

RFP A CONTROL TROUBLE

AUTO ACTIONS

NONE

CAUSES

1. Alarm initiation on local panel 2RFA-5009-2A for A RFPT

OBSERVATIONS

1. Local panel 2RFA-5009-2A in alarm

ACTIONS

1. Inform Turbine Building AO of alarm initiation and request investigation of alarm condition.

2. Monitor reactor water level and feedwater flow for possible loss of A RFP.

3. IF RFPT 2A *DFCS CTRL* light on RTGB XU-1 is NOT illuminated, THEN attempt to control RFP turbine speed as necessary using the *LOWER/RAISE* speed control switch.

4. Refer to 0AOP-23.0.

DEVICE/SETPOINTS

Any local alarm initiation at Panel 2RFA-5009-2A

POSSIBLE PLANT EFFECTS

1. Possible loss of A RFP

REFERENCES

1. LL-9351-38
2. 0AOP-23.0, Condensate/Feedwater System Failure

28. 261000K403 001

Which ONE of the following describes Standby Gas Treatment (SBGT) System design features that provide for removal of internal moisture that may collect on adsorbers and HEPA filters?

Each compartment within the SBGT train casing:

A. contains a desicant cartridge that eliminates moisture from the adsorbers and HEPA filters. These desicant cartridges must be periodically replaced.

B. has a water absorbent, dry type, replaceable prefilter that will hold any excessive moisture that may attempt to collect on adsorbers and HEPA filters. These filters must be periodically replaced.

C✓ has a drain line. In order to eliminate moisture from the adsorbers and HEPA filters flow must be established by operating the SBGT system periodically for several hours.

D. is serviced by a heater. The heater must be left in automatic operation continuously in order to maintain relative humidity <70% thus eliminating moisture from the adsorbers and HEPA filters.

Feedback

Reference SD-10 Rev. 4 page 7 and SR 3.6.4.3.1 TS Bases rev. 30 page B 3.6.4.3-5 - States that SBGT must be operated 10 hours continuously every 31 days to eliminate moisture. Flow required to be established to force moisture out drain lines contained in each compartment.

Distractor Analysis

A - incorrect - no desicant cartridges in compt.

B - incorrect - there is only one prefilter per SBGT casing not one in each compt. Prefilter is designed to remove large particulate matter.

C - correct

D - incorrect - operation of heater alone will not prevent collection of moisture. Heater is designed to lower relative humidity of entrained gases to <70%.

Notes

SYSTEM: 261000 Standby Gas Treatment System

K4. Knowledge of STANDBY GAS TREATMENT SYSTEM design feature(s) and/or interlocks which provide for the following:

(CFR: 41.7)

K4.03 Moisture removal 2.5 2.7

This question matches the k/a in that it measures the RO's knowledge of SBGT design features associated with moisture removal.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.5	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	SD-10 TS BASES
? Cognitive Level:	M OR FK	? Source:	NEW

2.0 SYSTEM DESCRIPTION/DESIGN DATA

2.1 Major Components

2.1.1 Moisture Separator

The Moisture Separator provides the first stage of moisture, removal from the gases entering the SBGT filter train. The separator consists of three, two-feet by two-feet, vertically mounted woven mesh screens arranged in alternate layers. Screen inlet baffles are constructed with traps which strip away the entrained water. The removed water drains to the filter train drains. Each compartment within the SBGT train casing (7) has a drain line. These drains are collected into a common trough located on the side of each unit.

The trough itself contains water above the level of the drain lines which forms a loop seal. The loop seal prevents ventilation flow from entering the reactor building via these drains up to a pressure of approximately 0.7 psig. The overflow from the drain trough flows to the Radwaste System. This trough is filled manually as required from the Well Water System.

The separator is rated 99.9% efficient at a gaseous flow rate of 3000 scfm containing 0.005 pounds of moisture, consisting of water spray or condensed steam, per cubic foot of air.

Removal of excessive moisture is necessary to prevent loading the HEPA Filters and the resultant damage caused by operation with a high differential pressure across the filter. Damage to the HEPA Filter can then allow moisture and large particles to enter the Carbon Filters. Moisture occupies the adsorption sites on the charcoal, decreasing the efficiency of the Carbon Filters and particles can plug the Carbon Filters and cause high differential pressure.

In the case of either HEPA Filter or Carbon Filter depletion and damage, the filter train can become inoperable, requiring entry into a Technical Specification Limiting Condition for Operation (LCO).

2.1.2 Heater

The electric Heater raises the temperature of the gas entering the filter train to lower the relative humidity to less than 70%.

BASES

ACTIONS

D.1, D.2.1 and D.2.2 (continued)

assemblies would not be a sufficient reason to require a reactor shutdown.

E.1 and E.2

When two SGT subsystems are inoperable, if applicable, movement of recently irradiated fuel assemblies in secondary containment must immediately be suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must immediately be initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since recently irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving recently irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving recently irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of recently irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating each SGT subsystem, by initiating (from the control room) flow through the HEPA filters and charcoal adsorbers, for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on automatic control for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

29. 262001K102 001

Unit One (1) is operating at rated power, when the feeder breaker to 125V DC panel 9A trips.

Which ONE of the following describes how this loss of DC power will affect the manually initiated, automatically executed fast bus transfer capability of 4Kv Switchgear Busses 1B, 1C and 1D?

Manually initiated, automatically executed fast bus transfers will:

A. occur if attempted for busses 1B, 1C, and 1D.

B. not occur if attempted for busses 1B, 1C, and 1D.

C✓ occur if attempted for bus 1B, but will not occur if attempted for busses 1C and 1D.

D. occur if attempted for busses 1C and 1D, but will not occur if attempted for bus 1B.

Feedback

Reference 0OI-50 Rev. 29 pages 57-59 - BOP bus 1B has AUTO control power transfer capability where 1C and 1D do not.

Distractor Analysis

A, C, and D. incorrect - 1B control power AUTO swaps to 125Vdc panel 10A and will have ability to operate at RTGB but 1C and 1D have no control powet therefore cannot be operated at RTGB.

B - correct

Notes

SYSTEM: 262001 A.C. Electrical Distribution

K1. **Knowledge of the physical connections and/or causeeffect relationships between A.C. ELECTRICAL DISTRIBUTION and the following:**

(CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.02 D.C. electrical distribution 3.3 3.6

This question matches the k/a in that it measures the RO's knowledge of the cause and effect relationship between 4KV AC power and a loss of DC control power.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.3	Facility Objective:	CLS-LP-50.1*15A
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-50
? Cognitive Level:	M OR FK	? Source:	NEW

PANEL: 9A Reference Drawing: LL-30024-12	LOCATION: Unit 1 Turbine Building, 20', Switch gear area	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: Switchboard 1A (Mechanical Interlock)

CKT #	LOAD	EFFECT
3	4KV Switchgear Bus 1B Control Power	1. Automatic Bus Transfer to alternate power, Panel 10A, ckt. 11. Note: This control power feeds the following loads: a) 1A and 1B MG Set Logic • Drive motor closing and trip circuit. • Breaker local and remote indication • *Recirc Gen. Field Breaker, control, trip and indication • *Recirc Lube Oil Pumps A-1 and A-2 or B-1 and B-2 control and indication. • *Recirc Scoop Tube Power Failure Lock & Reset. • *ATWS Trip Logic *These loads have alternate power from Panels 3A and 3B. If power swaps to the alternate source, annunciator A-06-6-4 and/ or A-06-6-5 would be received. The local lockout switch must be reset to clear the alarms. b) Incoming Line Breakers from the UAT and SAT, control and indication. c) BOP 4KV Bus B undervoltage and overcurrent protection. d) SAT Cross Tie Breaker. (Not used)
4	Bus Common A Control Power`	1. Loss of control power to 4KV loads on Common A 2. Loss of 4KV breaker operation, manual or automatic. 3. 4KV breakers fail as is. 4. Loss of breaker indication locally and on the RTGB.

PANEL: 9A Reference Drawing: LL-30024-12	LOCATION: Unit 1 Turbine Building, 20', Switch gear area	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: Switchboard 1A (Mechanical Interlock)

CKT #	LOAD	EFFECT
5	Generator Field Rectifiers High Temperature Alarm Circuit	1. Receive annunciator 1-UA-02-5-9. 2. Local high temp lights will not illuminate.
6	4KV Switchgear Bus 1C Control Power	1. Loss of control power to 4KV loads on Bus 1C. 2. Loss of 4KV breaker operation, manual or automatic. 3. Loss of 4KV protective functions for Bus 1C and 4KV loads. 4. Bus 1C or 1D will not automatically 'dead bus' auto transfer from the UAT to SAT. (Loss of high speed transfer sync. circuit) 5. Bus 1C or 1D cannot be manually initiated, auto executed, fast bus transferred. (Loss of high speed transfer sync. circuit) 6. Loss of breaker indication locally and on the RTGB. 7. Div II Loss of BOP Bus DG auto start signal INOP 8. Div II SAT secondary winding undervoltage LOOP signal INOP.
7	Spare	Spare
8	Spare	Spare
9	Deluge Valve Control Station, Main Transformers.	1. Deluge system for Main Transformers A & C will not auto initiate. 2. Local bell and horn in Transformer yard power lost.
10	Deluge Valve Control Station, Main Transformer and UAT	1. Deluge system for Main Transformer B and UAT will not auto initiate.
11	4KV Switchgear Bus 2B Alternate Control Power	1. Automatic Bus Transfer, normal is Panel 10A, ckt. 3.
12	H2 and Stator Cooling Control Power	1. Loss of Turbine Protection circuitry associated with Stator Water. 2. Receive annunciator 1-UA-02-5-7. 3. Loss of standby pump auto start on low pressure. Backup low-pressure switch will still function. 4. Local alarm bell will sound. (Can be cut out locally with cabinet switch)
13	Spare	Spare

PANEL: 9A Reference Drawing: LL-30024-12	LOCATION: Unit 1 Turbine Building, 20', Switch gear area	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: Switchboard 1A (Mechanical Interlock)

CKT #	LOAD	EFFECT
14	Deluge Valve Control Station, MG Set Room.	1. Deluge system for MG Sets 1A and 1B will not auto initiate. 2. Loss of Power to local bell and horn. 3. Manual Pull Station will not function.
15	Spare	Spare
16	Deluge Valve Control Station, RFP Rooms	1. Deluge system for RFP 1A and 1B will not auto initiate. 2. Loss of Power to local bell and horn. 3. Manual Pull Station will not function.
17	Spare	Spare
18	Deluge Valve Control Station, Caswell Beach Transformer and SAT	1. Deluge system for Caswell Beach Transformer and SAT will not auto initiate.
19	Spare	Spare
20	Deluge Valve Control Station, Hydrogen Seal Oil Skid and Main Lube Oil Room	1. Deluge system for H2 Seal Oil skid and Main Lube Oil room will not auto initiate. 2. Loss of Power to local bell and horn. 3. Manual Pull Stations will not function.
21	4KV Switchgear Bus 1D Control Power	1. Loss of control power to 4KV loads on Bus 1D. 2. Loss of 4KV breaker operation, manual or automatic. 3. Loss of 4KV protective functions for Bus 1D and 4KV loads. 4. Bus 1D will not 'dead bus' auto transfer from the UAT to SAT. 5. Bus 1D cannot be manually transferred between the UAT and SAT. 6. Loss of breaker indication locally and on the RTGB. 7. Div I Loss of BOP Bus DG auto start signal INOP 8. Div I SAT secondary winding undervoltage LOOP signal INOP. 9. Loss of BOP Bus LOCA load shed circuit.

30. 262002G2.2.25 001

OP-52, in reference to Uninterruptible Power Supply (UPS), states that Technical Specifications require that no combination of more than two power conversion modules, consisting of either two lighting inverters or one lighting inverter and one plant UPS unit, shall be aligned to a B Division DC Bus.

Which ONE of the following describes the bases of this Technical Specification requirement?

This alignment would exceed the:

A. battery's amperage limit during normal rated thermal power operation.

B✓ battery's amperage limit during the first minute of a design basis accident.

C. battery charger's amperage limit during normal rated thermal power operation.

D. battery charger's amperage limit during the first minute of a design basis accident.

Feedback

Reference OP-52 and TS Bases for SR 3.8.7.2 rev. 30 page B 3.8.7-14 (Unit 2)
No combination of more than two power conversion modules, consisting of either two lighting inverters or one lighting inverter and one plant UPS unit, shall be aligned to a B Division DC Bus. This alignment would exceed the battery's amperage limit during the first minute of a design basis accident (Technical Specification SR 3.8.7.2). Aligned to Division II Bus B means that DC breaker is closed for inverter onto the DC bus regardless of ability of inverter to automatically transfer loads to DC supply.

Distractor Analysis
A and C incorrect - applicable safety analysis states that this alignment would exceed the battery's amperage limit during the first minute of a design basis accident not during normal ops.
D - incorrect - DBA assumes LOOP and INOP EDG therefore charger is deenergized. This alignment would exceed the battery's amperage limit during the first minute of a design basis accident.
B - correct

Notes

SYSTEM: 262002 Uninterruptable Power Supply (A.C./D.C.)

2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
(CFR: 43.2)
IMPORTANCE RO 2.5 SRO 3.7

This question matches the k/a in that it measures the RO's knowledge of TS bases behind an OP precaution.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.5	Facility Objective:	CLS-LP-052*005
Ref Req'd Y or N:	NO	Technical Ref.:	TS AND OP-52
? Cognitive Level:	M OR FK	? Source:	NEW

3.0 PRECAUTIONS AND LIMITATIONS

3.1 *MANUAL BYPASS SWITCH, S1*, should **NOT** be used to transfer UPS load from the Alternate source to the inverter while the static switch is in the inverter position. This could cause damage to the inverters.

3.2 The following Technical Specification requirement shall be observed for the 120 VAC Electrical System:

1. Section 3.8 — Electrical Power Systems

3.3 No combination of more than two power conversion modules, consisting of either two lighting inverters or one lighting inverter and one plant UPS unit, shall be aligned to a B Division DC Bus. This alignment would exceed the battery's amperage limit during the first minute of a design basis accident (Technical Specification SR 3.8.7.2). Aligned to Division II Bus B means that DC breaker is closed for inverter onto the DC bus regardless of ability of inverter to automatically transfer loads to DC supply.

3.4 Both plant UPS Units, 1A and 1B, should have their space heater *ON* when the units are deenergized to protect the solid state components from moisture.

3.5 The UPS Unit shall have its power supplied from the Alternate source prior to operating the *MANUAL BYPASS SWITCH* to prevent inverter damage, unless the Alternate source is unavailable.

3.6 Loss of the UPS System can result in the loss or malfunction of various systems or components. Refer to 0OI-50.5, 120V UPS Bus 1-1A and Bus 2-2A Electrical Load List, for UPS System load descriptions and the effects on loss of power.

31. 262002K315 001

Unit Two (2) is operating at rated power.

Which ONE of the following describes the effect that a loss of the Uninteruptible Power Supply (UPS) will have on Main Turbine Operation?

Main Turbine:

A. Supervisory Instrumentation (TSI) will be disabled.

B. Trip logic will be disabled due to loss of EHC DC power.

C✓ High Reactor Water Level trip circuit "A" will fail in the tripped condition.

D. Lube Oil temperature control valve (TCC-TV-615) will stroke full CLOSED resulting in maximum cooling of turbine lube oil.

Feedback

Reference 0OI-50.5 various circuits
Distractor Analysis
A - incorrect - TSI only disabled on Unit One
B - incorrect - EHC DC will have backup from PMG with turbine online
C - correct
D - TCV will stroke OPEN for max. cooling of lube oil not CLOSED

Notes

SYSTEM: 262002 Uninterruptable Power Supply (A.C./D.C.)
K3. Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following:
(CFR: 41.7 / 45.4)
K3.15 Main turbine operation: Plant-Specific 2.6 2.7

This question matches the k/a in that it measures the RO's knowledge of a loss of UPS on Main Turbine operation which is different between units.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.6	Facility Objective:	CLS-LP-052*008
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-50.5
? Cognitive Level:	C/A	? Source:	NEW

120V UPS Distribution Panel 1-1A Load Summary

Load: 120V UPS Distribution Panel 1-V7A (HG8)
Location: Control Building 49' NE
Drawing Reference: F-90098
Upstream Power Source: **120V UPS Distribution Panel 1-1A-UPS**

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
2	RTGB Benchboard 1-H12-P603: Digital Feedwater Control System (DFCS) Panel Display Stations (PDS) for RFPT 1A, 1B Manual/Auto Control 1-C32-SIC-R601A, R601B, RFPT Master Control 1-C32-SIC-R600, and FW S/U Level Control Valve 1-FW-LIC-3269; OTEK Digital Display Meters, 1-RFA(B)-SI-7325(6); Recorders 1-C32-FR-R607, LPR-R608, PR-R609 and 1-C51-NR/NI-R602, R603A, R603B, R603C, R603D (TS 3.3.1.1, 3.3.3.1, 3.4.10)	If the PDS was selected to AUTO, DFCS will continue to control level automatically, and the level setpoint cannot be adjusted; if the PDS was in MANUAL, the last processed signal from the PDS will be used to maintain speed control/valve position. Will lose RFPA(B) OTEK digital display meters. Will lose recorders for feed/steam flow, vessel level/pressure, and SRM/IRM/APRM/RBM. Will lose APRM and RBM Operator Display Assemblies (ODAs).
3	Unit 1 RTGB Pressure Indications for Turbine Oil, EHC, Seal Oil, Steam Chest, Sealing Steam, and Steam Packing Exhauster Vacuum	Loss of 1-XU-3 indicators TO-PI-EPT-1, TO-PI-EPT-2, EHC-PI-EPT-6, SO-PI-EPT-7, SO-PI-EPT-8; loss of XU-2 indicators MS-PI-EPT-3, OG-PI-EPT-4, OG-PI-EPT-9
6	Unit 1 Turbine Supervisory Instrument (TSI) Panel	The TSI turbine trip circuit will be disabled.
7	1-H12-P603 Rod Display Fan Motor / Power Transfer Unit	Control rod matrix cooling fan will continue operating if power is available from Panel 1AB, circuit #9.
8	RTGB PM4 Plug Mold For Recorders on 1-XU-4	Loss of TSI-TXR-638, TSI-XR-640, TSI-TR-642, GEN-WMR-760, SY-VMR-759, and watt telemeter
10	Unit 1 Offgas / SJAE Train A & B Main Steam Pressure Control	UA-39-5-2, UA-39-6-1, UA-40-5-2 and UA-40-6-1 will be disabled

120V UPS Distribution Panel 2-2A Load Summary

Load: 120V UPS Distribution Panel 2-V10A (HG9)
Location: Control Building 49' SW
Drawing Reference: F-03027
Upstream Power Source: **120V UPS Distribution Panel 2-2A-UPS**

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
1	5VDC and 28VDC Power Supplies for Control Rod Position Indication System (RPIS) Cabinet 2-H12-P615; Rod Worth Minimizer (RWM) NUMAC Drawer 2-C11-CNV-5516, 2-H12-P607 (TS 3.1.3.1, 3.3.2.1)	Loss of rod position indication on four-rod display panel and full core display, loss of RWM, cannot move control rods, A-05-5-2 will alarm.
2	EHC Control Pnl 2-EHC-XY-644 (Bay F): * +30 VDC Power Supply * -22 VDC Power Supply * +24 VDC Power Supply Chest Warming Logic Cabinet Fan Motors, Bays A through E Alarm & Trip Logic Load Reference Logic Valve Test Logic Isolation Amplifier Circuit Monitor Panel Meter Relays Transformers T10, T11 for Operation of 60 Hz Valve Position Demodulator	If loss of power occurs while the main turbine is in operation, redundant power is available from the permanent magnet generator (PMG) for the loads designated with *, and the main turbine will remain on line. Other loads will lose power. If loss of power occurs while the main turbine is tripped, the bypass valves will not open and instrumentation on XU-1 will lose power.
3	Unit 2 FW Control System: RFPT A & B/Main Turbine High Level Trip Circuit "A" MV/I converters for 2-C32-TE-N006A and 2-C32-TE-N006B Digital FWCS Rx Scram B Input Power supplies: 2-C32-ES-5782A & B (Digital FWCS) 2-C32-ES-5783A & B (Digital FWCS) 2-C32-ES-5784A & B (Digital FWCS) 2-C32-ES-5786A & B (Digital FWCS) 2-C32-K620 for 2-C32-PT-N007 (Turbine Steam Flow) and 2-C32-PT-N008 (Reactor Pressure)	High reactor level trip circuit "A" will fail in the tripped condition (high level trip logic will actuate if level "B" or "C" receives a trip signal) No effect from loss of digital FWCS power supplies if backup power is available from 125VDC Distribution Panel 4B, circuit #10. Will lose the following indicators: FW-TI-53, 54 (XU-2) C32-PR-R609 (Rx pressure only) and ERFIS/process computer inputs from C32-PT-N007, C32-PT-N008. Will disable the C71-K21B input to the DFWCS logic from scram trip system "B" (DFWCS will not detect a scram signal on trip system "B"). DFWCS level setdown to 170" will also be disabled.
4	Unit 2 Main Turbine Master Trip Alarm Circuit	UA-23-1-4 will be disabled.

120V UPS Distribution Panel 2-2A Load Summary

Load: 120V UPS Distribution Panel 2-V8A (HG8)
Location: Control Building 49' SE
Drawing Reference: F-03027
Upstream Power Source: **120V UPS Distribution Panel 2-2A-UPS**

CKT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
11	2-COD-FY-44, 2-COD-FY-45, 2-FW-FY-648, 2-TCC-TT-615 (as per F-03027, Rev 54) NOTE: F-09023, sheet 2 indicates that 2-TCC-TT-615 is terminated to TB36-4,5 in 2-XU-9, which is powered from UPS Panel 2-V10A, circuit #11. ESR 94-00738 has been generated to resolve this discrepancy.	RTGB indicators FW-FI-648-1, COD-FI-44-1, and COD-FI-45-1 on XU-2 will fail; turbine lube oil temperature control valve TCC-TV-615 will stroke full open when TCC-TIC-615 on XU-2 loses power.
12	2-C12-PS6 (+28 VDC Power Supply), 2-C12-PS7 (-28 VDC Power Supply), 2-C12-PS8 (+28 VDC Power Supply) for Rod Select Pushbuttons and Lights; 2-C12-J601 (5 VDC and 6 VDC Power Supplies) for RWM Buffer (TS 3.1.3, 3.3.2.1)	Reactor manual control system (RMCS) will be disabled, cannot select or position control rods.
13	Unit 2 RTGB Pressure Indications for Turbine Oil, EHC, and Seal Oil	Loss of XU-3 indicators TO-PI-EPT-1, TO-PI-EPT-2, EHC-PI-EPT-6, SO-PI-EPT-7, and SO-PI-EPT-8.

2.0 AUTOMATIC ACTIONS

2.1 **IF** Stack Radiation Monitor power is lost, **THEN** the following actions occur for Unit 1 and Unit 2:

- Reactor Building Ventilation isolation ☐
- SBGTS auto start ☐
- Group 6 Isolation ☐

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

None

3.2 Supplementary Actions

3.2.1 **IF** RPIS or RMCS is lost, **THEN REFER** to 0AOP-02.0. ☐

3.2.2 **ATTEMPT** to re-energize UPS Bus from the available inverter or reserve bus in accordance with 1(2)OP-52. ☐

3.2.3 **IF** Stack Radiation Monitor power supply was lost, **THEN TRANSFER** Stack Radiation Monitor power in accordance with 1(2)OP-52. ☐

3.2.4 **MONITOR** nuclear instrumentation. ☐

3.2.5 **IF** necessary, **THEN THROTTLE CLOSED** *MAIN TURBINE OIL COOLERS TBCCW OUTLET ISOLATION VALVE, TCC-V117,* to maintain normal turbine lube oil temperature. ☐

TCC-TV-615 STROKE S OPEN FOR MAX COOL.

3.2.6 **IF** a total loss of UPS occurred on Unit 2 **OR** loss of UPS Panel 2-V6A, **THEN NOTIFY** Security for implementation of contingency actions as required by the Site Security Plan. ☐

3.2.7 **REFER** to 0OI-50.5 for a list of affected equipment. ☐

3.2.8 **WHEN** UPS Bus power has been restored, **THEN RETURN** the Main Turbine Oil System to normal, if necessary. ☐

32. 263000K602 001

Unit Two (2) has experienced a Loss of Coolant Accident (LOCA) and has lost the normal feed to E3 from BOP Bus 2D. All systems respond per design.

Which ONE of the following describes the status of the Unit Two (2) Div I D.C. Electrical Distribution System?

Div I 24 Vdc Battery Chargers:

A.✓ and Div I 125Vdc Battery Chargers will all be energized.

B. and Div I 125Vdc Battery Chargers will all be deenergized.

C. will be energized but Div I 125Vdc Battery Chargers will be deenergized.

D. will be deenergized but Div I 125Vdc Battery Chargers will be energized.

Feedback

Reference SD-51 page 9 of rev. 3 - Both 24Vdc and 125Vdc chargers are E-bused powered ensuring DC system reliability.

LOCA added to the stem to ensure that LOCA load stripping is considered in answer to make question more comprehensive. LOCA load stripping does not include 480Vac E-buses.

Distractor Analysis
B, C, and D - incorrect as all chargers in question will be energized by E-Bus that is powered from the EDG.
A - correct - E3 powered from EDG#3, E7 from E3 MCC-2CA and DP-2A will both be energized.

Notes

SYSTEM: 263000 D.C. Electrical Distribution
K6. Knowledge of the effect that a loss or malfunction of the following will have on the D.C. ELECTRICAL DISTRIBUTION :
(CFR: 41.7 / 45.7)
K6.01 A.C. electrical distribution 3.2 3.5

This question matches the k/a in that it measures the RO's knowledge of the effect that a loss of AC will have on DC power. Requires RO to understand that both 24Vdc and 125Vdc chargers are E-bus powered and not associated with load stripping.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.2	Facility Objective:	CLS-LP-051*1A AND B
Ref Req'd Y or N:	NO	Technical Ref.:	SD-51
? Cognitive Level:	C/A	? Source:	NEW

Each unit contains a 24/48 VDC Distribution system which consists of two separate divisions. Each division contains two 24 VDC batteries wired in a series/parallel arrangement to it's distribution panel thus allowing for 24 VDC or 48 VDC loading. Each battery contains it's own independent charger. This system does not contain ground detection devices. The major loads supplied by this system are startup nuclear instrumentation and process radiation monitoring instrumentation. This accounts for 4 more of the 10 independent systems mentioned above.

Each of the 125 VDC and 24 VDC battery chargers discussed above are provided with AC power via their respective division Emergency Bus. This ensures that the battery charger has multiple sources of AC power thus ensuring DC system reliability.

There are 4 independently ventilated battery rooms located in the Control Building. Each room contains a separate division of both the 125/250 VDC and the 24/48 VDC Distribution systems as shown in Figure 51-1.

The Caswell Beach Pumping Station contains it's own independent 125 VDC distribution system consisting of a battery, a charger and a distribution panel. The major loads supplied by this system are breaker control and relay logic for both units 1 and 2. This system does not contain any ground detection devices.

The last DC distribution system is the Caswell Beach Microwave system DC power supply. This system consists of a battery and a charger and is connected directly to the microwave system master panel. The Microwave Telemetering System provides control and indication of Caswell Beach Station components via airwave transmission. This system does not contain any ground detection devices.

Although some construction differences do exist, all of the battery systems consist of individual cells rated at approximately 2 volts each. The exception is the Caswell Beach microwave battery system which contain 4 multi cell batteries rated at approximately 12 volts each.

33. 264000K106 001

Unit Two (2) was operating at power when a reactor scram occurred. Current plant status:

RPV water level	+175 inches
RPV pressure	900 psig
Drywell pressure	2.2 psig
Main generator	Tripped on reverse power
Bus 2C	De-energized (failed to transfer, SAT breaker failed to close)
DG4	Tied to E4, in Auto
Bus E4	Undervoltage alarmed, but is now clear

Which ONE of the following identifies the automatic start signals that are currently present for DG4?

A. Loss of E Bus and LOOP only.

B. Loss of BOP Bus and LOOP only.

C. Loss of E Bus and Loss of BOP Bus only.

D✓ Loss of E Bus, Loss of BOP Bus and LOOP.

Feedback

Randomly selected from LOI systems Bank - LOI-CLS-LP-039-A*003 015
Reference SD-39 Rev. 3 pages 135-137
A, B, & C - incorrect since Loss of E Bus signal present due to master/slave & tie breakers open with DG in Auto (even though Bus UV is clear). Loss of BOP Bus present due to Bus 2C undervoltage with UAT UV. LOOP signal present due to reverse power trip and the resultant primary L/O, which is a Div I LOOP signal.
D correct

Notes

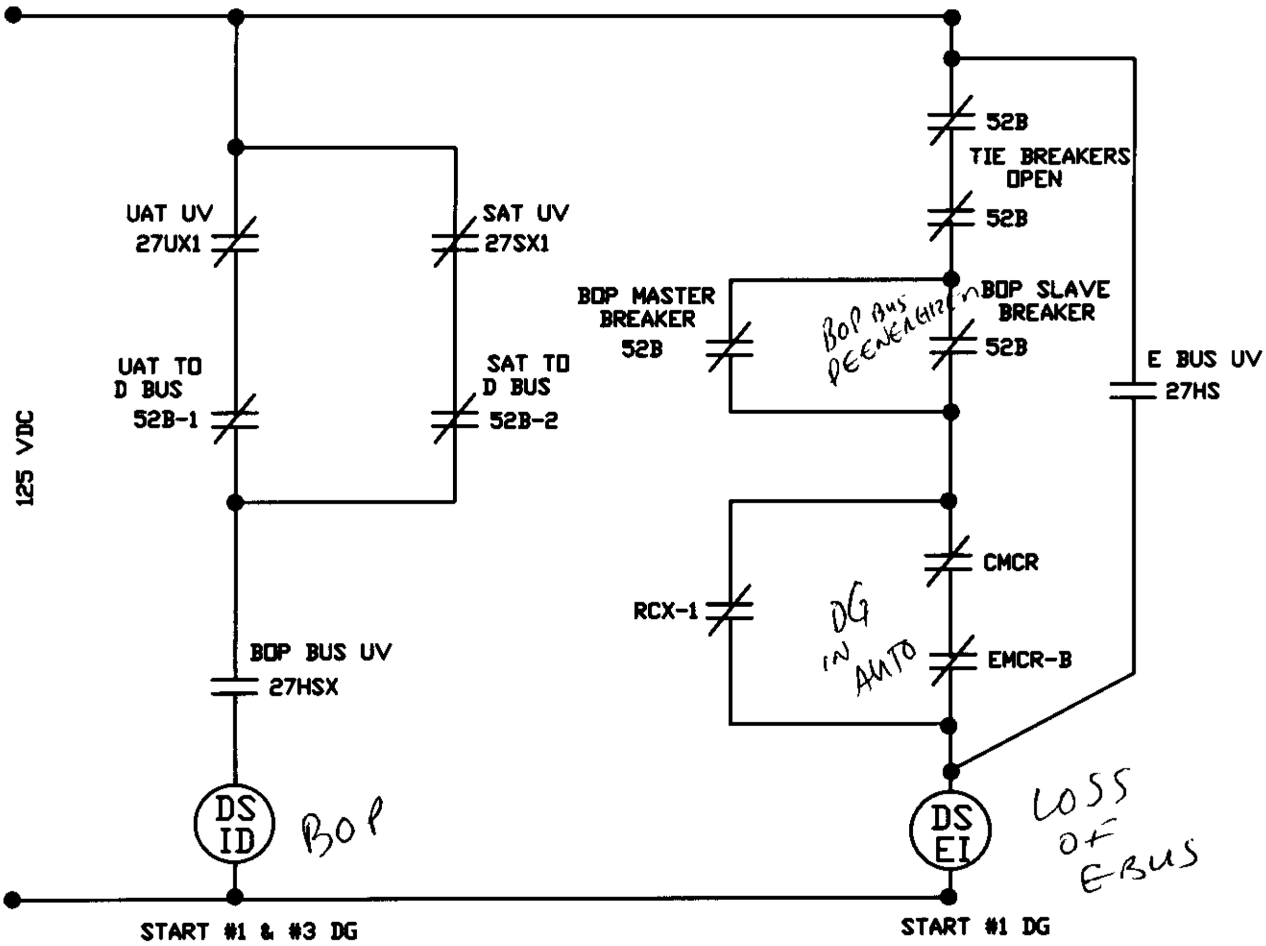
SYSTEM: 264000 Emergency Generators (Diesel/Jet)
K1. Knowledge of the physical connections and/or cause effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following:
(CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.06 Starting system 3.2 3.2

This question matches the k/a in that it measures the ROs knowledge of all the EDG starting logic.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.2	Facility Objective:	CLS-LP-039*003
Ref Req'd Y or N:	NO	Technical Ref.:	SD-39
? Cognitive Level:	C/A	? Source:	BANK LOI

FIGURE 39-13
Electrical System Fault Divisional and Individual Diesel Start Logic

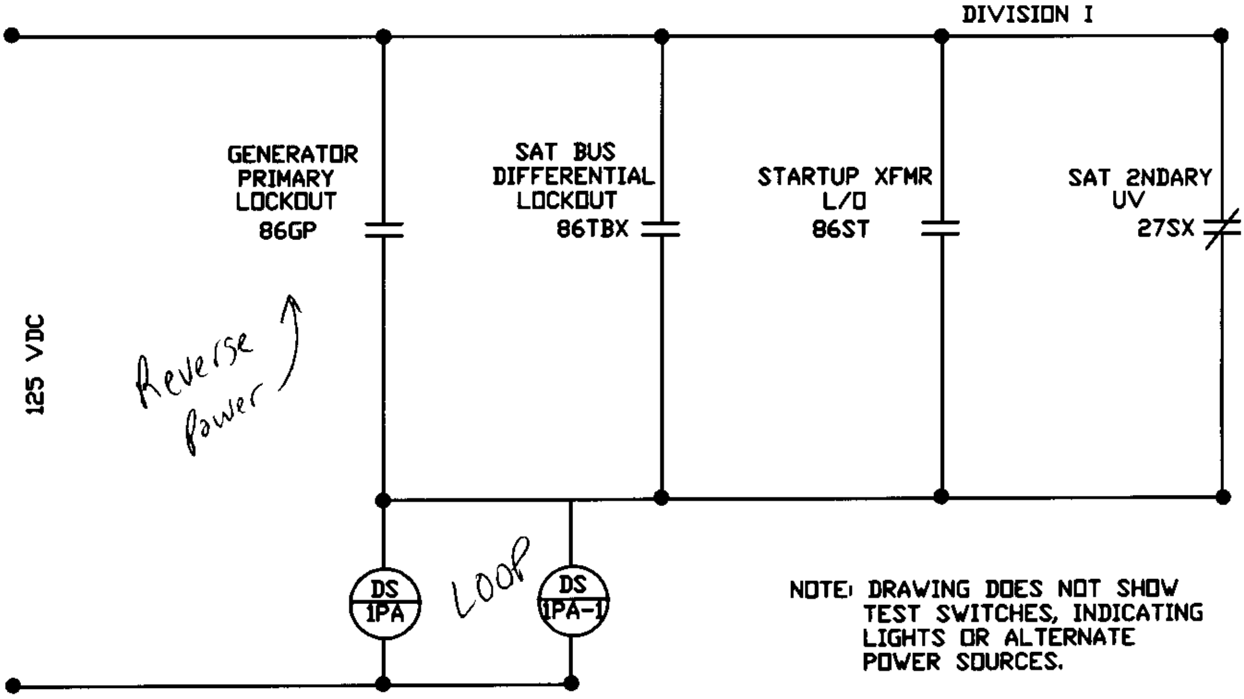


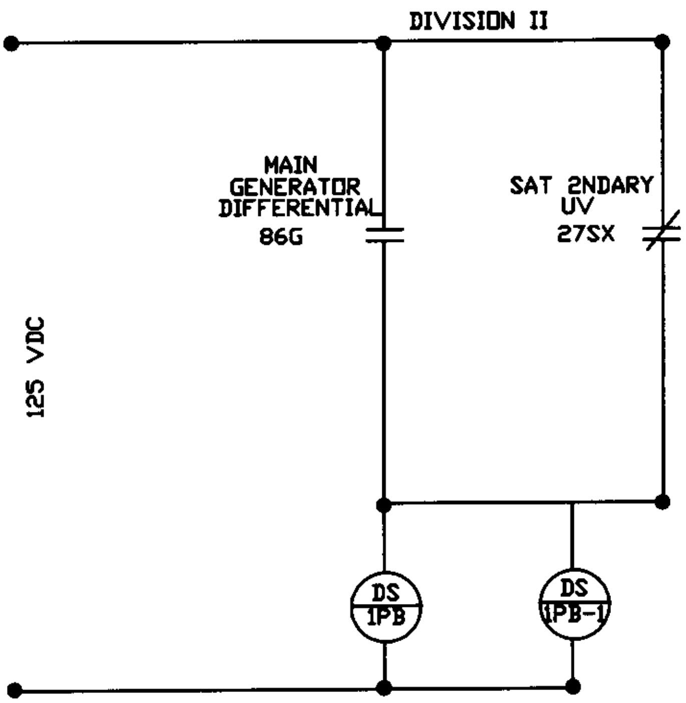
REF. F-9116

NOTE: INDICATING LIGHTS AND TEST SWITCHES NOT SHOWN

FIGURE 39-12

Electrical System Fault Logic Circuit Units 1 and 2





34. 264000K601 001

Emergency Diesel Generator #2 (DG2) is running and tied to E2 in response to a Loss of Off-Site Power. A loss of 100 psig control air to the diesel engine occurs.

Which ONE of the following describes how DG2 will respond to the loss of 100 psig control air?

A.✓ Fuel rack repositions to the no fuel position.

B. Governor transfers to the mechanical backup.

C. Engine lockout relay actuates on low air pressure.

D. Engine lockout relay actuates on engine overspeed.

Feedback

REFERENCE SD-39
100 psig control air required to position fuel racks from the no-fuel position.

RANDOMLY SELECTED FROM BANK LOI-CLS-LP-039-A*004 001

DISTRACTOR ANALYSIS
A. CORRECT
B. INCORRECT - Governor is hydraulic
C. INCORRECT - Not a lockout signal
D. INCORRECT - Will not overspeed

Notes

SYSTEM: 264000 Emergency Generators (Diesel/Jet)
K6. Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) :
(CFR: 41.7 / 45.7)
K6.01 Starting air 3.8 3.9

This question matches the k/a in that it measures the ROs knowledge of the effect that a loss of air has on EDG operation.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 3.8	Facility Objective:	CLS-LP-039*004
Ref Req'd Y or N:	NO	Technical Ref.:	SD-39
? Cognitive Level:	M OR FK	? Source:	BANK LOI

Initial rolling of the EDG prior to combustion pressure being reached is accomplished by the starting air system. On an EDG start signal high pressure air (250 psig) is directed to each cylinder in engine firing sequence through individual starting air valves. The starting air valves are sequentially controlled by a starting air distributor that is driven by the engine camshaft. Each air start valve is spring loaded closed. The starting air distributor ports air onto the top of a piston integral to the air start valve which opens the valve. Valve opening admits 250 psig air into the Diesel cylinders to rotate the engine. Annunciators UA-19-22, DG-1-4 LOW AIR START PRESSURE is available to warn the control room operator when pressure drops below 170 psig.

High pressure starting air (250 psig) is also supplied to the components associated with the engine control circuits during Diesel start until the air start solenoid valves close which removes the starting air. 250 psig starting air from the right starting air header is supplied to the boost cylinder. The boost cylinder provides pressure to the governor oil system until the oil pumps, internal to the governor, that are driven off the engine can deliver sufficient oil pressure.

On an engine start, 250 psig starting air is reduced to 125 psig. 125 psig air supplied from each starting air header is applied to the Overspeed Start Emergency Boost Cylinder and to the Fuel Rack Limit Cylinder. The Overspeed Start Emergency Boost Cylinder pressurizes engine "control" oil. The pressurized oil is directed to the Overspeed Pressure Switch and the pilot of the Overspeed Shutdown Valve. Pressurization continues until the start solenoid valves close. The pressurized "control" oil actuates the pilot of the Overspeed Shutdown Valve to provide 100 psig control air to the fuel rack Run/Stop Cylinders. Due to the long length of supply tubing, the Overspeed Start Emergency Boost Cylinder only provides partial pressurization of the Overspeed Pressure Switch. Failure to pressurize the switch in 45 seconds will result in initiation of an engine lockout signal and subsequent engine trip. As a result, actuation of the pressure switch is inhibited for 45 seconds on a EDG start sequence to allow sufficient time for the engine lube oil system to pressurize this pressure switch. In an overspeed condition, the Overspeed Trip Valve dumps "control" oil back to the engine sump causing the Overspeed Pressure Switch to actuate and initiate an engine lockout. The Fuel Rack Limit Cylinder prevents overfueling of the engine on a start sequence. Overfueling washes the lube oil from the cylinder liner walls which can lead to cylinder liner scuffing.

Engine control air is supplied from each starting air receiver to the fuel rack run/stop cylinders. The control air is filtered, dried and reduced to 100 psig. Moisture indicators in the control air supply downstream of the filter dryers are normally green and turn yellow on excessive moisture. If the standby dryer/filter must be placed in service due to indications of clogging or excessive moisture, the standby filter dryer must be placed in service prior to removing the in service filter dryer to prevent loss of control air supply to the diesel engine. This air is applied during an Engine start through the run control solenoid valve to the fuel rack stop/run cylinders. These cylinders either place the fuel racks at the "no fuel" position or permit governor control of the fuel racks during engine operation.

2.5 Diesel Engine Cooling System (Figure 39-7)

Part of the heat developed during combustion is transmitted from the gases to the cylinder walls. Additional high local temperature is developed by friction between the piston rings and cylinder walls. This heat is removed primarily by the engine jacket water and, to a lesser extent, by the lube oil.

The jacket water cooling system is a closed loop system which removes most of the heat generated by the engine during operation by cooling engine components and the lube oil circulating through the engine. At normal rated load conditions, the engine jacket water temperature will be about 170°F (165°F-175°F), and the lube oil temperature will be about 165°F (160°F-170°F).

Jacket water flows through the engine around the cylinder liners in a jacket, around the cylinder heads and turbocharger jackets and exits through flow orifices which control the flow of this high velocity water through the engine. Water exiting the cooled components travels to headers containing temperature control valves, jacket water cooler, lube oil cooler and turbocharger intercooler.

Motive force for this flowpath is provided by either the engine driven jacket water or auxiliary jacket water pump. The pumps discharge into a common header for redistribution to the engine components. The water itself is chemically treated with corrosion inhibitor to protect components served by this cooling system. Jacket water chemistry is monitored periodically and each time makeup water is added to ensure a proper concentration of water treatment chemicals. Proper jacket water chemistry minimizes corrosion to ensure maximum heat transfer capability of the jacket water cooling system and interfacing EDG Service Water System. Poor heat transfer capability results in increased engine operating temperatures, lower intercooler performance, reduced engine performance, and eventual cooling system component damage due to corrosion.

35. 288000K502 001

High wind speeds cause Reactor Building static pressure to increase to +1.1 inches of water on Unit Two (2).

Which ONE of the following describes the operational implications of this condition as it applies to Reactor Building Ventilation?

Reactor Building ventilation:

A. supply and exhaust fans will trip requiring a manual start of the SBGT system to maintain Reactor Building pressure negative.

B. supply and exhaust fans will trip and the SBGT system will AUTO start to maintain Reactor Building pressure negative.

C. exhaust fan vortex dampers throttle OPEN but a Supply/Exhaust fan configuration change or manual start of the SBGT system may be required to maintain Reactor Building pressure negative.

D✓ supply fan vortex dampers throttle CLOSED but a Supply/Exhaust fan configuration change or manual start of the SBGT system may be required to maintain Reactor Building pressure negative.

Feedback

REFERENCE - APP UA-12 3-3 Rev. 23 page 32 and UA-05 6-7 Rev. 37page 75

DISTRACTOR ANALYSIS

A. INCORRECT - RB vent supply and exhaust fans trip at +/- 4.0 inches of water. Recent plant modification. This would have been correct previous setpoint was 1.0 inches of water

B. INCORRECT - RB vent supply and exhaust fans trip at +/- 4.0 inches of water. Recent plant mod. SBGT does not AUTO start on high RB press.

C. INCORRECT - RB vent exhaust fan vortex dampers aredisabled in the full open position and therefore do not modulate to control D/P

D. CORRECT- RB vent supply vortex dampers will throttle closed but may not have adequate capacity to lower pressure this high thus requiring change of fan configuration or SBGT start.

Notes

SYSTEM: 288000 Plant Ventilation Systems

K5. Knowledge of the operational implications of the following concepts as they apply to PLANT VENTILATION SYSTEMS :

(CFR: 41.7 / 45.4)

K5.02 Differential pressure control 3.2 3.4

This question matches the k/a in that it measures the RO's knowledge of operational implications as it applies to the RB ventilation system and response to control D/P.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 3.2	Facility Objective:	CLS-LP-037-B*004
Ref Req'd Y or N:	NO	Technical Ref.:	SD-37.1
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

RX BLDG DIFF PRESS HIGH/LOW
(Reactor Building Differential Pressure High/Low)

AUTO ACTIONS

1. Reactor Building supply and exhaust fans trip.

CAUSE

1. High or low differential pressure between the Reactor Building and atmospheric pressure.
2. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, 2-VA-PI-1297, on RTGB Panel XU-3.

ACTIONS

1. If secondary containment integrity is required and differential pressure is low, enter 0EOP-03-SCCP, Secondary Containment Control, and execute concurrently with this procedure.
2. Inform E&RC Chemistry Reactor Building Ventilation is not in service.
3. Verify that the valve lineup is correct per 2OP-37.1, Reactor Building Heating and Ventilation System.
4. Start up the system per Section 5.1 of 2OP-37.1.
5. If a circuit malfunction is suspected, ensure that a W/O is submitted.

DEVICE/SETPOINTS

Reactor Building Static Press Hi Switch 2-VA-PDS-3779	+4.0 inches of water
Reactor Building Static Press Lo Switch 2-VA-PDS-3780	-4.0 inches of water

POSSIBLE PLANT EFFECTS

NONE

REFERENCES

1. LL-9354 - 31
2. 2OP-37.1, Reactor Building Heating and Ventilation System
3. 0EOP-03-SCCP

RX BLDG STATIC PRESS DIFF-LOW

AUTO ACTIONS

NONE

CAUSE

1. Low negative pressure differential in the Reactor Building.
2. High wind speeds
3. Circuit malfunction.

OBSERVATIONS

1. Reactor Building Static Pressure Indicator, VA-PI-1297, located on RTGB Panel XU3.

ACTIONS

1. If the low differential pressure is due to high wind speeds then refer to OP-37.1 for Supply/Exhaust fan configuration.
2. Start the standby Reactor Building exhaust fan.
3. If a standby exhaust fan is not available, stop a Reactor Building supply fan.
4. If negative pressure cannot be maintained, start SBGT System per OP-10, Standby Gas Treatment System.
5. If secondary containment integrity is required, and Reactor Building pressure cannot be maintained negative, enter EOP-03-SCCP, Secondary Containment Control.
6. Notify E&RC Counting Room that reactor building ventilation has been secured.
7. If positive pressure in the Reactor Building is indicated, (VA-PI-1297 on XU-3), trip all Reactor Building supply fans.

DEVICE/SETPOINTS

Pressure Differential Switch YA-PDS-1508	0.1 inches water

POSSIBLE PLANT EFFECTS

1. If static pressure in the Reactor Building increases to 4 inches water, the Reactor Building supply and exhaust fans trip.

REFERENCES

1. 9527-LL-9354 - 20
2. OP-10, Standby Gas Treatment System
3. OP-37.1, Reactor Building Heating and Ventilation System Operating Procedure
4. EOP-03-SCCP

36. 290003K601 001

Control Building ventilation initiates automatically in the radiation mode with the controls positioned as follows:

CBEAF A	STBY
CBEAF B	PREF

2B Emergency Recirculation Supply Fan starts but several minutes later MCC-2CB deenergizes due to an electrical fault.

Which ONE of the following describes how the 2A Emergency Recirculation Supply Fan will respond to the loss of MCC-2CB?

The 2A Emergency Recirculation Supply Fan will automatically start:

A.✓ as soon as the 2B fan deenergizes.

B. ten (10) seconds after the 2B fan deenergizes.

C. immediately only if its' control switch is placed in the PREF position.

D. ten (10) seconds after its' control switch is placed in the PREF position.

Feedback

MODIFIED LOI SYSTEMS BANK QUESTION LOI-CLS-LP-037-A*004 1
ORIGINAL QUESTION
Control Building ventilation initiates automatically in the radiation mode with the controls positioned as follows:

CBEAF A STBY
CBEAF B PREF

How will the 2A Emergency Recirculation Supply Fan respond if the 2B Emergency Recirculation Supply Fan started and then immediately tripped?

The 2A Emergency Recirculation Supply Fan will automatically start:
A. as soon as the 2B fan trips.
B. ten (10) seconds after the initiation signal.
C. only if its' control switch is placed in the PREF position.
D. ten (10) seconds after its' control switch is placed in the PREF position.

Answer: B

REFERENCE - SD-37 pages 27, 58 and 59 The STBY fan will start immediately after PREF fan loses power. 10 sec delay applies ONLY after initiating signal i.e. a tripped fan upon initiation.

DISTRACTOR ANALYSIS
A. CORRECT
B, C, & D. INCORRECT -AUTO actions state that the STBY fan will start immediately after PREF fan loses power. 10 sec delay after initiating signal only applies to a tripped fan upon initiation.

Notes

SYSTEM: 290003 Control Room HVAC
K6. Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC :
(CFR: 41.7 / 45.7)
K6.01 Electrical power 2.7 2.9

This question matches the k/a in that it measures the RO's knowledge of how a loss of electrical power will effect CBHVAC operation during an automatic initiation.

Categories

Tier:	TIER 2	Group:	GROUP 2
Importance Rating:	RO 2.7	Facility Objective:	CLS-LP-037*004
Ref Req'd Y or N:	NO	Technical Ref.:	APP UA-05 5-6
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

3.2.2 Emergency Air Filtering Trains

The emergency air filtering (CREV) trains may be operated in the automatic or manual mode. Each filter train is provided with a three position, STBY-PREF-ON, control switch (2-VA-CS-915A and 2-VA-CS-915B). Both control switches are located on the Unit 2 XU-3 Panel, with status indicating lights located on each Unit's XU-3 Panel.

1. An automatic start signal is initiated by any of the following:

   a. Any one of three Area Radiation Monitors

      (1) Control Room (Channel 1) 1 mr/hr ± .05mr increasing
      (2) Control Building Ventilation Intake (Channel 2 or 3) 7 mr/hr ± .05mr increasing

   b. A combination of:

      (1) Ionization (smoke) detector or manual pull station in Zone C4 (Unit 2 Electronic Equipment Room)

      <u>AND</u>

      (2) Ionization (smoke) detector or manual pull station in Zone C5 (Unit 1 Electronic Equipment Room)

During normal operation, one filtering train control switch is selected to the PREF (preferred) position and the second train is selected to the STBY (standby) position. The initiation of an automatic start signal places the preferred filtering train in operation.

When a start signal is received, the inlet and outlet Emergency Air Dampers on the preferred filtering train open. In the fully open position, each damper actuates a limit switch to initiate the start of the emergency recirculation fan. If the fan fails to start or trips, a start signal for the standby filtering train is initiated after a 10 second time delay. The starting sequence is identical to that of the preferred train. An air flow switch closes when sufficient air flow is reached to indicate the unit is running. The train will continue to operate until it is secured by operator action from the RTGB.

FIGURE 37-5
CBEAF FAN 2A Logic

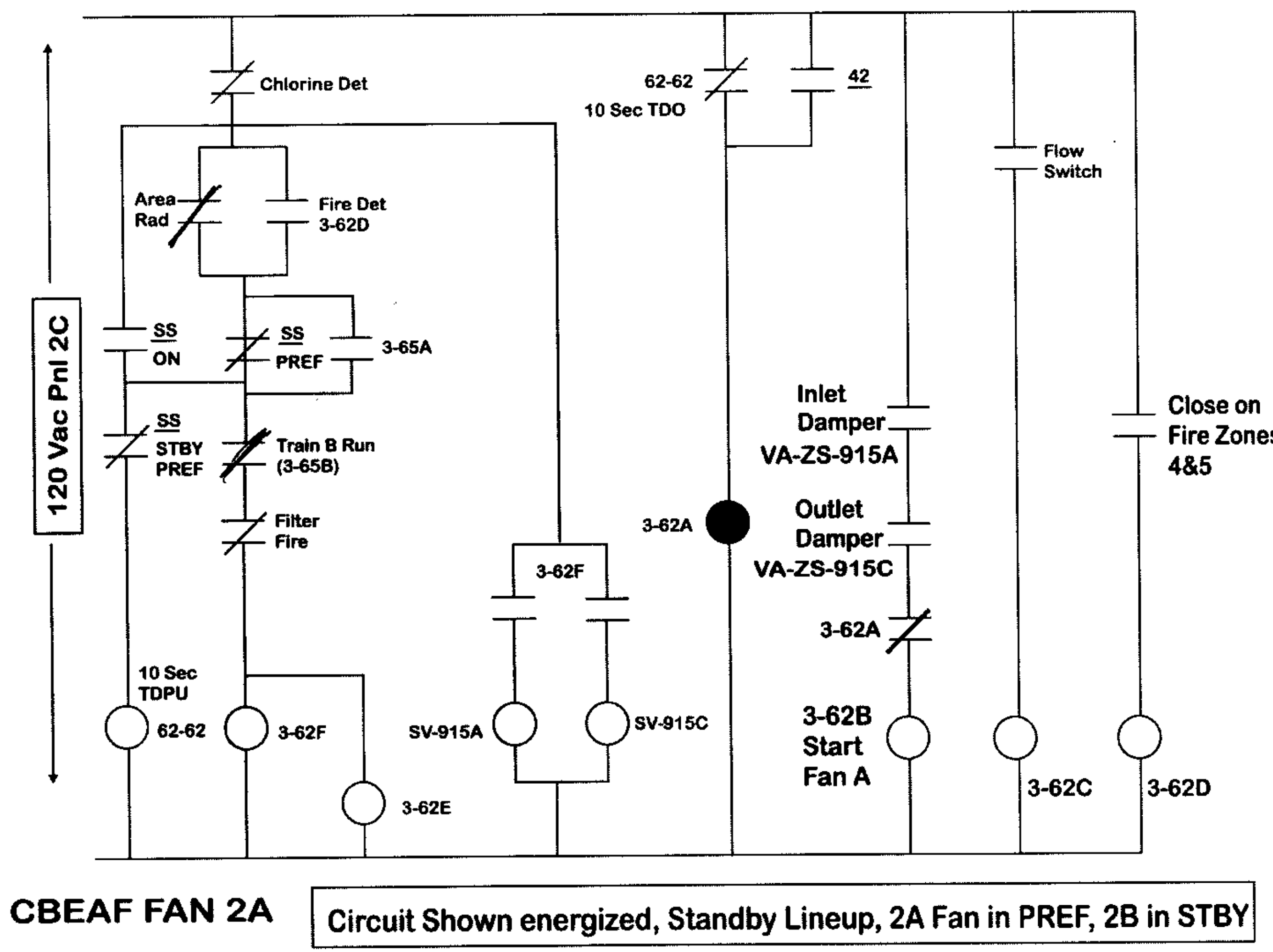
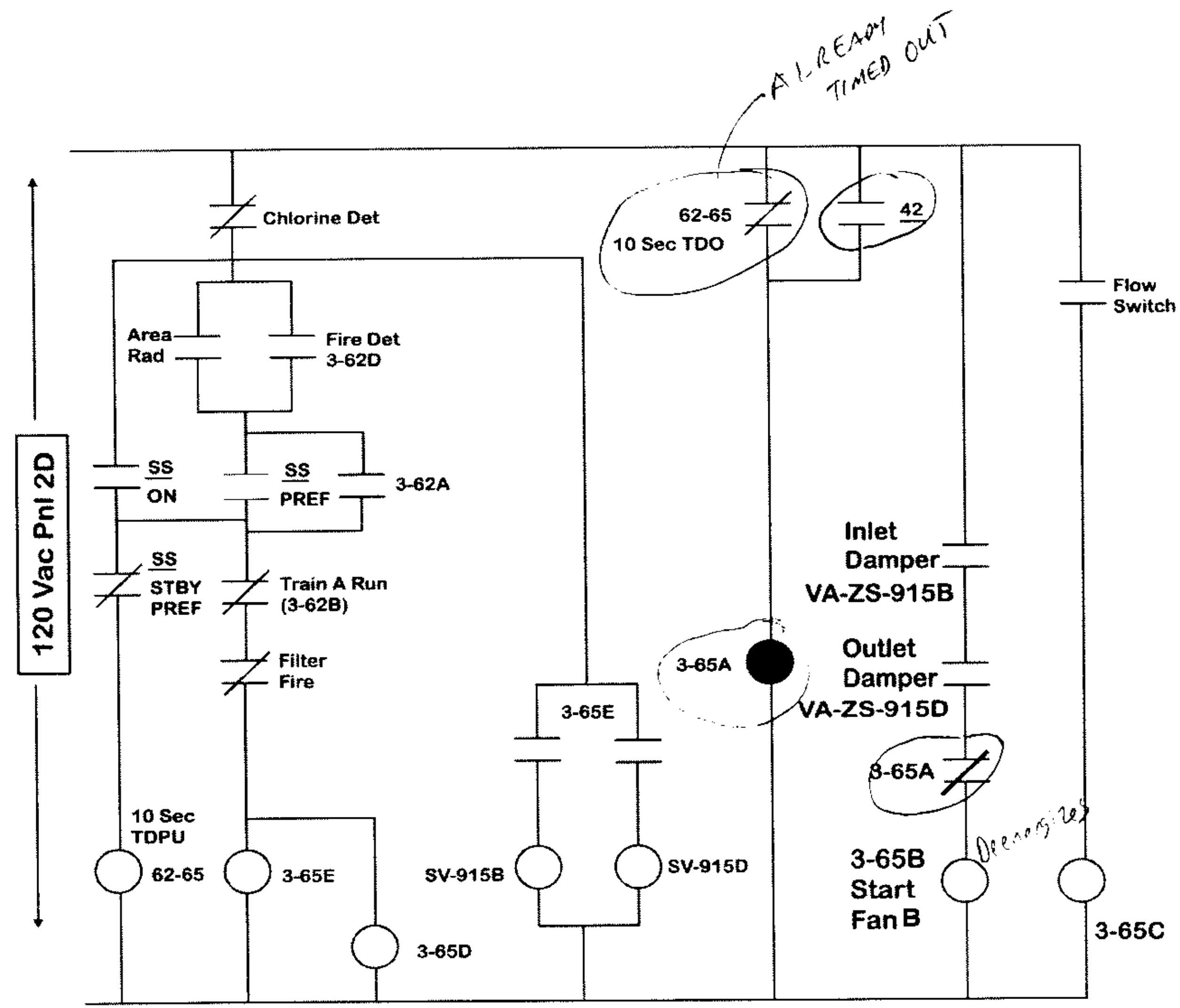


FIGURE 37-6
CBEAF FAN 2B Logic



CBEAF FAN 2B

Circuit Shown energized, Standby Lineup, 2A Fan in PREF, 2B in STBY

37. 295001AK101 001

Unit Two (2) had just been placed in Cold Shutdown when off-site power is lost. All group isolations occur as expected, including shutdown cooling.

The operators are executing AOP-15.0 to restore shutdown cooling, but are having difficulty opening inboard suction isolation valve (E11-F009). Reactor water level is being maintained 200"-220" for natural circulation.

Which ONE of the following parameters must be monitored for determination of a mode change to Hot Shutdown?

A.✓ Reactor vessel pressure.

B. Reactor bottom head temperature.

C. Reactor recirculation loop temperature.

D. RHR heat exchanger inlet temperature.

Feedback

RANDOMLY SELECTED BANK ITEM - LOI-CLS-LP-302-L*003 001

REFERENCE AOP-15 Rev. 16 page 4
Natural circulation cannot be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore cannot be utilized for vessel coolant temperature monitoring for indication of boiling. Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination. (AOP-15.0)

DISTRACTOR ANALYSIS
A. Correct - Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination.
B. and C. Incorrect - Natural circulation cannot be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore cannot be utilized for vessel coolant temperature monitoring for indication of boiling.
D. Incorrect - SDC isolated no flow through heat exchanger to provide adequate temperature indication

Notes

APE: 295001 Partial or Complete Loss of Forced Core Flow Circulation

AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :
(CFR: 41.8 to 41.10)
AK1.01 Natural circulation.................................. 3.5 3.6

This question matches the k/a in that it measures the RO's knowledge of natural circulation conditions following a loss of forced circulation. The operational implication being that a mode change can result and under natural circ. conditions the RO must know that reactor pressure is the only valid indication available to detect this heatup and mode change.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-302-L*003
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-15
? Cognitive Level:	C/A	? Source:	BANK LOI

3.0 OPERATOR ACTIONS

R22

CAUTION

Natural circulation can **NOT** be depended on to provide adequate flow through the bottom head region or the recirculation loops. The recirculation loop suction temperatures and bottom head temperatures therefore can **NOT** be utilized for vessel coolant temperature monitoring for indication of boiling. Under natural circulation conditions, reactor vessel pressure must be monitored for coolant temperature determination. If coolant temperature was initially less than 212°F, pressure must be closely monitored for indications of a trend of increasing pressure. If this trend is established, it must be assumed that 212°F has been exceeded, boiling is occurring, and a mode change has taken place.

3.2.4 **MONITOR** reactor coolant heatup/cooldown in accordance with 1(2)PT-01.7 for any unexpected trends. ☐

NOTE: If the time to boiling in the reactor vessel can **NOT** be determined, it must be assumed that 212°F will be exceeded.

3.2.5 **OBTAIN** the approximate time to boiling in the reactor vessel based on current plant conditions (values should be in Daily Schedule Report). ☐

3.2.6 **REFERENCE** Technical Specifications 3.4.7, 3.4.8, 3.9.7, and 3.9.8 for actions required for loss of RHR shutdown cooling. ☐

38. 295002AK302 001

Which ONE of the following describes why the Main Turbine automatically trips on lowering condenser vacuum?

A. Lowering condenser vacuum decreases the NPSH to the condensate pumps and the turbine is tripped to prevent a loss of feedwater to the reactor due to the Condensate pumps tripping on low suction pressure.

B✓ Lowering condenser vacuum reduces the amount of energy that can be removed from steam entering the turbine which causes increased dynamic loading on the last stage blades and increases turbine vibration.

C. The turbine is tripped on lowering condenser vacuum to prevent a steam leak into Secondary Containment due to a positive pressure occurring within the condenser and rupturing the turbine casing overpressure discs.

D. The turbine is tripped on lowering condenser vacuum to prevent damage to the turbine shaft due to increased torque from trying to push the steam through the turbine.

Feedback

References: SD-26 Rev. 7, page 71

Bank question from NRC 2003 Exam QID 295002AK3.02 1 - No changes made
Distractor Analysis
A. Incorrect since lowering condenser vacuum will not affect the suction pressure to the condensate pumps enough to trip the pumps.

B. Correct answer.

C and D. Incorrect since the reason the turbine is tripped is due to higher vibrations and increased loading on the last turbine blades.

Notes

APE: 295002 Loss of Main Condenser Vacuum
AK3. Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM :
(CFR: 41.5 / 45.6)
AK3.02 Turbine trip.. 3.4 3.4

This question matches the k/a in that it measures the RO's knowledge of the bases of the turbine trip due to low condenser vacuum.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-026*27R
Ref Req'd Y or N:	NO	Technical Ref.:	SD-26
? Cognitive Level:	M OR FK	? Source:	BANK NRC

4.3.11 Loss or Malfunction of UPS

The Turbine Supervisory Instrumentation for Unit 1 uses UPS as the power source to the cabinets, a loss would remove these protective features from the turbine. (Unit 2 Turbine Supervisory Instrumentation is powered from 120 VAC.)`

4.3.12 Loss or Malfunction of DC Power

The Trip System is supplied DC Power from either the Normal Inverter (Panel 10A) powered 125 VDC Power Supply in the EHC Cabinet or from 125 VDC Panel 4AB through Switch S2 in the EHC Cabinet. Power to the Turbine Trip Logic will automatically transfer to backup on failure of the normal supply.

4.3.13 Loss or Malfunction of FWLCS

The Feedwater Level Control System maintains Reactor Vessel level in a required band. A malfunction causing a High Vessel Level would cause a Turbine Trip, a malfunction causing Low Vessel Level will actuate an RPS Trip on Low Level and a Turbine Trip. The trip units which generate a Turbine Trip are part of the FWLC System.

4.3.14 Loss of Vacuum

A lowering vacuum will reduce the amount of energy that can be removed from the steam entering the turbine. Generator output will go down. A loss of vacuum will result in a Turbine trip at 22.4 in. Hg. Vacuum, decreasing. As steam flow decreases, last stage blading dynamic loading increases which can cause added vibration.

4.3.15 Loss or Malfunction of Stator Water Cooling

The Stator Water System is required to cool the Generator Components above 30%. A loss of cooling requires a reduction in generator load to ensure no overheating occurs. If the load is not reduced in the required amount of time a Turbine Trip is activated to protect the Generator.

39. 295003G2.1.28 001

Unit One (1) was operating at power with DG2 under clearance. A dual unit Loss of Off-Site Power occurred simultaneously with a line break in the Unit One (1) drywell.

DGs 1, 3, & 4 have tied to their respective E buses. A LOCA initiation signal has NOT been received on either unit. The Unit One (1) SCO has directed cross-tie of E2 to E4.

Which ONE of the following identifies why AOP-36.1 directs that the E2-E4 cross-tie breakers' selector switches be placed in MAINT?

A. The MAINT position will not de-energize E2 in the event of a LOCA signal on Unit One (1).

B. The MAINT position will not de-energize E2 in the event of a LOCA signal on Unit Two (2).

C✓ The MAINT position protects DG4 from exceeding maximum load limit in the event of a LOCA signal on either unit.

D. The MAINT position protects the E2-E4 tie bus from exceeding maximum current limit in the event of a LOCA signal on either unit.

Feedback

RANDOMLY SELECTED FROM LOI AOP BANK LOI-CLS-LP-302-G*09G 001

SD-50.1 REV. 6 PAGE 59

Distractor Analysis
A and B - INCORRECT- In MAINT, the tie breakers will trip on LOCA signal in either unit.
C - CORRECT- With DG4 carrying two E buses, the maximum load limit of 3850 KW could easily be exceeded if a LOCA signal causes load sequencing of ECCS pumps.
D - INCORRECT - The E2-E4 tie bus is rated at 1200 amps (the same rating as bus E2) which is far in excess of the current corresoponding to 3850 KW.

Notes

APE: 295003 Partial or Complete Loss of A.C. Power
2.1.28 Knowledge of the purpose and function of major system components and controls.
(CFR: 41.7)
IMPORTANCE RO 3.2 SRO 3.3

This question matches the k/a in that it measures the RO's knowledge of major system components and control for cross-tie breakers which would be used in a loss of AC power scenario.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.2	Facility Objective:	CLS-LP-302-G*09G
Ref Req'd Y or N:	NO	Technical Ref.:	SD-50.1
? Cognitive Level:	C/A	? Source:	BANK LOI

In the MAINT position LOCA trips for the crosstie breakers are enabled. A LOCA signal from either division of either unit will energize a LOCA trip relay (TR) and also start a 10 minute timer. At the end of the 10 minute time delay, the LOCA trip relay drops out and the crosstie can be re-established. Note that if the breaker is placed in MAINT with a LOCA already present, the 10 minute time delay will start at the time the selector switch is placed in MAINT no matter how long the LOCA has been present. The LOCA signal comes from the ESS logic cabinets and is initiated by Core Spray initiation logic.

Control of E1 to E2 Cross-Tie Breaker

Control of the E1 to E2 cross-tie breakers is described below. Breaker closure is accomplished when directed by ASSD procedures by placing the selector switch to the FIRE position. Refer to Figure 23. This aligns 125 VDC control power to the breaker closing circuit through a local control switch. The Control Room indications and controls will remain isolated. The only closing permissive is the overcurrent lockout relay for the crosstie breaker reset.

Breaker tripping in the FIRE position can be accomplished by the local control switch. Refer to Figure 24. Automatic tripping of the breaker will occur on actuation of the overcurrent lockout relay associated with the cross-tie breaker, or if the slave breaker to both associated E-Buses (from 1C and 1D) are closed, or if an E-Bus undervoltage (27EX) condition occurs on the associated bus. Note that the undervoltage trip is blocked while holding the local control switch in the CLOSE position to allow the breaker to be closed onto a dead bus. Breaker closing sequence must be from the energized to the de-energized bus.

There are no plant procedures that currently allow use of the MAINT position on these breakers.

Control of E3 to E4 Cross-Tie Breakers

The E3 to E4 crosstie breakers are maintained in the racked out position with their control power fuses removed. There are currently no plant procedures that allow use of this crosstie capability.

40. 295004AA202 001

During normal full power operation of Unit Two (2), the following alarms are received:

UA-07	5-1, 5-2, 5-4	UA-13	5-7, 5-8
UA-08	5-1, 5-2, 5-4	UA-15	2-3
UA-09	4-2, 5-1, 5-3, 6-2	UA-17	2-3
UA-10	5-1, 5-2, 5-4	UA-19	6-3
UA-11	5-1, 5-2, 5-4	UA-21	6-3

Which ONE of the following identifies the 125 VDC distribution system that has been lost?

A. 2A-1.

B.✓ 2A-2.

C. 2B-1.

D. 2B-2.

Feedback

MODIFIED LOI AOP BANK QUESTION LOI-CLS-LP-302-G*02C 008

ORIGINAL QUESTION
During normal full power operation of Unit Two (2), the following alarms are received:

UA-09	2-5
UA-13	4-1, 5-10, 6-2, 6-10
A-01	1-6, 2-5, 2-8, 5-5, 6-4
A-02	4-4
A-03	1-3, 2-2
A-04	1-8, 5-1
A-05	1-7, 2-6, 3-6, 5-3, 5-6
A-06	1-6, 5-6, 5-8

Which 125 VDC distribution system has been lost?
A. 2A-1.
B. 2A-2
C. 2B-1
D. 2B-2

PROVIDE THE FOLLOWING AS REFERENCE - AOP-39.0, Figure 2 and Attachment 1 to be provided as reference for this question.

DISTRACTOR ANALYSIS
A, C, D, - INCORRECT - Annunciators indicate loss of Panels 2A, and 8A which are powered from 2A-2.
B. - CORRECT

Notes

APE: 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

This question matches the k/a in that it measures the RO's ability to determine the extent of the DC power loss by using AOP.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-302-G*02C
Ref Req'd Y or N:	Y AOP-39 FIG2 & AT 1	Technical Ref.:	AOP-39
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

PROVIDED AS REFERENCE

ATTACHMENT 1

Page 1 of 3

Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
UA-19 6-3	X										
UA-21 6-3	X										
UA-15 2-3/UA-17 2-3	X										
UA-09 4-2		X			X						
UA-09 5-1		X					X				
UA-09 5-3		X					X				
UA-09 6-2		X									
UA-13 5-7		X					X				
UA-13 5-8		X					X				
UA-07 2-3							X				
UA-07 5-1		X									
UA-07 5-2		X					X				
UA-07 5-4		X									
UA-08 2-3							X				
UA-08 5-1		X									
UA-08 5-2		X					X				
UA-08 5-4		X									
UA-10 2-3							X				
UA-10 5-1		X									
UA-10 5-2		X					X				
UA-10 5-4		X									
UA-11 2-3							X				
UA-11 5-1		X									
UA-11 5-2		X					X				
UA-11 5-4		X									

Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
A-06 5-8			X						X		
UA-13 6-10			X		X			X	X	X	
A-03 1-3			X								
A-03 2-2			X						X		
A-01 5-5			X								
A-01 6-4			X								
A-01 2-5			X								
A-01 1-6			X								
A-01 2-8			X								
A-02 4-4			X								
UA-09 2-5				X							
A-04 5-1					X						
A-04 1-8					X						
A-05 1-7					X						
A-05 5-6					X					X	
A-05 3-6					X					X	
A-05 2-6					X					X	
A-05 5-3					X						
A-06 1-6					X						
A-06 5-6					X						
UA-13 4-1					X						
UA-13 6-2					X						

ATTACHMENT 1
Page 3 of 3
Annunciators Associated with Losses of Various DC Panels

	1A/2A	7A/8A	3A/4A	5A/6A	11A/12A	1B/2B	7B/8B	3AB/4AB	3B/4B	11B/12B	9A/10A
UA-13 5-10					X						
UA-20 6-3						X					
UA-22 6-3						X					
UA-16 2-3/UA-18 2-3						X					
UA-05 4-2								X			
A-01 6-5									X		
A-03 1-4									X		
A-03 6-5									X		
A-03 1-6									X		
A-03 2-7									X		
A-02 4-8									X		
A-04 2-8										X	
A-04 6-1										X	
A-05 2-7										X	
A-05 5-4										X	
A-06 2-6										X	
A-06 6-6										X	
UA-13 5-2										X	
UA-13 6-3										X	
A-06 6-4											X
A-06 6-5											X

BRUNSWICK OCT/NOV 2004

EXAM 50-325, 324/2004-301
OCTOBER 29, 2004 &
NOVEMBER 2 - 10, 2005

DRAFT RO WRITTEN EXAM
RO #40

Figure 2, Unit 2 DC Distribution Figure
Rev 18 - Page 14 of 18

INTENTIONALLY OMITTED
PER SISP REVIEW

41. 295005AA208 001

Unit Two (2) is operating at 20% power during unit shutdown per GP-05. Electrical plant status:

BOP Bus 2C & 2D	Power from the SAT
230 KV Bus 2A	All PCBs closed except PCB 29A
230 KV Bus 2B	All PCBs closed
MOD 26A	Closed
MOD 26B	Open

The operator trips the Main Turbine per OP-26. The Main Generator backup lockout relay trips. PCB 29B fails to trip, resulting in a 230 KV Bus 2B lockout and a primary generator lockout.

Which ONE of the following describes how the 4KV electrical distribution system and Diesel Generators (DGs) will respond?

A. BOP Buses 2C and 2D de-energize, Emergency Buses E3 and E4 de-energize, DGs auto start and tie to Emergency Buses E3 and E4.

B. BOP Buses 2C and 2D remain energized, Emergency Buses E3 and E4 remain energized from BOP distribution, all DGs remain in standby.

C. BOP Buses 2C and 2D remain energized, Emergency Buses E3 and E4 de-energize, DGs auto start and tie to Emergency Buses E3 and E4.

D✓ BOP Buses 2C and 2D remain energized, Emergency Buses E3 and E4 remain energized from BOP distribution, all DGs auto start due to a LOOP signal.

Feedback

RANDOMLY SELECTED FROM LOI BANK LOI-CLS-LP-050-A*08A 002

REFERENCE - 2OP-26 Rev. 95 page 49, 2APP-UA-09 (1-3) rev. 13, and 2APP-UA-13 (1-1) Rev. 29

DISTRACTOR ANALYSIS
A, B, and C - INCORRECT -Distribution is aligned per procedure to shutdown Turbine. BOP buses transferred to SAT and generator PCB is open to 230 KV bus feeding SAT. This is done so that if a breaker failure does occur during turbine trip, off-site power feed to SAT is unaffected. 230 KV Bus 2A remains energized when 230 KV Bus 2B locks out, the SAT and BOP buses remain energized, no trip signal for Master/Slave to E Buses and they remain energized from SAT, DGs auto start however from LOOP signal from primary gen L/O.
D - CORRECT

Notes

APE: 295005 Main Turbine Generator Trip
AA2. Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP :
(CFR: 41.10 / 43.5 / 45.13)
AA2.08 Electrical distribution status........................ 3.2 3.3

This question matches the k/a in that it measures the RO's ability to determine electric plant status following a turbine trip.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.2	Facility Objective:	CLS-LP-050-A*08A
Ref Req'd Y or N:	NO	Technical Ref.:	SD-50.1 OP-27
? Cognitive Level:	C/A	? Source:	BANK LOI

7.1.2 **Procedural Steps**

5. **ROTATE** the five Cuno filter's T-handles clockwise two complete revolutions to clean the lift pumps' Cuno filters. ☐

6. **START** the Motor Suction Pump. ☐

7. **START** the Turning Gear Oil Pump **AND ENSURE** the lift pumps start by observing all six amber *PRESS NORM* lights come on. ☐

CAUTION

IF the bypass valve does NOT open, THEN tripping the main turbine may result in a main turbine trip without bypass valves with resultant high reactor pressure and APRM upscale alarms or trips.

8. **PERFORM** the following to test the operation of *LOW CONDENSER VACUUM SWITCHES, OG-PS-110* and *OG-PS-111*, the pressure set control, and the bypass valves:

   a. **DEPRESS** the Bypass Valve Opening Jack *INCREASE* push button to open a bypass valve 12% to 18%. ☐

   b. **DEPRESS** the Bypass Valve Opening Jack *DECREASE* push button to fully close the bypass valve. ☐

9. **MONITOR** main generator megawatts using computer point B059 **OR** Net Unit Megawatts 2-GEN-WMR-760. ☐

295005AA208 QUESTION BASED ON THIS STEP

10. **OPEN** the generator output breaker feeding the SAT, either *GENERATOR 2 TO BUS 2A 230 KV PCB 29A* **OR** *GENERATOR 2 TO BUS 2B 230 KV PCB 29B*. ☐

230 KV BUS 2B LOCKOUT RELAY TRIP

AUTO ACTIONS

1. 230 KV PCB-27B trips and locks out.
2. 230 KV PCB-28B trips and locks out.
3. 230 KV PCB-29B trips and locks out.
4. 230 KV PCB-30B trips and locks out.
5. 230 KV PCB-31B trips and locks out.
6. 230 KV MOD 89-26B trips and locks out. SAT on 26-A
7. If fault is on SAT, 230 KV MOD 89-ST2 also trips and locks out.
8. If fault is on Transformer Bus 2, 230 KV MODs 89-ST2, 89-26A, and 89-T2 also trip and lock out.

CAUSE

1. 230 KV Bus 2B differential overcurrent.
2. 230 KV Bus 2B overcurrent.
3. Breaker failure from any one of the following:
   a. 230 KV PCB-27B.
   b. 230 KV PCB-28B.
   c. 230 KV PCB-29B.
   d. 230 KV PCB-30B.
   e. 230 KV PCB-31B.
4. SAT differential.
5. Transformer Bus 2 fault.
6. Circuit malfunction.

OBSERVATIONS

1. The following indicate open on RTGB Panel XU5:
   a. 230 KV PCB-27B.
   b. 230 KV PCB-28B.
   c. 230 KV PCB-29B.
   d. 230 KV PCB-30B.
   e. 230 KV PCB-31B.
   f. 230 KV MOD 89-26B.
2. If fault is on SAT, 230 KV MOD 89-ST2 indicates open.
3. If fault is on Transformer Bus 2, 230 KV MODs 89-ST2, 89-26A and 89-T2 indicate open.
4. XFMR BUS 2 LOCKOUT RELAY TRIP (UA-09 1-2) alarm.
5. SAT L/O RELAY TRIP (UA-13 1-9) alarm.
6. CASWELL XFMR LOCKOUT RELAY TRIP (UA-09 1-4) alarm.

ACTIONS

1. Record the following relay indications in the switchyard relay house and inform the System Load Dispatcher:
   a. Differential Relay 87P-2B.
   b. Differential Relay 87B-2B.

ACTIONS (Continued)

1. c. Breaker Failure Relay 62BF/27B.
   d. Breaker Failure Relay 62BF/28B.
   e. Breaker Failure Relay 62BF/29B.
   f. Breaker Failure Relay 62BF/30B.
   g. Breaker Failure Relay 62BF/31B.
2. Request permission from the System Load Dispatcher to reset any tripped relay and the lockout.
3. Request permission from the System Load Dispatcher to energize any deenergized line.
4. If fault is not on SAT or Transformer Bus 2, energize SAT from 230 KV Bus 2A per OP-50, Plant Electric System.
5. If fault is not on Caswell Beach transformer or Transformer Bus 2, energize the Caswell Beach pumping station from 230 KV Bus 2B per OP-50, Plant Electric System.
6. If fault is on the SAT, refer to APP UA-13 1-9, SAT L/O RELAY TRIP.
7. If fault is on transformer Bus 2, refer to APP UA-09 1-2, XFMR BUS 2 LOCKOUT RELAY TRIP.
8. If fault is on Caswell Beach Transformer, refer to APP UA-09 1-4, CASWELL XFMR LOCKOUT RELAY TRIP.
9. If off-site power is lost, refer to AOP-36.1, Loss of Any 4KV Buses, for additional actions.
10. If any relay trips again or if a circuit malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Device 86P actuated by:	Energized
230 KV Bus 2B Differential Overcurrent Relay 87P/2A	
Phase A, B, C 230 KV Bus 2B Instantaneous Overcurrent Relay 50	700 amps
Phase A, B, C 230 KV Bus 2A Phase Timed Overcurrent Relay 51	600 amps @ 3.75 seconds

POSSIBLE PLANT EFFECTS

1. Loss of SAT and Caswell Beach pumping station.
2. Diesel generators running.
3. Reduced reactor power.

REFERENCES

1. 9527-LL-9355 - 23
2. OP-50, Plant Electric System
3. APP UA-09 1-2, XFMR BUS 2 LOCKOUT RELAY TRIP
4. APP UA-13 1-9, SAT L/O RELAY TRIP
5. APP UA-09 1-4, CASWELL XFMR LOCKOUT RELAY TRIP
6. AOP-36.1, Loss of Any 4KV Buses

R5 GEN-XFMR PRIMARY L/O UNIT TRIP

AUTO ACTIONS

1. Generator output breakers trip and lockout.
2. Turbine trips.
3. If 4.16 KV Bus 2C is being powered from the UAT, the UAT to Bus 2C circuit breaker opens and the SAT to Bus 2C circuit breaker closes.
4. If 4.16 KV Bus 2D is being powered from the UAT, the UAT to Bus 2D circuit breaker opens and the SAT to Bus 2D circuit breaker closes.
5. Diesel generators start.
6. Exciter field breaker trips and locks out.
7. Heater Drain Pumps trip.

CAUSES

1. UAT differential
2. Generator reverse power
3. Unit differential
4. System fault current
5. Loss of field
6. Generator Out-of-Step (OOS)
7. Generator Accidental Energization
8. Circuit malfunction

OBSERVATIONS

1. Generator output breakers open
2. Generator MW meter decreasing to zero
3. Power being fed from SAT
4. TURBINE MASTER TRIP (UA-23 1-4) alarm
5. Diesel generators running
6. Exciter field breaker open
7. If trip due to a fault or OOS condition, UA-13 (4-1) in alarm
8. Running Heater Drain Pumps tripped

ACTIONS

1. For a turbine trip, refer to APP UA-02 and APP UA-23.
2. If a fire is suspected, shut down bus duct cooling fans at the cooling unit control panel or at MCC 2TJ, Compartment MF4 and MCC 2TM, Compartment MP7.
3. If a Digital Relay Trip caused the unit trip, then refer to UA-13 (4-1).

DEVICE/SETPOINTS

Device 86GP1-2 actuated by:	Energized
UAT Differential Relay 87UT	3.1 amps
Reverse Power Relay 32	2080 KW
Unit Differential Relay 87GT	3.7 amps
SEL-300G Digital Relay (21G-2/78)	Transformer impedance/OOS detected/Generator Accidental Energization
Loss of Field Relay 40	20% restraint

POSSIBLE PLANT EFFECTS

1. Loss of unit generator.

REFERENCES

1. LL-9351 - 28
2. 2APP UA-02
3. 2APP UA-23
4. APP-UA-13(4-1), DGTL RELAY TRIP/TROUBLE ALARM
5. SOER 02-03, Recommendation 3b R5

42. 295006AA1.07 001

A manual scram from rated thermal power was inserted on Unit One (1). 128 Control Rods have fully inserted and nine control rods are at position 02.

Which ONE of the following identifies the expected indications on the RWM CONFIRM SHUTDOWN OD screen?

A.✓ ALL RODS IN: NO
SHUTDOWN: NO

B. ALL RODS IN: YES
SHUTDOWN: NO

C. ALL RODS IN: NO
SHUTDOWN: YES

D. ALL RODS IN: YES
SHUTDOWN: YES

Feedback

MODIFIED BANK - CLS-LP-07.1*012

REFERENCE SD-7.1 Rev. 2 page 53

DISTRACTOR ANALYSIS

A. - CORRECT - ALL RODS IN: NO means that at least one rod is not at 00. SHUTDOWN NO on Unit One (1) indicates that at least one control rod is withdrawn beyond position N as defined in the Cycle Management Report. N=00 for Unit One and N=02 for Unit Two. Since nine rods are not full in then all nine rods are withdrawn beyond 00.

B - INCORRECT - Nine control rods are not at 00 so RWM should read ALL RODS IN: NO

C- INCORRECT- SHUTDOWN: YES can only be indicated if all rods are at 00 on Unit One. This choice would be correct for Unit Two. EOPs state that on Unit One that 10 control rods can be withdrawn to 02 and the reactor is shutdown under all conditions without boron provided no control rod is withdrawn beyond 02. The RWM does not factor that into the criteria for yes. So, in this scenario, the reactor can be said to be shutdown under all conditions without boron but the RWM will indicate SHUTDOWN: NO.

D - INCORRECT- This would require all rods to be at 00.

Notes

APE: 295006 SCRAM

AA1. Ability to operate and/or monitor the following as they apply to SCRAM :

(CFR: 41.7 / 45.6)

AA1.07 Control rod position.................................. 4.1 4.1

This question matches the k/a in that it measures the RO's ability to monitor control rod position with the RWM following the scram.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 4.1	Facility Objective:	CLS-LP-07.1*012
Ref Req'd Y or N:	NO	Technical Ref.:	SD-7.1 & OI-37.3
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

5. Data Upload Facility

Control rod drive data and control rod scram time data both occupy the same storage memory in the RWM-CD as part of the rod scram time facility. The data is uploaded to ERFIS and analyzed by the program RODTIM. RODTIM uses the total notch time to determine if the uploaded data is drive or scram data.

3.11 Confirmation of Scram Rods and Full-In Indication

This function provides for indication that an all rod full-in condition and that a reactor shutdown has occurred following a scram.

3.11.1 All Rods Full-In Indication

A positive indication that all rods are full-in is provided within 500 milliseconds following the existence of that condition. The indication is retained unless any rod is observed at a higher numbered position after it is observed at position 00.

The all rods full-in test is automatically initiated following a scram event. A single all rods full-in status signal is transmitted to ERFIS once per second.

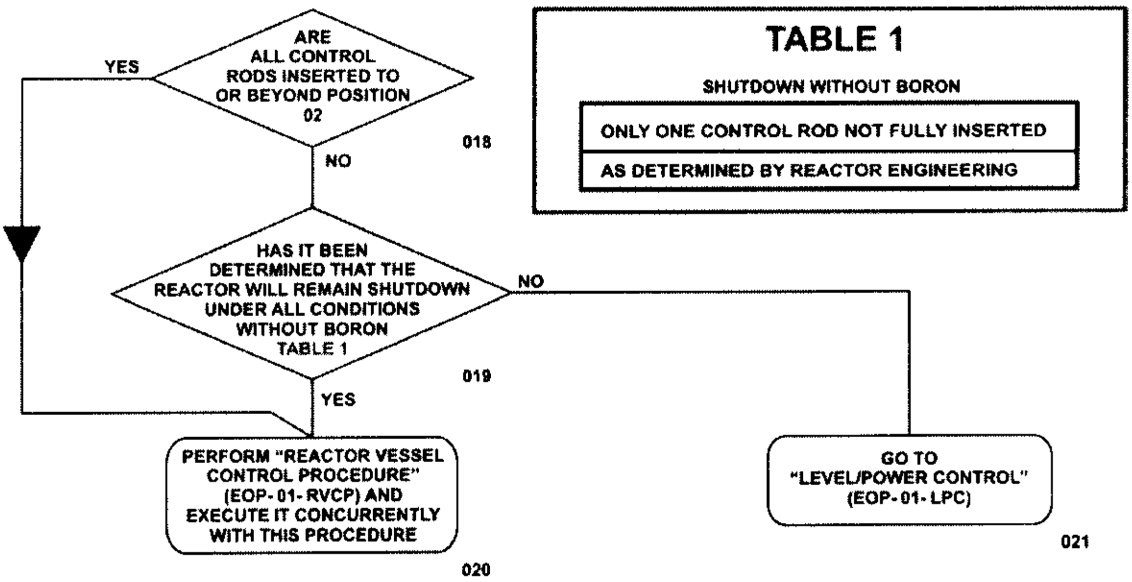
3.11.2 Reactor Shutdown Verification

Reactor shutdown is defined as being subcritical at 68°F, xenon free conditions. The shutdown margin as defined in Technical Specification may or may not be met.

A positive indication of reactor shutdown is provided within 500 milliseconds following the initiation of the scram event and is updated every 500 milliseconds until it is superseded by the all rods full-in indication or until manual intervention occurs.

Criteria for reactor shutdown is defined as: No rod is withdrawn beyond position N, where N is selectable as a set parameter variable. The value of N is defined in the current Cycle Management Report published by the Nuclear Fuels Management and Safety Analysis Section. The identity and position of rods which are not full-in can be displayed following each evaluation cycle. All 137 control rods may be displayed at 16 rods per screen if all are not at or past position 00.

STEPS 018 through 021



STEP BASES:

These steps determine whether entry to the Reactor Vessel Control Procedure or the Level/Power Control procedure is required based on whether or not a cold shutdown control rod configuration exists. If cold shutdown is not assured on control rods alone, then entry to the Level/Power Control procedure is directed where the required actions to control reactor water level, pressure, and power and to insert control rods are found. If cold shutdown is assured, then entry to the Reactor Vessel Control Procedure, where guidance on the control of reactor water level and pressure are found, is directed. Execution of the Reactor Vessel Control Procedure and the remainder of the Reactor Scram Procedure are then performed concurrently.

Positive confirmation that the reactor will remain shut down under all conditions is best obtained by determining that no control rod is withdrawn beyond the Maximum Subcritical Bank Withdrawal Position, of position 00 (Unit 1 Only) [position 02 (Unit 2 Only)]. Table 1 has been added to provide a listing of those conditions for the reactor being shutdown under all conditions without boron. This was added specifically for Unit 1 where 10 control rods could be withdrawn to position 02 as long as no control rod is withdrawn beyond position 02.

On a loss of UPS or any other condition where control rod position can not be determined, then entry to the Level/Power Control procedure is required.

43. 295009AK103 001

Which ONE of the following describes the operational implications as it applies to Reactor Vessel Water Level - Low Level 1?

The Reactor Vessel Water Level - Low Level 1 Allowable Value is selected to ensure that during normal operation the:

A. Narrow Range reference leg is not uncovered in order to preclude reference leg flashing.

B✓ steam dryer seal skirt and lowest steam separator skirt are not uncovered to protect available recirculation and jet pump NPSH from significant carryunder.

C. steam quality exiting the reactor vessel is sufficiently high enough to minimize carryover which could cause excessive turbine blade and steam piping erosion.

D. level is sufficiently above Reactor Water - Low Level 3 to ensure that initiation of low pressure ECCS subsystems will not be required in the event of a small break LOCA.

Feedback

Reference TS Bases 3.3.1.1 Applicable Safety Analysis pg B 3.3.1.1-17 rev. 30
The Reactor Vessel Water Level - Low Level 1 Allowable Value is selected to ensure that during normal operation that the steam dryer seal skirt and lowest steam separator skirt are not uncovered in order to protect available recirculation and jet pump NPSH from significant carryunder.
Reference SD-01 page 23 rev. 6
Excessive carryunder results in lower density fluid reaching reactor recirc. and jet pumps thus decreasing available NPSH.
Distractor Analysis
A - incorrect - reference leg flashing will occur with elevated reference leg area temperature not low level.
B - correct - as stated in TS bases
C- incorrect - LL1 AV is not based upon carryover. This distractor describes carryover which results with reactor water level too high not too low.
D- incorrect - LL1 is also based on ensuring that initiation of low pressure ECCS subsystems at Reactor Water - Low Level 3 will not be required in the event of a loss of high pressure feed not SBLOCA.

Notes

APE: 295009 Low Reactor Water Level
AK1. Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL :
(CFR: 41.8 to 41.10)
AK1.03 Jet pump net positive suction head: Not-BWR-1&2...... 2.7 2.7

This question matches the k/a in that it measures the RO's knowledge of the operational implications described in Tech Specs for operating with low water level and the effect on jet and recirc pump NPSH.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 2.7	Facility Objective:	CLS-LP-001*009
Ref Req'd Y or N:	NO	Technical Ref.:	TS AND SD-01
? Cognitive Level:	M OR FK	? Source:	NEW

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Reactor Vessel Water Level—Low Level 1 (continued)

are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

The Reactor Vessel Water Level—Low Level 1 Allowable Value is selected to ensure that during normal operation the steam dryer seal skirt and lowest steam separator skirt are not uncovered (this protects available recirculation pump net positive suction head (NPSH) from significant carryunder) and, for transients involving loss of all normal feedwater flow, initiation of the low pressure ECCS subsystems at Reactor Vessel Water—Low Level 3 will not be required. The Allowable Value is referenced from reference level zero. Reference level zero is 367 inches above the vessel zero point.

The Function is required in MODES 1 and 2 where considerable energy exists in the RCS resulting in the limiting transients and accidents. ECCS initiations at Reactor Vessel Water Level—Low Level 2 and Low Level 3 provide sufficient protection for level transients in all other MODES.

5. Main Steam Isolation Valve—Closure

MSIV closure results in loss of the main turbine and the condenser as a heat sink for the nuclear steam supply system and indicates a need to shut down the reactor to reduce heat generation. Therefore, a reactor scram is initiated on a Main Steam Isolation Valve—Closure signal before the MSIVs are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analyses of References 4, 7, and 8, the Average Power Range Monitor Fixed Neutron Flux—High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 2.

The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

b. **Steam Carryunder**

Steam carryunder is defined as that steam entrained with the liquid draining to the downcomer from the steam separators and dryers. Carryunder is always present to some extent, but can become excessive due to a low reactor water level condition when steam is pulled down into the bulk water region below the dryer skirt and mixed with feedwater. The problem with an excessive steam carryunder condition is that this entrained steam results in a lower density fluid reaching the reactor recirculation pumps and jet pumps and decreasing the available net positive suction head (NPSH). The decrease in NPSH increases the chance of recirculation pump and jet pump cavitation. Excessive steam carryunder also decreases the margin to Core Thermal Limits (MCPR).

15. **Jet Pump Assemblies** (Figures 01-15 and 01-16)

High performance jet pumps located within the reactor vessel are used in the BWR Recirculation system. The jet pumps, which have no moving parts, provide a continuous internal circulation path for a major portion of the core coolant flow. The jet pump recirculation system provides forced circulation flow through BWR cores. This design allows natural circulation flow whenever the Reactor Recirculation (RR) pumps are not running.

The Reactor Recirculation system consists of two loops. Each RR pump takes suction from the downcomer annulus and discharges into a manifold containing five risers. Each riser in turn penetrates the reactor vessel and continues upward into the annulus. The water delivered to the top of the riser supplies the driving flow to two jet pumps. The riser top and two jet pump assemblies are commonly called a Rams Head because of the appearance. The high velocity driving flow discharged from the jet pump nozzle creates a low pressure area inducing surrounding water (driven flow) from the downcomer to be drawn into the jet pump throat. The water flows downward through the mixing section, to the diffuser and discharges into the lower core plenum.

There are 20 jet pump assemblies located in two semicircular groups (10 each) in the downcomer annulus between the core shroud and the reactor vessel wall. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser.

44. 295012G2.1.23 001

During execution of 0-ASSD-02, Drywell Temperature Indications from 2-CAC-TR-778 are as follows:

Pt 1 = 285°F
Pt 3 = 268°F
Pt 4 = 230°F

Which ONE of the following identifies Average Drywell Temperature as determined per 0-ASSD-02?

A. 239.7°F.

B.✓ 252.9°F.

C. 261.0°F.

D. 270.5°F.

MODIFIED LOI AOP BANK LOI-CLS-LP-304-A*13P 001

ORIGINAL QUESTION

During execution of 0-ASSD-02, Drywell Temperature Indications from 2-CAC-TR-778 are as follows:

Pt 1 = 285°F
Pt 3 = 235°F changed to 268°F
Pt 4 = 230°F

Calculate the Average Drywell Temperature.

A. 239.7°F. left as is old correct answer
B. 250.0°F. changed to 252.9 new correct
C. 252.7°F. changed to 261 - straight average w/o weight factors
D. 257.3°F. changed to 270.5 - numbers reversed in calc sheet

A. Performed from Calculation Sheet 1 in 0-ASSD-02.
PT 1(.14) + PT 3(.4) + PT4(.46) = AVG DW TEMP
285(.14) + 268(.4) + 230(.46) = AVG DW TEMP
39.9 + 107.2 + 105.8 = AVG DW TEMP
239.7 = AVG DW TEMP

CALCULATION SHEET 1

Values Obtained From Recorder CAC-TR-778

80' elev

PT No. 1 ______ ______ ______ ______ ______ ______ ______ ______

x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14

A______ A______ A______ A______ A______ A______ A______ A______

28' elev

PT No. 3 ______ ______ ______ ______ ______ ______ ______ ______

x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4

B______ B______ B______ B______ B______ B______ B______ B______

13' elev

PT No. 4 ______ ______ ______ ______ ______ ______ ______ ______

x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46

C______ C______ C______ C______ C______ C______ C______ C______

Add the numbers obtained in lines A, B, and C, to obtain average DW temp.

Average
DW Temp ______________________________

Notes

APE: 295012 High Drywell Temperature
2.1.23 Ability to perform specific system and integrated plant procedures during different modes of plant operation.
(CFR: 45.2 / 45.6)
IMPORTANCE RO 3.9 SRO 4.0

This question matches the k/a in that it measures the RO's ability to determine average drywell temperature while operating oustide the control room due to fire with high drywell temperature.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.9	Facility Objective:	CLS-LP-304-A*13P
Ref Req'd Y or N:	Y ASSD-02 CALC SH 1	Technical Ref.:	0-ASSD-02
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

CALCULATION SHEET 1

Values Obtained From Recorder CAC-TR-778

80' elev

PT No. 1 ______ ______ ______ ______ ______ ______ ______ ______

x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14 x 0.14

A____ A____ A____ A____ A____ A____ A____ A____

28' elev

PT No. 3 ______ ______ ______ ______ ______ ______ ______ ______

x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4 x 0.4

B____ B____ B____ B____ B____ B____ B____ B____

13' elev

PT No. 4 ______ ______ ______ ______ ______ ______ ______ ______

x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46 x 0.46

C____ C____ C____ C____ C____ C____ C____ C____

Add the numbers obtained in lines A, B, and C, to obtain average DW temp.

Average
DW Temp ______ ______ ______ ______ ______ ______ ______ ______

45. 295013AA201 001

Unit Two (2) is operating at 19% power and operators are performing 0PT-11.1.2, Automatic Depressurization System and Safety Relief Valve Operability Test. The following "Blue Bar" annunciators are received:

UA-12 (5-4) SPTMS DIV I BULK WTR SETPOINT TS1
UA-12 (5-5) SPTMS DIV II BULK WTR SETPOINT TS1

Which ONE of the following identifies the correct interpretation of Suppression Pool Temperature as it applies to receiving the above annuciators?

Suppression Pool temperature has just reached the annunciator setpoint of:

A.✓ 95°F.

B. 100°F.

C. 105°F.

D. 110°F.

Feedback

REFERENCE UA-12 REV. 27 PAGES 51 & 52
Blue Bar alarms for SPT is 95°F.

DISTRACTOR ANALYSIS
a. correct
b. incorrect - homogeneous SPT distractor
c. incorrect - UA-12 4-3 and 5-2 setpoint
d. incorrect - corresponds to BIIT

Notes

APE: 295013 High Suppression Pool Temperature
AA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE :
(CFR: 41.10 / 43.5 / 45.13)
AA2.01 Suppression pool temperature.......................... 3.8 4.0

This question matches the k/a in that it measures the RO's ability interpret that the alarms indicate that SPT has reached 95°F.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.8	Facility Objective:	CLS-LP-300-L*002
Ref Req'd Y or N:	NO	Technical Ref.:	UA-12
? Cognitive Level:	C/A	? Source:	NEW

SPTMS DIV I BULK WTR TEMP SETPOINT TS1

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TS1 indicator illuminated (CAC-TY-4426-1).

ACTIONS

1. If suppression pool temperature is approaching 95°F and no testing is in progress that could add heat to the suppression pool, then refer to AOP-14.0, Abnormal Primary Containment Conditions.
2. If suppression pool temperature is greater than 95°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	95°F

POSSIBLE PLANT EFFECTS

1. Manual Reactor Scram required if suppression pool temperature exceeds 110°F.
2. This annunciator is required to be operable to support Suppression Chamber Temperature Instrumentation operability; annunciator inoperability may result in a TRM Compensatory Measure.

REFERENCES

1. Technical Specifications 3.6.2.1, 3.3.3.1-1, Function 4
2. AOP-14.0, Abnormal Primary Containment Conditions
3. EOP-02-PCCP, Primary Containment Control
4. TRM Section 3.8

SPTMS DIV II BULK WTR TEMP SETPOINT TS1

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-2A indicates increasing suppression pool temperature.
2. TS1 indicator illuminated (CAC-TY-4426-2).

ACTIONS

1. If suppression pool temperature is approaching 95°F and no testing is in progress that could add heat to the suppression pool, then refer to AOP-14.0, Abnormal Primary Containment Conditions.
2. If suppression pool temperature is greater than 95°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-2	95°F

POSSIBLE PLANT EFFECTS

1. Manual Reactor Scram required if suppression pool temperature exceeds 110°F.
2. This annunciator is required to be operable to support Suppression Chamber Temperature Instrumentation operability; annunciator inoperability may result in a TRM Compensatory Measure.

REFERENCES

1. Technical Specifications 3.6.2.1, 3.3.3.1-1, Function 4
2. AOP-14.0, Abnormal Primary Containment Conditions
3. EOP-02-PCCP, Primary Containment Control
4. TRM Section 3.8

SPTMS DIV I BULK WTR TEMP SETPT TMAX

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-1A indicates increasing suppression pool temperature.
2. TMAX indicator illuminated (CAC-TY-4426-1).

ACTIONS

1. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions, if not already entered.
2. If suppression pool temperature is approaching 105°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 105°F, then stop all testing and enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, then ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-1	105°F

POSSIBLE PLANT EFFECTS

1. Manual Reactor Scram required if suppression pool temperature exceeds 110°F.

REFERENCES

1. Technical Specifications 3.6.2.1
2. AOP-14.0, Abnormal Primary Containment Conditions
3. EOP-02-PCCP, Primary Containment Control

SPTMS DIV II BULK WTR TEMP SETPT TMAX

AUTO ACTIONS

NONE

CAUSE

1. High suppression pool bulk average water temperature.

OBSERVATIONS

1. Recorder Channel 1 on CAC-TR-4426-2A indicates increasing suppression pool temperature.
2. TMAX indicator illuminated (CAC-TY-4426-2).

ACTIONS

1. If suppression pool temperature is greater than 95°F and no testing is in progress that could add heat to the suppression pool, then enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions, if not already entered.
2. If suppression pool temperature is approaching 105°F due to adding heat to the suppression pool from approved testing procedures, then refer to the appropriate test procedure to maintain suppression pool temperature below 105°F.
3. If suppression pool temperature is greater than 105°F, then stop all testing and enter EOP-02-PCCP, Primary Containment Control, and AOP-14.0, Abnormal Primary Containment Conditions.
4. If a circuit or equipment malfunction is suspected, then ensure that a WR/WO is prepared.

DEVICE/SETPOINTS

SPTMS Microprocessor CAC-TY-4426-2	105°F

POSSIBLE PLANT EFFECTS

1. Manual Reactor Scram required if suppression pool temperature exceeds 110°F.

REFERENCES

1. Technical Specifications 3.6.2.1
2. AOP-14.0, Abnormal Primary Containment Conditions
3. EOP-02-PCCP, Primary Containment Control

46. 295016AA206 001

The Control Room has been evacuated due to toxic gas in the Control Building. Shutdown from outside the Control Room is in progress per 0-AOP-32.

Reactor cooldown is in progress. The following reactor pressure values are recorded at the indicated times:

0000	900 psig
0015	750 psig
0030	600 psig
0045	450 psig
0100	300 psig

Which ONE of the following identifies the approximate Reactor Vessel cooldown rate and the expected actions required per 0-AOP-32?

Reactor Vessel cooldown rate is:

A. <100°F/Hr, and should be kept below 100°F/Hr.

B. <100°F/Hr, and should be raised to 100-120°F/Hr.

C.✓ >100°F/Hr, and should be lowered below 100°F/Hr.

D. >100°F/Hr, and should be maintained below 120°F/Hr.

Feedback

REFERENCE 0-AOP-32 REV. 36 PAGE 17; Provide Appendix 3 and Appendix 4 as reference to plot cooldown rate.

DISTRACTOR ANALYSIS

A, B and D - INCORRECT - The current cooldown rate is ~110°F/Hr which exceeds AOP-32 limit of 100°F/Hr. Distractors B and D developed as per 0-ASSD-02, Shutdown Outside Control Room for fire, Section C, Step 1.10.1: "OPEN as many SRVs as necessary to achieve a cooldown rate between 100-120°F/Hr".

C - Correct

Notes

APE: 295016 Control Room Abandonment

AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT :

(CFR: 41.10 / 43.5 / 45.13)

AA2.06 Cooldown rate.. 3.3 3.5

This question matches the k/a in that it measures the RO's ability to calculate cooldown rate and interpret results per procedure.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.3	Facility Objective:	CLS-LP-302-E*004
Ref Req'd Y or N:	Y AOP 32 APP 3 & 4	Technical Ref.:	AOP-32
? Cognitive Level:	C/A	? Source:	NEW

CAUTION

HPCI will not automatically initiate if the auxiliary oil pump Normal/Local switch is in LOCAL and the pump is not running.

(12) WHEN the HPCI turbine is not needed to maintain reactor vessel level, SECURE the HPCI turbine as follows:

(a) Station 3, CHECK that the HPCI turbine is tripped by observing the HPCI Injection Valve, E41-F006, is closed at MCC 1(2)XDA Compt B17, Row C2.

(b) Station 3, Approximately 15 minutes after the turbine has stopped rolling, SECURE the HPCI auxiliary oil pump at MCC 1(2)XDA Compt B11, Row A3.

(13) REDUCE reactor pressure as follows:

NOTE: When cycling SRVs, use sequences B, E, G to evenly distribute the heat load to the suppression pool.

NOTE: When SRVs are opened to reduce reactor pressure, the SRV should be left open for three minutes or until a 150 psig decrease in reactor pressure has occurred to minimize the number of SRV openings.

(a) PLOT cool down rate on Appendix 3 of this procedure by plotting saturation temperature from Appendix 4 against time.

(b) IF decay heat is low enough that reactor pressure is not increasing, allow the reactor to depressurize without lifting relief valves.

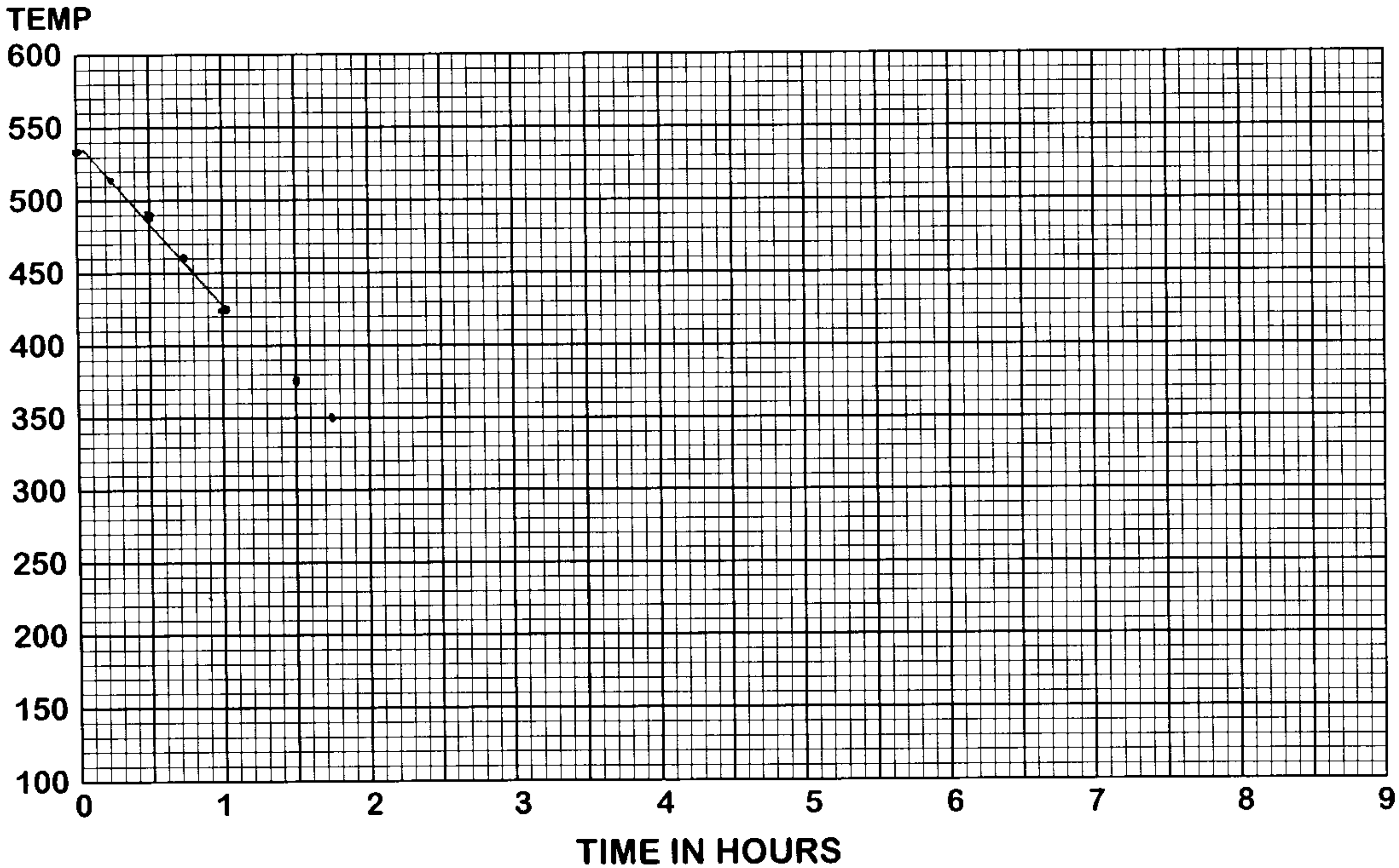
(c) IF RCIC is not needed to maintain reactor water level, perform Section 3.2(8) of this procedure to utilize RCIC for reactor pressure control.

CAUTION

Maintain reactor vessel level above 170" during relief valve operation.

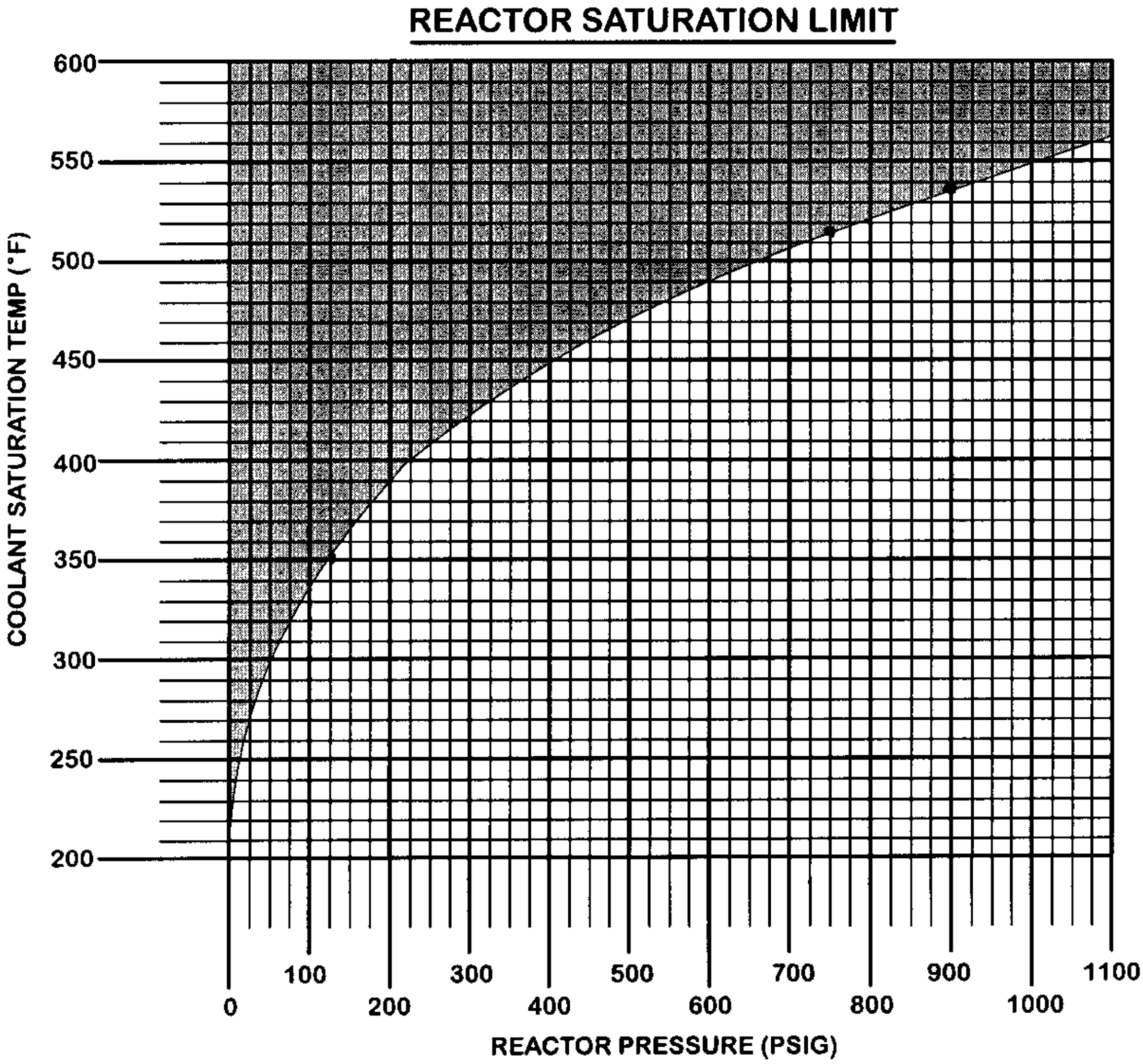
(d) OPEN SRVs as required to establish a cool down rate not to exceed 100°F/hour.

APPENDIX 3
COOLDOWN PLOT



APPENDIX 4

SATURATION CURVE



47. 295018G2.4.50 001

Unit One (1) is operating at rated power when annunciator UA-24 (1-1) CONV HDR SERV WATER PRESS-LOW was received. Conventional Service Water (CSW) header pressure on XU-2 indicates 38 psig. The RO observes the following CSW System indications:

1A CSW Pump	Selected to MANUAL and ON, Running
1B CSW Pump	Selected to AUTO and ON, Running
1C CSW Pump	Selected to MANUAL and ON, Running

After dispatching an AO to investigate two (2) minutes have elapsed and the RO notes that CSW header pressure remains at 38 psig and that the TBCCW service water supply valves, SW-V3 and SW-V4 indicate full OPEN.

Which ONE of the following describes the system response and required operator response to these conditions?

A. The CSW system responded properly since 1B CSW pump AUTO started in response to the low header pressure alarm. The RO should continue to investigate the cause of the AUTO start but no other actions are required.

B. TBCCW service water supply valves, SW-V3 and SW-V4 failed to isolate and the RO should hold the respective control switches in the CLOSED position until the valves indicate fully CLOSED.

C✓ TBCCW service water supply valves, SW-V3 and SW-V4 failed to stroke to their throttled position and the RO should momentarily place the respective control switches in CLOSED and verify that the valves travel to the throttled position.

D. With three (3) CSW pumps operating and CSW header pressure still below 40 psig it is apparent that pressure can NOT be immediately restored so the RO should manually SCRAM the reactor and perform AOP-19 concurrently.

Feedback

REFERENCE UA-24 (1-1) AND AOP-19 - CSW header press <40 psig for 70 sec. SW-V3/4 stroke to a throttled position to restore header pressure above 40 psig. If header pressure cannot be restore immediately >40 psig with V3 and V4 throttled then a SCRAM is required per AOP-19.

DISTRACTOR ANALYSIS

A. INCORRECT - V3/V4 did not stroke to throttled position

B. INCORRECT - V3/V4 do not stroke full CLOSED automatically

C. CORRECT

D. INCORRECT - SCRAM requirements not met as V3/V4 have not been stroked to throttled position

Notes

APE: 295018 Partial or Complete Loss of Component Cooling Water

2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

(CFR: 45.3)

IMPORTANCE RO 3.3 SRO 3.3

This question matches the k/a in that it measures the RO's ability verify automatic actions and operate controls to make a failed automatic action occur.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.3	Facility Objective:	CLS-LP-302-H*01C,1D
Ref Req'd Y or N:	NO	Technical Ref.:	UA-24 (1-1)
? Cognitive Level:	C/A	? Source:	NEW

CONV HDR SERV WATER PRESS-LOW

AUTO ACTIONS

1. If the service water conventional header pressure decreases to 40 psig, the standby conventional service water pump selected to AUTO will start.
2. If the service water conventional header pressure remains less than 40 psig for 70 seconds, then the TBCCW Service Water Supply Valves, SW-V3 and SW-V4, close to their throttled position.

CAUSE

1. Conventional Service Water Pump A (B or C) tripped.
2. Conventional Service Water Pump A (B or C) strainer clogged.
3. Improper valve alignment.
4. System piping failure.
5. Excessive system usage.
6. Circuit malfunction.

OBSERVATIONS

1. CONV HDR SW PUMP A (B or C) TRIP [UA-01 1-8 (4-8, 4-9)] alarm.
2. Service water conventional header pressure, as indicated on SW-PI-131-1 on RTGB Panel XU-2, indicating less than or equal to 42 psig.
3. CONV SW PUMP STRAINER A (B or C) DIFF-HIGH [UA-01 2-8 (5-8, 5-9)] alarm.
4. If pressure remained less than 40 psig for 70 seconds, 1-SW-V3 and 1-SW-V4 will be at their throttled positions.

ACTIONS

1. If the standby conventional service water pump did not automatically start, start the pump.
2. If the conventional service water pump strainer is not automatically backwashing, start the strainer backwashing.
3. Check the valve lineup.
4. If piping failure, isolate if possible.
5. If conventional header service water pressure has been restored to normal and no piping failure is indicated, then open 1-SW-V3 and 1-SW-V4.
6. If a complete loss of conventional service water occurs, and it cannot be immediately restored, refer to AOP-19.0, Conventional Service Water System Failure.
7. If a circuit or equipment malfunction is suspected, ensure that a WR/WO is prepared.

1.0 SYMPTOMS

1.1 *CONV HDR SW PUMP A TRIP* (UA-01, 1-8) in alarm.

1.2 *CONV HDR SW PUMP B TRIP* (UA-01, 4-8) in alarm.

1.3 *CONV HDR SW PUMP C TRIP* (UA-01, 4-9) in alarm.

1.4 Unit 1 only: *CONV HDR SERV WATER PRESS-LOW* (UA-24, 1-1) in alarm.

1.5 Unit 2 only: *CONV HDR SERV WATER PRESS-LOW* (UA-01, 1-9) in alarm.

1.6 Conventional Service Water Header pressure low as read on *SW-PI-131-1*, located on Panel XU-2.

R8 1.7 High Convention Service Water Header pressure approaching pump shutoff head (approximately 90 psig).

2.0 AUTOMATIC ACTIONS

2.1 Standby pump selected to the Conventional Service Water Header starts at 40 psig. ☐

2.2 **IF** all Conventional Service Water Pumps are tripped, **THEN**:

2.2.1 *SW TO CW PUMPS INBD VLV, SW-V36* closes. ☐

2.2.2 *SW TO CW PUMPS OTBD VLV, SW-V37* closes. ☐

2.2.3 Circulating Water Intake Pumps trip on low bearing lubricating water flow (7 gpm ± 1 gpm), resulting in a loss of condenser vacuum. ☐

2.3 **IF** Conventional Service Water Header pressure remains below 40 psig for 70 seconds, **THEN**:

2.3.1 *SW TO TBCCW HXS OTBD ISOL, SW-V3* closes to a throttled position. ☐

2.3.2 *SW TO TBCCW HXS INBD ISOL, SW-V4* closes to a throttled position. ☐

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

None

3.2 Supplementary Actions

3.2.1 **IF** the Conventional Service Water Header pressure is below 40 psig (*SW-PI-131-1* on Panel XU-2) **AND** *SW-V3* **OR** *SW-V4* is at its throttled position **AND** pressure can **NOT** be immediately restored, **THEN PERFORM** the following:

NOTE:	Tripping all Conventional Service Water Pumps will result in a trip of all Circulating Water Intake Pumps **AND** a loss of condenser vacuum.

1. **MANUALLY SCRAM** the reactor **AND PERFORM** 1(2)EOP-01-RSP, concurrently with this procedure. ☐

2. **TRIP** Conventional Service Water Pumps supplying the Conventional Service Water Header. ☐

3.2.2 **IF** the Conventional Service Water Header is in service to the RBCCW Heat Exchangers, **THEN REFER** to 0AOP-16.0. ☐

3.2.3 **IF** a Conventional Service Water Header break is suspected, **THEN PERFORM** the following in an attempt to isolate the break:

1. **CLOSE** *SERVICE WATER TO CHLORINATION, SW-V294.* ☐

2. **CLOSE** *SERVICE WATER TO CHLORINATION, SW-V295.* ☐

3. **CLOSE** *SW TBCCW HXS OTBD ISOL, SW-V3.* ☐

4. **CLOSE** *SW TBCCW HXS INBD ISOL, SW-V4,* **AND REFER** to 0AOP-17.0. ☐

5. **CLOSE** *CONVENTIONAL SERVICE WATER SUPPLY VALVE, SW-V101.* ☐

48. 295019AK202 001

Which ONE of the following describes how the Reactor Building Closed Cooling Water (RBCCW) Heat Exchanger Temperature Control Valve responds in the event of a loss of pneumatic pressure to the valve positioner?

The RBCCW Heat Exchanger Temperature Control Valve fails:

A. as is to maintain the current Service Water flow through the heat exchanger.

B. OPEN resulting in maximum Service Water flow through the heat exchanger.

C✓ CLOSED resulting in maximum Service Water flow through the heat exchanger.

D. to a preset throttled position which provides optimum Service Water flow during rated power conditions.

Feedback

REFERENCE SD-21 REV. 2 PAGE 10

A, C, D. INCORRECT - FAILS CLOSED, it is a HX. Bypass that fails CLOSED to ensure maximum cooling with Service Water
B - CORRECT

Notes

APE: 295019 Partial or Complete Loss of Instrument Air
AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following:
(CFR: 41.7 / 45.8)
AK2.02 Component cooling water.............................. 2.9 3.0

This question matches the k/a in that it measures the RO's knowledge of a loss of inst. air on RBCCW TCV operation.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 2.9	Facility Objective:	CLS-LP-021*AO4B
Ref Req'd Y or N:	NO	Technical Ref.:	N
? Cognitive Level:	M OR FK	? Source:	NEW

2.5 RBCCW Heat Exchanger Temperature Control Valve

As mentioned earlier, the RBCCW heat exchanger combined outlet has a temperature indicating controller installed to control outlet temperature. A pneumatically operated valve installed on the bypass line around the heat exchanger receives this signal and positions itself to control flow through and around the heat exchangers. The valve is positioned by air from the non-interruptible instrument air header and will fail closed upon a loss of supply. The temperature indicating controller is located on the 50' elevation of the Reactor Building.

2.6 2D RBCCW Pump

The 2D RBCCW pump is a 100-percent capacity, horizontal, split case, double suction centrifugal pump. The pump is designed to provide sufficient cooling water to RBCCW components located below elevation 50' or essential RBCCW components in the event temporary cooling is required. The 2D RBCCW pump can be powered from MCC RWD, Compartment BR9.

2D RBCCW pump is locally operated and has no auto start feature. Annunicator UA-05-5-10, Drywell Chiller Trip has been installed on U-2 for remote indication of problems with the Temporary Cooling System.

2.7 RBCCW Radiation Monitor

The RBCCW radiation monitor provides an indication of radioactivity within the RBCCW cooling medium. RBCCW return flow is monitored to determine the possibility of a leak from a contaminated system into the RBCCW system. The most probable source of leakage, containing radioactivity into RBCCW, would be from the RWCU system.

49. 295020AK206 001

Following a loss of feedwater on Unit Two (2), HPCI has automatically initiated on low reactor water level and is injecting to the reactor vessel. Current plant conditions are:

Reactor level	+90 inches
Reactor pressure	920 psig
Drywell pressure	0.5 psig
HPCI flow	4300 gpm

HPCI Initiation Signal/Reset white light is LIT

Which ONE of the following describes HPCI system response if the HPCI Manual Isolation System A pushbutton is inadvertantly depressed?

A. HPCI will continue to inject to the reactor.

B. The Steam Supply Valve, E41-F001, only will CLOSE.

C. The Outboard Steam Isolation Valve, E41-F003, and Steam Supply Valve, E41-F001, will CLOSE.

D✓ The Outboard Steam Isolation Valve, E41-F003, will CLOSE and the HPCI Turbine Stop Valve, E41-V8, will TRIP.

Feedback

RANDOMLY SELECTED BANK QUESTION LOI SYSTEMS LOI-CLS-LP-019-A*001 002

REFERENCE SD-19 Rev. 8 pages 37, 40 and 46 and 1APP-A-01 (3-1) rEV. 39 page 44 - Isolation Pushbutton seals in A logic only if LL2 or Hi DW pressure condition exists. a is outboard for HPCI

DISTRACTOR ANALYSIS
A, B, AND C - INCORRECT -With level <LL2 Isolation Pushbutton seals in A logic and isolates outboard for HPCI, also a turbine trip
D - CORRECT

Notes

APE: 295020 Inadvertent Containment Isolation
AK2. Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following:
(CFR: 41.7 / 45.8)
AK2.06 HPCI: Plant-Specific................................ 3.8 3.8

This question matches the k/a in that it measures the RO's ability determine the expected HPCI response to an inadvertant isolation signal.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.8	Facility Objective:	CLS-LP-019-A*001
Ref Req'd Y or N:	NO	Technical Ref.:	SD-19
? Cognitive Level:	C/A	? Source:	BANK LOI

Steam Line High Flow (sensed as high differential pressure) indicates a large break in the HPCI Turbine steam line. Differential pressure transmitters provide the flow signal and actuate a 5 second time delay relay on a high flow condition. If, after the 5 second delay, the high flow condition still exists, the isolation signal is actuated. This time delay is to prevent spurious isolation due to transient flow peaks that may occur during the turbine start sequence. NORMAL/TEST switches, located on Panels H12-P618 and P620, allow testing of the Steam Line High Flow circuitry without actually initiating an isolation signal.

The isolation system is divided into two logic systems, Logic Bus A and Logic Bus B. Logic Bus A isolates the outboard valves, E41-F003 and E41-F041, while Logic Bus B isolates the inboard valves, E41-F002 and E41-F042. A Manual Isolation pushbutton is provided on Panel P601 to permit the operator to insert an isolation should plant conditions require it. The Manual Isolation only initiates a Logic A isolation and is only in effect when a Reactor Low Level Two or a High Drywell Pressure signal is present.

With the exception of the Steam Line Low Pressure isolation, an automatic isolation signal seals in both the A and B Logic Buses and must be reset by the operator depressing the Auto Isolation Signal A and B Reset pushbuttons on Panel P601. A manual isolation also seals in on the A Logic Bus and must be reset by the operator depressing the Auto Isolation Signal A Reset pushbutton on Panel P601. The isolation logic buses will reset provided the initiating signal is no longer present.

The Steam Line Low Pressure isolation operates independently of the other Group 4 isolation logic (i.e., independent relays and contacts provide signals for the automatic closing of the pump Suppression Pool Suction Valves, E41-F041 and E41-F042, and the Steam Line Isolation Valves, E41-F002 and E41-F003). Therefore, a Steam Line Low Pressure isolation signal does not seal in and, once the low pressure condition clears, the valves may be reopened if no other isolation signal exists.

3.4.9 Power Supplies

The normal power supply to Panel P601 HPCI flow instruments, flow indicating controller and pressure transmitters is 125 Vdc Bus A. 125 Vdc power from Panel 3(4)A powers a 24 Vdc power supply, E41-ES-K603, which supplies power to the Johnson-Yokogawa FIC. Panel 3(4)A also powers a 52.5 Vdc power supply, E41-ES-K600, which supplies power for the vertical board HPCI pressure instrumentation. HPCI is not affected by a loss of the 52.5 Vdc power supply except that P601 indication is lost. In the event of a 24 Vdc power supply failure, Annunciator HPCI FIC POWER LOSS (APP A-01 2-5) alarms, resulting in the flow controller failing down scale due to the loss of power. If the HPCI Turbine is operating, turbine speed would run back to below the 2100 rpm minimum and the system would have to be secured.

3.5 HPCI Turbine Trip Control (Figures 19-25 and 26)

The HPCI turbine will automatically shutdown (Turbine Stop Valve closes) upon receipt of one of the signals listed in Table 19-7, below.

Table 19-7 - HPCI Trips

Signal	Setpoint	Tech Spec
Turbine Overspeed	5000 rpm (125% of original rated speed - 4000 rpm)	N/A
Reactor High Water Level	206"	≤207"
HPCI Pump Low Suction Pressure	15 inches after 13 sec. time delay	N/A
Turbine High Exhaust Pressure	157.5 psig	N/A
HPCI System Isolation	See Table 19-8	See Table 19-8
Manual Trip	N/A	N/A
Low Steam Line Pressure*	115 psig	≥104 psig

* This low steam line pressure trip comes from the Div. I Isolation Logic (Bus A) instruments, independent of the other HPCI System Isolation Logic.

In addition to the automatic overspeed trip, a manual trip may also be provided by the "Pull-To-Trip" manual trip knob located on the overspeed trip body on the bearing pedestal.

The other turbine trips are initiated by energizing the remote-operated solenoid oil dump valve, E41-C002-SV1. When this valve is energized by the trip circuitry, the dump valve opens, reducing the oil pressure downstream of the 3/16-inch orifice to zero psig. The Turbine Stop Valve will rapidly close as discussed above. This trip will reset when the dump valve is deenergized and, if oil pressure is available, the Turbine Stop Valve will reopen and the turbine will restart.

There is no seal in of the turbine trips, except for the trips caused by Reactor High Water Level and an isolation, thus the operator must be alert to conditions that could lead to the turbine tripping and then resetting when the trip condition clears. This could be repeated until equipment damage occurs.

A Turbine Trip pushbutton, on Panel P601, which also energizes the remote-operated solenoid oil dump valve (E41-C002-SV1) as discussed above, may be used to shut down the turbine.

A Reactor High Water Level trip is initiated and the signal seals-in when a high level is sensed by two instruments, both powered from 125 VDC Bus. Once the high level condition clears, the trip is reset by a subsequent Reactor Low Level 2 signal or upon depressing the Panel P601 High Level Trip Reset pushbutton. A High Drywell Pressure Initiation signal will not reset the High Water Level trip.

During a HPCI Turbine start, pump suction pressure could possibly drop below the trip initiation setpoint. For this reason, a 13 second time delay has been added to prevent spurious trips upon system initiation.

The power supply for the turbine trip circuitry is 125 VDC Bus A.

There are two annunciators associated with turbine trips. The HPCI TURB TRIP SOL ENER annunciator will alarm when the trip solenoid is energized (via HPCI auxiliary relay K12); while the HPCI TURB TRIP annunciator will alarm when the Turbine Stop Valve, E41-V8, is fully closed (via HPCI auxiliary relay K13) and the Turbine Steam Supply Valve, E41-F001, is not fully closed (via HPCI auxiliary relay K58).

HPCI TURB TRIP

AUTO ACTIONS

1. If the HPCI turbine trips, the following occurs:
   a. If open, the Turbine Stop Valve, E41-V8, trips closed.
   b. If open, the HPCI Injection Valve, E41-F006, closes.
   c. If open, the Minimum Flow Bypass To Suppression Pool Valve, E41-F012, closes.
2. If the HPCI turbine isolates, the following occurs:
   a. If open the Steam Supply Inboard Isolation Valve, E41-F002, closes.
   b. If open the Steam Supply Outboard Isolation Valve, E41-F003, closes.
   c. If open, the Turbine Stop Valve, E41-V8, closes.
   d. If open the HPCI Injection Valve, E41-F006, closes.
   e. If open the Minimum Flow Bypass To Suppression Pool Valve, E41-F012, closes.
   f. If open, the Suppression Pool Suction Valve, E41-F041, closes.
   g. If open, the Suppression Pool Suction Valve, E41-F042, closes.

CAUSE

1. High reactor vessel water level (206 inches).
2. Mechanical overspeed trip (5000 rpm).
3. High turbine exhaust pressure (157.5 psig).
4. Low HPCI pump suction pressure (15 inches Hg vacuum after 10 second time delay).
5. High turbine exhaust diaphragm pressure (7 psig).
6. High steam line differential pressure
7. Low steam supply pressure (115 psig).
8. High HPCI room area ambient temperature (165°F).
9. High HPCI steam line area ambient temperature (190°F).
10. High HPCI steam line tunnel temperature (165°F).
11. High HPCI steam line area differential temperature (47°F).
12. Turbine trip push button.
13. Circuit malfunction.

OBSERVATIONS

> **NOTE:** Once the turbine trips, exhaust pressure and suction pressure will return to zero or a positive value.

1. Reactor vessel water level greater than 206 inches (multiple RTGB indications).

50. 295021AA104 001

Following a dual unit Loss of Offsite Power, Unit One (1) has lost Shutdown Cooling. No other method of decay heat removal is available. Alternate Shutdown Cooling is being established per AOP-15.0, Loss of Shutdown Cooling. Conditions are as follows:

EDG#3	Tripped
EDG#1, #2, and #4	Running loaded
RHR Loop B	Suppression Pool Cooling

Which ONE of the following identifies the preferred pump for injection per AOP-15.0?

A. RHR Pump 1A.

B.✓ RHR Pump 1C.

C. Core Spray Pump 1A.

D. Core Spray Pump 1B.

Feedback

REFERENCES: AOP-15 Rev. 16 pages 10-15
RANDOMLY SELECTED FROM BANK and then MODIFIED - LOI-CLS-LP-302-L*006 001

ORIGINAL QUESTION
Following a dual unit Loss of Offsite Power, Unit Two (2) has lost Shutdown Cooling. No other method of decay heat removal is available. Alternate Shutdown Cooling is being established. Conditions are as follows:

EDG#1	Tripped
EDG#2, #3, and #4	Running loaded
RHR Loop B	Suppression Pool Cooling

The preferred pump for injection per AOP-15.0 is:

A. RHR Pump 2A. CORRECT
B. RHR Pump 2C.
C. Core Spray Pump 2A.
D. Core Spray Pump 2B.

DISTRACTOR ANALYSIS
A. INCORRECT - RHR pump 1A has no power available with EDG#3 tripped.
B. CORRECT - RHR is preferred over core spray due to injection through jet pumps. RHR pump 1C has power from E1
C and D - INCORRECT - RHR is preferred over core spray due to injection through jet pumps.

Notes

APE: 295021 Loss of Shutdown Cooling
AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING :
(CFR: 41.7 / 45.6)
AA1.04 Alternate heat removal methods........................ 3.7 3.7

This question matches the k/a in that it measures the RO's ability to determine and operate preferred alternate decay heat removal methods available with a loss of SDC.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.7	Facility Objective:	CLS-LP-302-L*006
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-15
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

3.0 OPERATOR ACTIONS

6. **IF** RHR has **NOT** been restored in accordance with Step 3.2.11.5, **THEN PLACE** the RHR loop that was operating in Shutdown Cooling back in service in accordance with 1(2)OP-17 as soon as conditions permit. ☐

3.2.12 **IF** necessary to minimize reactor coolant temperature rise, **THEN PERFORM** one of the following feed and bleed combinations: ☐

FEED	BLEED
COND/FW in accordance with 1(2)OP-32	RWCU Reject in accordance with 1(2)OP-14
CRD in accordance with 1(2)OP-08	Main Steam Line Drains in accordance with 1(2)OP-32
Core Spray in accordance with 1(2)OP-18	
LPCI in accordance with 1(2)OP-17	

3.2.13 **IF NEITHER** RHR loop can be placed in Shutdown Cooling, **THEN PLACE** the Condensate System in Condenser Cooling in accordance with 1(2)OP-32. ☐

3.2.14 **IF ALL** of the above methods can **NOT** maintain reactor vessel coolant temperature below 212°F, **THEN INITIATE** alternate Shutdown Cooling with the SRVs as follows:

1. **ENSURE ALL** control rods are fully inserted. ☐

2. **CONFIRM** reactor vessel head is installed and tensioned. ☐

3. **IF** the Reactor Recirculation Pumps are running, **THEN PERFORM** the following:

   a. **RAISE AND MAINTAIN** reactor water level between 200" and 220" as read on *B21-LI-R605A(B)*, or as directed by Shift Superintendent based on plant conditions. ☐

   b. **STOP** the running Reactor Recirculation Pumps in accordance with 1(2)OP-02. ☐

3.0 OPERATOR ACTIONS

4. **SHUT DOWN** the RHR loop that was operating in Shutdown Cooling in accordance with 1(2)OP-17. ☐

5. **PLACE** one RHR loop in the Suppression Pool Cooling mode in accordance with 1(2)OP-17. ☐

6. **IF** Suppression Pool temperature rises above 95°F, **THEN GO TO** 0EOP-02-PCCP, Primary Containment Control Procedure **AND PERFORM CONCURRENTLY** with this procedure. ☐

7. **ENSURE** the following valves are closed:

   a. *INBOARD MSIV A VLV, B21-F022A* ☐

   b. *INBOARD MSIV B VLV, B21-F022B* ☐

   c. *INBOARD MSIV C VLV, B21-F022C* ☐

   d. *INBOARD MSIV D VLV, B21-F022D* ☐

   e. *OUTBOARD MSIV A VLV, B21-F028A* ☐

   f. *OUTBOARD MSIV B VLV, B21-F028B* ☐

   g. *OUTBOARD MSIV C VLV, B21-F028C* ☐

   h. *OUTBOARD MSIV D VLV, B21-F028D* ☐

   i. *STEAM SUPPLY INBOARD ISOL VLV, E41-F002* ☐

   j. *STEAM SUPPLY OUTBOARD ISOL VLV, E41-F003* ☐

   k. *STEAM SUPPLY INBOARD ISOL VLV, E51-F007* ☐

   l. *STEAM SUPPLY OUTBOARD ISOL VLV, E51-F008* ☐

   m. *INBOARD RX HEAD VENT VLV, B21-F003* ☐

   n. *OUTBOARD RX HEAD VENT VLV, B21-F004* ☐

3.0 OPERATOR ACTIONS

o. *REACTOR INBOARD HIGH POINT MANUAL VENT VALVE, B21-F001* ☐

p. *REACTOR OUTBOARD HIGH POINT MANUAL VENT VALVE, B21-F002* ☐

q. *MAIN STEAM LINE DRAIN INBD ISOL VLV, B21-F016* ☐

r. *MAIN STEAM LINE DRAIN OUTBD ISOL VLV, B21-F019.* ☐

8. **SELECT** one SRV based upon the desired cool down rate using the following table:

NOTE:	All SRVs within the same block on the table will produce a similar cooldown rate; therefore, to effect a change in cooldown rate, an SRV in a different box must be used.

	RHR A/C	RHR B/D	CS A	CS B
HIGHEST COOLDOWN	B21-F013F B21-F013H	B21-F013A B21-F013B	B21-F013K	B21-F013E B21-F013L
	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D
	B21-F013A B21-F013B B21-F013K	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013A B21-F013B B21-F013K
	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J
LOWEST COOLDOWN	B21-F013E B21-F013L	B21-F013K	B21-F013A B21-F013B	B21-F013F B21-F013H

9. **PLACE** the control switch for the desired SRV to *OPEN.* ☐

3.0 OPERATOR ACTIONS

NOTE:	Raising RPV water level slowly using CRD is preferred to reduce stresses induced in the RPV vessel and piping when injecting cold water. RHR and CS may be used if necessary, but should be considered only after determining other methods are not effective.

10. **RAISE AND MAINTAIN** reactor water level greater than 254 inches. ☐

NOTE:	The RHR pumps are preferred for injection.
NOTE:	Monitoring T_{SAT} in accordance with 1(2)PT-01.7 may **NOT** be valid under these special conditions due to reactor pressure **NOT** necessarily relating to T_{SAT}. Therefore, SRV tailpipe temperature recorder *B21-TR-R614* on Panel H12-P614, and/or ERFIS trending should be utilized for monitoring reactor coolant cool down rate.

11. **IF** any low pressure injection system, other than the RHR Loop operating in Suppression Pool Cooling, is available, **THEN PERFORM** the following:

    a. **START** one RHR or Core Spray Pump. ☐

    b. **THROTTLE OPEN** the injection valve on the affected pump until the SRV opens. ☐

12. **IF** the only low pressure injection system available is the RHR Loop in Suppression Pool Cooling, **THEN PERFORM** the following:

    a. **CLOSE** *TORUS COOLING ISOL VLV, E11-F024A(B.* ☐

    b. **THROTTLE OPEN** *OUTBOARD INJECTION VLV, E11-F017A(B),* until the selected SRV opens. ☐

13. **IF** reactor pressure can **NOT** be maintained less than 164 psig above Suppression Chamber pressure, **THEN PLACE** another SRV control switch to *OPEN*. ☐

3.0 OPERATOR ACTIONS

14. **PERFORM** the following as necessary to maintain cool down rate less than 100°F per hour:

    a. **THROTTLE CLOSE** the injection valve on the affected pump until the desired SRV closes. ☐

    b. **RECORD** reactor pressure at which the SRV closes. ☐

       _________ psig

    c. **THROTTLE OPEN** the injection valve on the affected pump until the SRV reopens. ☐

    d. **THROTTLE CLOSE** the injection valve on the affected pump until reactor pressure is 10 to 20 psig greater than the pressure at which the SRV closed in Step 3.2.14.14.b. ☐

    e. **IF** it is desired to adjust the cool down rate, **THEN CLOSE** the open SRV **AND OPEN** the next SRV that will adjust the cool down rate in the desired direction. ☐

15. **REPEAT** Step 3.2.14.14 until vessel coolant and Suppression Pool temperature are within 100°F. ☐

16. **CONTROL** Suppression Pool temperature as necessary to maintain vessel coolant temperature above 75°F. ☐

3.0 OPERATOR ACTIONS

17. **WHEN** a normal method of Shutdown Cooling can be established, **THEN SHUT DOWN** alternate Shutdown Cooling as follows:

    a. **STOP** the ECCS pump(s) used for vessel injection. ☐

    b. **WHEN** the SRV(s) that were opened have closed, **THEN PLACE** the control switch for the SRV(s) to *CLOSE* **OR** *AUTO*. ☐

    c. **IF** the reactor coolant temperature is less than 212°F, **THEN OPEN** the following valves:

    - *INBOARD RX HEAD VENT VLV, B21-F003* ☐
    - *OUTBOARD RX HEAD VENT VLV, B21-F004*. ☐

R21

CAUTION

IF reactor coolant temperature is greater than 212°F, **AND** reactor water level has been raised to greater than 218 inches for 10 minutes or more, **THEN** a false RPV low level signal could result when the reference leg condensing pot N12A(B) nozzle is uncovered as level is subsequently lowered below 218 inches.

    d. **RESTORE AND MAINTAIN** reactor water level between 200" and 220", or as directed by the Shift Superintendent, based on plant conditions. ☐

    e. **WHEN** directed by the Shift Superintendent, **THEN SHUT DOWN** the RHR loop used for Suppression Pool Cooling in accordance with 1(2)OP-17. ☐

51. 295023AA104 001

Unit Two (2) is performing refueling operations when the refueling SRO reports that a spent fuel bundle has been dropped in the cattle chute. Radiation monitoring alarm status as follows:

AREA RAD REFUEL FLOOR HIGH (UA-03 3-7) is in alarm.
PROCESS RX BLDG VENT RAD HI (UA-03 4-5) is in alarm.

Which ONE of the following operator actions are required to be performed immediately per AOP-5.0, RADIOACTIVE SPILLS, HIGH RADIATION, AND AIRBORNE ACTIVITY, in response to this event?

A. Verify Group 6 isolation.

B. Evacuate all personnel from the refuel floor.

C✓ Place Control Room Emergency Ventilation System (CREVS) in operation.

D. Isolate Reactor Building Ventilation and place Standby Gas Treatment (SBGT) trains in operation.

Feedback

REFERENCE AOP-5.0
Immediate Actions - IF a fuel assembly was dropped or damaged, **THEN ENSURE** the Control Room Emergency Ventilation System (CREVS) is in operation.

DISTRACTOR ANALYSIS
A. INCORRECT - GROUP 6 requires *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) in alarm.
B. INCORRECT - AOP-5.0 does not require that refuel floor be evacuated immediately with above indications.
C. CORRECT - see above
D. INCORRECT - RBHVAC isolation and SBGT start requires *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) in alarm.

Notes

APE: 295023 Refueling Accidents
AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS :
(CFR: 41.7 / 45.6)
AA1.04 Radiation monitoring equipment........................ 3.4 3.7

This question matches the k/a in that it measures the RO's ability to understand radiation alarms and system operation associated with a refueling accident.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-302-J*008
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-5
? Cognitive Level:	M OR FK	? Source:	NEW

1.0 SYMPTOMS

1.1 *AREA RAD REFUEL FLOOR HIGH* (UA-03 3-7) is in alarm.

1.2 *AREA RAD NEW FUEL STORAGE HIGH* (UA-03 4-7) is in alarm.

1.3 *PROCESS RX BLDG VENT RAD HI* (UA-03 4-5) is in alarm.

1.4 Area Radiation Monitor (ARM) is in alarm.

1.5 Continuous Air Monitor (CAM) is in alarm.

1.6 Routine surveys indicate high radiation, contamination and/or airborne activity.

1.7 Report of spill, leak, or potential damage to new or spent fuel.

2.0 AUTOMATIC ACTIONS

2.1 **IF** *PROCESS RX BLDG VENT RAD HI-HI* (UA-03 3-5) is in alarm, **THEN** the following actions occur:

- Reactor Building Ventilation isolation ☐
- SBGTS auto start ☐
- Group 6 Isolation. ☐

3.0 OPERATOR ACTIONS

3.1 Immediate Actions

R1 3.1.1 **IF** a fuel assembly was dropped or damaged, **THEN ENSURE** the Control Room Emergency Ventilation System (CREVS) is in operation. ☐

52. 295024EK301 001

Following a line break in the drywell, Unit One (1) conditions are:

Drywell pressure	20 psig
Drywell temperature	250°F
Suppr chamber pressure	18.5 psig
Suppr chamber level	-27"

Which ONE of the following identifies why EOP-02-PCCP directs initiation of drywell spray under these conditions?

A. To prevent operation of suppression chamber to drywell vacuum breakers.

B. To reduce drywell pressure below the Pressure Suppression Pressure Limit.

C✓ To redistribute non condensibles from the suppression chamber to the drywell.

D. To lower drywell temperature as the design temperature limit has been exceeded.

Feedback

RANDOMLY SELECTED FROM LOI BANK - LOI-CLS-LP-300-L*04B 001

REFERENCE OI-37.8 REV. 3 PAGES 26 AND 27
Drywell design temp limit is 300°F (Torus is 220°F). Conditions are safe in PSPL. Drywell spray forces operation of torus-drywell vacuum breakers to force non condensible gases from torus to drywell to preclude chugging. **Provide DWSIL and PSPL curves as a reference.**

Notes

EPE: 295024 High Drywell Pressure

EK3. Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE :
(CFR: 41.5 / 45.6)
EK3.01 Drywell spray operation: Mark-I&II.................. 3.6 4.0

This question matches the k/a in that it measures the RO's knowledge of why the containment is sprayed with a high drywell pressure condition.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-300-L*04B
Ref Req'd Y or N:	Y DWSIL & PSPL	Technical Ref.:	OI-37.10
? Cognitive Level:	C/A	? Source:	BANK LOI

STEPS PC/P-04 through PC/P-07 (continued)

The Suppression Chamber Spray Initiation Pressure is defined to be the lowest suppression chamber pressure which can occur when 95% of the noncondensibles in the drywell have been transferred to the airspace of the suppression chamber. This pressure is utilized to preclude chugging: the cyclic condensation of steam at the downcomer openings of the drywell vents.

When a steam bubble collapses at the exit of the downcomers, the rush of water filling the void (some of it drawn up into the downcomer pipe) induces a severe stress at the junction of the downcomer and the vent header. Repeated application of this stress can cause these joints to experience fatigue failure (i.e., crack) thereby creating a pathway which bypasses the pressure suppression function of the containment. Subsequent steam discharges through the downcomers would directly pressurize the suppression chamber airspace rather than being discharged to and condensed in the suppression pool.

Scale model tests have demonstrated that chugging will not occur so long as the drywell atmosphere contains at least 1% noncondensibles. To preclude the occurrence of conditions under which chugging may happen, the Suppression Chamber Spray Initiation Pressure is conservatively defined by specifying 5% noncondensibles.

Although operation of suppression pool sprays may not, by itself, preclude chugging, suppression pool sprays are initiated before reaching the Suppression Chamber Spray Initiation Pressure (11.5 psig) to assure that operation of this system is attempted for reducing primary containment pressure before operation of drywell sprays is directed.

The operation of suppression pool sprays is terminated when suppression chamber pressure decreases to 2.5 psig to assure that primary containment pressure is not reduced below atmospheric. Maintaining a positive suppression chamber pressure precludes air from being drawn in through the vacuum relief system to deinert the primary containment, and also assures that a positive margin to the negative design pressure of the primary containment exists.

It is acceptable to use drywell pressure instead of suppression chamber if the suppression chamber instruments are not available. It should be noted however that during transient conditions; i.e., a steam leak in drywell, drywell pressure may be significantly higher than suppression chamber pressure.

The NPSH (Net Positive Suction Head) limits are defined to be the highest suppression pool temperature which provides adequate net positive suction head for pumps taking suction on the pool. The NPSH Limits are functions of pump flow and suppression chamber overpressure (airspace pressure plus the hydrostatic head of water over the

STEPS PC/P-04 through PC/P-07 (continued)

pump suction). It is utilized to preclude pump damage from cavitation. It should be noted that containment pressurization of up to 5 psig is credited for maintaining NPSH margins for BNP. Therefore, as actions are taken that reduce suppression chamber pressure (i.e. suppression pool cooling, containment sprays), pump NPSH requirements should be considered, and closer attention directed towards observing the performance of the RHR and Core Spray pumps for signs of NPSH problems.

The NPSH limit is addressed through a caution for the following reasons:

a. It is difficult to define in advance exactly when the limits should be observed and when pumps should be operated irrespective of the limits.

b. Pumps to which the limits apply are used in more than one parameter control path, or in different procedures. RHR pumps, for example, may be used in the Reactor Vessel Control Procedure, the Level/Power Control procedure, or the Primary Containment Control Procedure. Authorizing operation of the pumps irrespective of NPSH in one path may conflict with instructions in another path where flow would normally be controlled below the limits.

The identified systems should be operated within the NPSH and vortex limits if possible. If the situation warrants, however, the limits may be exceeded. A judgment as to whether a pump should be operated beyond its limits in a particular event should consider such factors as:

a. The availability of other systems

b. The current trend of plant parameters

c. The anticipated time such operation will be required

d. The degree to which the limit will be exceeded

e. The sensitivity of the pump to operation beyond the limit

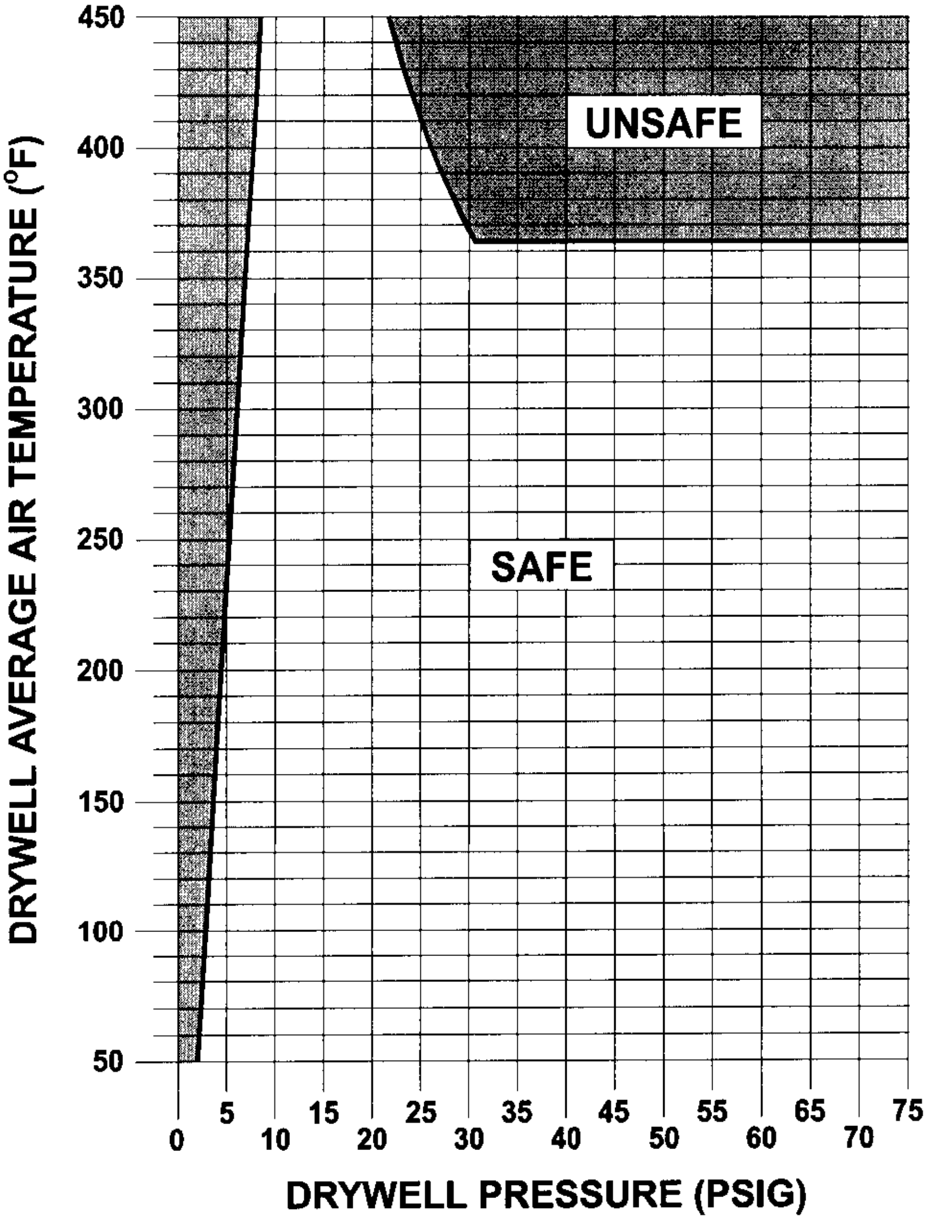
f. The consequences of *not* operating the pump beyond the limit

Immediate and catastrophic failure is not expected if a pump is operated beyond the NPSH or vortex limit.

Suppression chamber sprays are initiated per Suppression Pool Spray Procedure (EOP-01-SEP-03).

ATTACHMENT 5 (Cont'd)

FIGURE 1
DRYWELL SPRAY INITIATION LIMIT

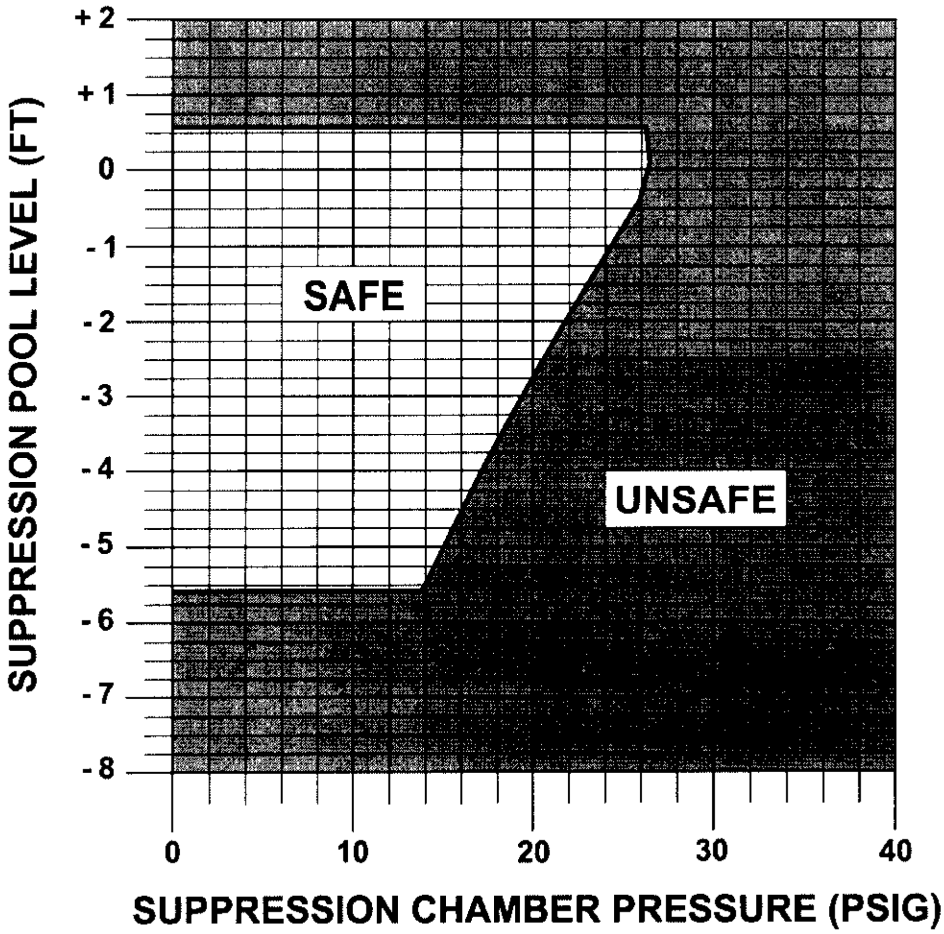


NOTE

DRYWELL AVERAGE AIR TEMPERATURE MAY BE DETERMINED
USING ATTACHMENT 4 OF THE "USER'S GUIDE"

ATTACHMENT 5 (Cont'd)

FIGURE 7
PRESSURE SUPPRESSION PRESSURE



53. 295025EK1.01 001

Unit One (1) is operating at rated power when Main Steam Isolation Valve B21-F022A inadvertently isolates.

Which ONE of the following describes the response of the reactor to this action?

The MSIV closure will result in:

A. essentially no change in steam flow as MSIV minimum stroke time is set so that a single MSIV closure will result in essentially no change in reactor pressure and power.

B.✓ a sudden reduction in steam flow which will cause reactor pressure to increase which causes a collapse of voids in the core. This adds positive reactivity and reactor power will increase. The reactor may scram on either high flux or high pressure.

C. essentially no change in steam flow as the turbine control valves quickly open in response to the steam line pressure transient. The remaining three steam lines are sized to accomodate power operation and as a result the transient will have a neglible affect on reactor power.

D. a sudden reduction in steam flow which will cause a shrink in reactor water level. This causes the reactor feedwater system to rapidly respond to restore water level and the feedwater injection will add positive reactivity so reactor power increases. The reactor may scram if an overfeed event causes a turbine trip.

Feedback

REFERENCE CLS-LP-110-A REV. 0 Pages 18, 20, 24, and 32

DISTRACTOR ANALYSIS
A. INCORRECT - MSIV minimum stroke time is set to minimize a pressure transient but not to make it have essentially no affect. MSIVs must close w/in 5 sec for DBA MSLB.
B. CORRECT
C. INCORRECT - EHC cannot respond fast enough
D. INCORRECT - level shrink is a result of collapsing voids which are caused by sudden reduction in steam flow with no immediate corresponding decrease in reactor power

Notes

EPE: 295025 High Reactor Pressure

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE :
(CFR: 41.8 to 41.10)
EK1.01 Pressure effects on reactor power..................... 3.9 4.0

This question matches the k/a in that it measures the RO's knowledge of a pressurization event and it's affect on reactor power.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.4	Facility Objective:	CLS-LP-110A*3D
Ref Req'd Y or N:	NO	Technical Ref.:	CLS-LP-110A
? Cognitive Level:	C/A	? Source:	NEW

E. PLANT PARAMETERS

> **NOTE:** The parameters listed in this material for Unit 1 and Unit 2 do not necessarily illustrate current BNP parameters in every case. The figures attached to this material were extracted from a nuclear engineering software program that is no longer being maintained however the figures themselves are still suited for learning the mechanics of Anticipated Operational Occurrences. Actual current plant parameters and transient response data may be found in Technical Specifications, UFSAR, COLR, and/or other controlled plant documents as applicable.

1. Reactor Pressure

Pressure is proportional to the energy of the steam leaving the reactor in a saturated system. Due to EHC characteristics, pressure is also affected by the EHC pressure setpoint and load. Finally, pressure in the steam dome is a function of steam flow, since pressure is controlled at a point significantly downstream from the reactor.

Any event, which causes an increase in core power without a corresponding immediate increase in steam flow, will result in a pressure rise. Conversely, any event, which causes a decrease in steam flow without a corresponding immediate decrease in core power, will cause an increase in vessel pressure. The reverse of the two situations above also applies. Examples of this principle would include main steam isolation valve (MSIV) closure (pressure rise) and safety/relief valve (SRV) lift (pressure drop).

The EHC System controls turbine control valve (TCV) position based on a pressure signal sensed at the transducer header between the MSIVs and TCVs. The system is set to achieve 0% to 100% power over a sensed 0 to 30 psi rise above the pressure setpoint. As a result, pressure and thus power, as measured by transducer header pressure, controls TCV position and, consequently, generator output.

As an example, if the EHC pressure setpoint is 945 psig and reactor power is 50%, actual transducer pressure would be 960 psig. An increase in reactor power to 75% would cause a pressure rise. TCVs would open to pass 75% steam flow to the turbine (neglecting non-main turbine flow), and header pressure would stabilize at 367.5 psig. It can be seen from the above that two EHC characteristics affect reactor vessel pressure. As the pressure setpoint is changed, EHC will attempt to maintain the new pressure by adjustment of TCVs, and as power rises, the EHC droop characteristic by itself, would cause a small rise in EHC transducer header pressure.

The other major parameter that affects void fraction is reactor pressure. A rise in pressure tends to suppress nucleate boiling; this causes a decrease in average void fraction and consequently, an increase in reactor power. This mechanism of power response is extremely important to transient analysis; the largest reduction in MCPR occurs as a result of turbine trip with bypass failure, resulting in a pressure rise and an analyzed worst-case neutron flux peak of several hundred percent.

Pressure transients present the BWR core with the only major potential instability. As pressure rises, core power will rise, which tends to cause a further pressure rise, leading to still higher power. High flux, high pressure, MSIV closure, and turbine trip scrams protect the core from damage.

Examples of transients for which the major plant response characteristics are driven by the void coefficient of reactivity would include:

- Reactor recirculation pump trip.
- MSIV closure.
- Pressure regulator failure.

b. Moderator Temperature Coefficient

Moderator temperature coefficient is second in magnitude affecting reactor power level. The mechanism for power change is the same as that for voids. As moderator temperature decreases, density increases and power rises. In an operating BWR, at power, (from 10% to 100%) moderator temp. changes very little. Although reactor pressure changes approximately 100 psig in going from 10% power to 100% power 945 to 1030 psig in a saturated environment, temperature changes only 15°-16°F. However, factors external to the reactor may cause significant changes in moderator temperature. The major system, which adds water to the vessel, the feedwater system, can have a marked effect on moderator temperature. At full power, the feedwater enters the vessel at a nominal 425.2°F.

Preheating of feedwater is accomplished in the condensate and feedwater chains by routing the water through heaters heated by main turbine extraction steam. Loss of feedwater heating, results in cooler feedwater entering the vessel, thus reducing moderator temperature, and causing a power rise. A similar effect is seen if the quantity of relatively cool water, (feedwater at 425.2°F compared to reactor water at 545°F) entering the vessel increases. This can arise from overfeeding with the reactor feedwater pumps, or with inadvertent HPCI or RCIC startup. Unplanned startup of low pressure ECCS Systems is not normally a threat, because discharge pressures of low pressure pumps is normally at least 500 psig below reactor vessel pressure and water cannot get into the vessel.

5. Total Steam Flow

   Total steam flow is affected by surface heat flux and changes in the steam flow paths including main steam isolation valves (MSIVs), turbine control valves (TCVs), turbine stop valves (TSVs), and bypass valves. When the core is generating steam, steam flow is regulated by the Pressure Regulator, using vessel pressure as a controlling parameter. The Pressure Regulator is normally capable of regulating pressure for all normal evolutions.

6. Feedwater Flow

   Feedwater flow is affected by the FWLC system. In three element control it senses a change in steam flow to anticipate the need for feed flow, the system also senses reactor water level and tries to maintain level set.

B. TRANSIENT CLASSIFICATION

Transients are classified by the original parameter they change. In instances where several parameters change, the transients are categorized with the group that has the most similar plant response characteristics.

The transient classification groups, and transients in these groups, are as follows:

1. Decreases in Reactor Coolant Temperature

   Events that result in a reactor vessel water temperature decrease are those that either increase the flow of cold water to the reactor, reduce the temperature of water delivered to the reactor, or increase the rate of heat removal from the reactor. Events analyzed in this group include:

   a. Loss of Feedwater Heating

   b. Feedwater Controller Failure – Maximum Demand

   c. Inadvertent HPCI or RCIC Pump Start

   d. Pressure Regulator Failure – Open

   e. Inadvertent Safety/Relief Valve Opening (Inadvertent Opening of a Relief Valve or Safety Valve

   f. Inadvertent RHR Operation (Inadvertent RHR Shutdown Cooling Operation)

(11) The SRVs cycle to remove the heat generated by the shut down core.

(12) The reduction in reactor water level causes the feedwater controller to increase feedwater flow. As reactor water level recovers, feedwater flow begins to decrease.

4. Turbine Trip Without Bypass Valves at Low Power (<30%)

Reactor scrams associated with turbine trips are bypassed when reactor power is less than 30% as sensed by 1st stage steam pressure. Since the Bypass valves are assumed to fail after the turbine trip, reactor pressure will begin to rise until pressure reaches or 1060 psig at which point a reactor scram will occur. The pressure will continue to increase until SRV lifting to relieve the pressure transient. This transient is less severe than the previously discussed turbine trip transient.

5. MSIV Fast Closure

Automatic circuitry or operator action can initiate a closure of the main steam isolation valves (MSIVs), which must close in 3 to 5 seconds according to the Technical Specifications. The worst case of a 3-second closure time is assumed in this analysis. Position switches on the valves initiate a reactor scram when the valves are less than 90% open. Closure of the MSIV inhibits steam flow to the feedwater turbine, eventually terminating feedwater flow. MSIV closure can also indirectly cause a trip of the main turbine and generator, but late enough that there is no impact on the transient.

With the MSIV inhibiting steam flow, reactor pressure increases and energy is released via the SRVs, pressure increases above the Recirc pump ATWS trip. Reactor water level decreases due to the collapsing of the voids and the loss of feedwater. The level decrease causes a HPCI/RCIC start and Recirculation Pump Trip ATWS.

MSIV Fast Closure Event Summary (Figures 6a, 6b, 6c)

(1) MSIVs closes to 90% open and a reactor scram is initiated decreasing power.

(2,3) AS the MSIVs closes total steam flow is restricted increasing reactor pressure until relief valves lift.

(4) Increased reactor pressure decreases feed flow.

(5,6) Core flow and loop flow increases due to decreased voids from the reactor scram and increased pressure.

(7,8) Reactor water level decreases due to the decrease in void concentration.

54. 295026EK304 001

During an ATWS transient in which heat is being added to the Suppression Pool, Standby Liquid Control (SLC) is initiated prior to reaching the Boron Injection Initiation Temperature Limit.

Which ONE of the following identifies the basis for this action?

This action assures that the Hot Shutdown Boron Weight is injected before:

A. the dynamic load limit of the Suppression Pool is exceeded.

B. the Suppression Pool Pressure Suppression Limit is exceeded.

C✓ the Suppression Pool Heat Capacity Temperature Limit is exceeded.

D. Suppression Pool temperature reaches 140°F which limits HPCI operation to suction from the CST only.

Feedback

REFERENCE - OI-37.5 REV. 6 PAGE 81 and 82

DISTRACTOR ANALYSIS - All distractors are homogeneous suppression pool related limits but only Heat Capacity Temperature limit is related to SLC injection.
A. INCORRECT - Formerly called SRV Tailpipe limit, has more to do with Torus level and is included in PSPL.
B. INCORRECT - Indicates failure of containment response and torus temp does not appear on this graph.
C. CORRECT
D. INCORRECT - LPC procedure includes provisions to defeat HPCI automatic suction transfer logic provided suppression pool temperature is approaching 140°F. Prior to approaching this temperature HPCI is run on the suppression pool to preclude raising the water level in the suppression pool.

Notes

EPE: 295026 Suppression Pool High Water Temperature

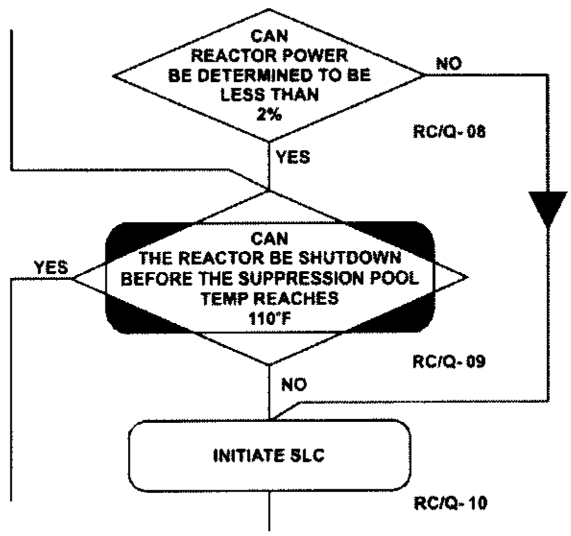
EK3. Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:
(CFR: 41.5 / 45.6)
EK3.04 †SBLC injection....................................... 3.7 4.1*

This question matches the k/a in that it measures the RO's knowledge of why SLC is injected under conditions of high suppression pool temperature.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.7	Facility Objective:	CLS-LP-300-E*14D
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.5
? Cognitive Level:	M OR FK	? Source:	NEW

STEPS RC/Q-08 through RC/Q-10



STEP BASES:

If reactor power is above 2%, the operator is directed to inject boron. This is a conservative action because with power above 2%, Suppression Pool temperature will steadily increase towards 110°F. This also allows sufficient time for the Hot Shutdown Boron Weight of boron to be injected. The extra time may be needed since the alternate systems used for boron injection require significantly more time to inject boron should the SLC System fail. The SLC system is initiated to shut down the reactor.

As long as the core remains submerged (the preferred method of core cooling), fuel integrity and reactor vessel integrity are not directly challenged even under failure-to-scram conditions. A scram failure coupled with an MSIV isolation; however, results in rapid heatup of the Suppression Pool due to the steam discharged from the reactor vessel via SRVs. The challenge to containment thus becomes the limiting factor which defines the requirement for boron injection.

If Suppression Pool temperature and reactor pressure cannot be maintained below the Heat Capacity Temperature Limit, rapid depressurization of the reactor vessel will be required. To avoid depressurizing the reactor vessel with the reactor at power, it is desirable to shut down the reactor prior to reaching the Heat Capacity Temperature Limit, thus minimizing the quantity of heat rejected to the Suppression Pool. The Boron Injection Initiation Temperature is defined so as to achieve this when practicable.

STEPS RC/Q-08 through RC/Q-10 (continued)

The Boron Injection Initiation Temperature is defined to be the greater of:

a. The Suppression Pool temperature at which initiation of a reactor scram is required by Technical Specifications, or

b. The highest Suppression Pool temperature at which initiation of boron injection using SLC will result in injection of the Hot Shutdown Boron Weight of boron before Suppression Pool temperature exceeds the Heat Capacity Temperature Limit.

Criterion b is a function of reactor power; a higher reactor power level causes higher integrated heat energy to be rejected to the Suppression Pool thus requiring a lower Suppression Pool temperature for initiation of boron injection if the Heat Capacity Temperature Limit is not to be exceeded before reactor shut down is achieved.

At Brunswick, a single value is used for Boron Injection Initiation Temperature (110°F) for procedure simplification.

55. 295028G2.2.25 001

Which ONE of the following identifies the bases for the Drywell Average Air Temperature Limiting Condition for Operation (LCO)?

In the event of a DBA, initial drywell average air temperature is assumed to be less than or equal to:

A. 135°F so that the resultant peak accident temperature is maintained below 300°F. This ensures that the primary containment will be able to perform its design function.

B. 135°F so that the resultant peak accident temperature is maintained below 340°F. This ensures that the Safety Relief Valves (SRVs) are able to perform their design function.

C✓ 150°F so that the resultant peak accident temperature is maintained below 300°F. This ensures that the primary containment will be able to perform its design function.

D. 150°F so that the resultant peak accident temperature is maintained below 340°F. This ensures that the Safety Relief Valves (SRVs) are able to perform their design function.

Feedback

The following bank question was modified
CLS-LP-004*018 001
Unit Two (2) is operating at rated power. In accordance with Technical Specification LCO 3.6.1.4:
A. Drywell average air temperature shall be ≤135°F.
B. Drywell average air temperature shall be ≤150°F.
C. Primary Containment average air temperature shall be ≤135°F.
D. Primary Containment average air temperature shall be ≤150°F.

REFERENCE TS BASES REV. 30 PAGE B3.6.1.4-1
In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured.

DISTRACTOR ANALYSIS
A - INCORRECT - Previous version of TS LCO based on PC Avg. Temp of 135°F
B - INCORRECT - Previous version of TS LCO based on PC Avg. Temp of 135°F, LCO not based on SRV operability.
C - CORRECT
D - INCORRECT - LCO not based on SRV operability

Notes

EPE: **295028 High Drywell Temperature**
2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.
(CFR: 43.2)
IMPORTANCE RO 2.5 SRO 3.7

This question matches the k/a in that it measures the RO's knowledge of the drywell average air temperature LCO bases.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 2.5	Facility Objective:	CLS-LP-004*002
Ref Req'd Y or N:	NO	Technical Ref.:	TS BASES
? Cognitive Level:	M OR FK	? Source:	MOD. LOI BANK

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Air Temperature

BASES

BACKGROUND	The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 and 2 safety analyses.
APPLICABLE SAFETY ANALYSES	Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Refs. 1 and 2). Among the inputs to the design basis analysis is the initial drywell average air temperature (Refs. 1 and 2). Analyses assume an initial average drywell air temperature of 150°F. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 300°F (Ref. 3). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads. Equipment inside primary containment required to mitigate the effects of a DBA is designed to operate and be capable of operating under environmental conditions expected for the accident. Drywell air temperature satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 4).
LCO	In the event of a DBA, with an initial drywell average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the drywell design temperature. As a result, the ability of primary containment to perform its design function is ensured.

(continued)

56. 295030EA105 001

Which ONE of the following is a limitation place on the High Pressure Coolant Injection (HPCI) System associated with Suppression Pool Level?

HPCI must be secured irrespective of adequate core cooling concerns if Suppression Pool Level:

A. decreases to -5.5 feet.

B.✓ decreases to -6.5 feet.

C. increases to +6 inches.

D. increases to +21 inches.

Feedback

REFERENCE OI-37.8 REV. 4 PAGE 48

DISTRACTOR ANALYSIS - All distractors are limits associated with Supp Pool Level (i.e. homogeneous)
A. INCORRECT - corresponds to downcomer vent openings SCRAM and ED per PCCP
B. CORRECT
C. INCORRECT - bottom of supp. pool to DW vacuum breakers, terminate DW sprays per PCCP
D. INCORRECT - ring header elevation, ED required per PCCP

Notes

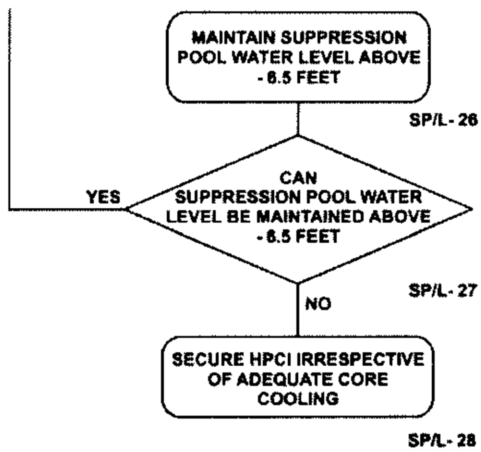
EPE: 295030 Low Suppression Pool Water Level
EA1. Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:
(CFR: 41.7 / 45.6)
EA1.05 HPCI.. 3.5 3.5

This question matches the k/a in that it measures the RO's knowledge of HPCI operating limitations with low suppression pool water level.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-300-L*014
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.8
? Cognitive Level:	M OR FK	? Source:	NEW

STEPS SP/L-26 through SP/L-28



STEP BASES:

Operation of the HPCI System with its exhaust discharge device not submerged will directly pressurize the suppression chamber. HPCI operation is therefore secured as required to preclude the occurrence of this condition. The consequences of not doing so may extend to failure of the primary containment from overpressurization, and thus HPCI must be secured irrespective of adequate core cooling concerns.

No instruction regarding RCIC operation is included in this step (or in an equivalent step) for two reasons:

a. The exhaust flow rate of RCIC is approximately equal to that of decay heat, and is thus consistent with the basis used for determining the Primary Containment Pressure Limit.

b. Elevated suppression chamber pressure will cause the RCIC turbine to trip much before the HPCI turbine would trip.

57. 295031EA204 001

During an accident, plant conditions are:

RPV pressure	25 psig
Core Spray B	Injecting to the RPV @ 5500 gpm
RHR Loop B	Suppression Pool cooling @ 11,500 gpm

Which ONE of the following is the LOWEST RPV water level that still assures adequate core cooling is being maintained?

A. TAF

B. LL4

C. LL5

D✓ -57.5".

Feedback

Randomly selected from BANK - LOI-CLS-LP-300-B*008 012

REFERENCE EOP-01-UG ACC DEFINITION - Adequate core cooling exists per EOP-UG if RPV level is at the jet pump suction with Core Spray injecting at @ 4700 gpm. Jet pump suction elevation is @ -59", specified in RVCP as -57.5 for instrument readability.

DISTRACTOR ANALYSIS
A., B. INCORRECT TAF and LL4 are above -57.5"
C. INCORRECT - assumes no injection source
D. CORRECT

Notes

EPE: 295031 Reactor Low Water Level

EA2. Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL :

(CFR: 41.10 / 43.5 / 45.13)
EA2.04 Adequate core cooling.......................... 4.6* 4.8*

This question matches the k/a in that it measures the RO's ability to determine if adequate core cooling exists with low RPV level.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 4.6	Facility Objective:	CLS-LP-300-B*008
Ref Req'd Y or N:	NO	Technical Ref.:	EOP-01-UG
? Cognitive Level:	M OR FK	? Source:	BANK LOI

ATTACHMENT 5
DEFINITIONS

ADEQUATE CORE COOLING

Heat removal from the reactor sufficient to prevent rupturing the fuel clad.

Four viable mechanisms of adequate core cooling exist within the EOPs:

a. Core submergence

b. Steam cooling with injection of makeup water to the reactor

c. Steam cooling without injection of makeup water to the reactor

d. Reactor water level at jet pump suction with at least one core spray pump injecting into the reactor vessel at 4700 gpm.

AFTER

Following in time or place.

ANTICIPATED TRANSIENT WITHOUT SCRAM

The reactor is not shutdown following a scram.

ALTERNATE INJECTION SYSTEMS

Systems which may be used to inject water to the reactor when the injection systems cannot supply sufficient injection water to the vessel. They are as follows:

a. Service Water

b. Fire Protection System

c. Demineralized water via ECCS Keepfill System

d. SLC System (boron solution or demineralized water)

e. Condensate Transfer System

f. Heater Drains System

APPROACHES

The value of an identified parameter is drawing near to a specified limit. In use, "approaches" is similar to "before", but indicates that action is to be delayed until the margin to the limit is small. If the limit has already been exceeded when the instruction is reached, the action should still be performed unless expressly prohibited.

58. 295032EA101 001

Unit Two (2) is operating at rated power with the ERFIS computer out of service. The RO is performing PT 9.2, HPCI OPERABILITY TEST when an AO reports that a loud noise can be heard from inside the mini-steam tunnel and the following blue bar annunciator alarms:

STM LEAK DET AMBIENT TEMP HIGH A-02 (5-7)

Assuming that a HPCI steam supply line break has just occurred the RO immediately isolates HPCI manually. There are no indications that any automatic isolation setpoints were exceeded.

Which ONE of the following identifies the availability of instrumentation that can be used by the RO to validate that the HPCI steam leak has been isolated?

Trend data to verify HPCI Steam Tunnel Area Temperature is lowering can:

A. be obtained at the Process Computer.

B. NOT be obtained since ERFIS is out of service.

C✓ be obtained at the NUMAC monitors on H12-P614.

D. be obtained at the RHR/HPCI/FPC SYS TEMP Recorder, 2-E41-TR-R605, on H12-P614.

Feedback

REFERENCE - APP A-02 (5-7)

DISTRACTOR ANALYIS
A. INCORRECT - Process Computer is not provided with this data.
B. INCORRECT - NUMAC has a TREND function for all temperatures monitored.
C. CORRECT - Operator must toggle to the TREND data screen.
D. INCORRECT - This recorder contains oil temperatures and other HPCI related temp. but has no leak detection temperature monitoring.

Notes

EPE: 295032 High Secondary Containment Area Temperature

EA1. Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE :
(CFR: 41.7 / 45.6)
EA1.01 Area temperature monitoring system.................... 3.6 3.7

This question matches the k/a in that it measures the RO's ability monitor temperature under conditions of high secondary containment temperature.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.6	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	APP A-02 5-7
? Cognitive Level:	M OR FK	? Source:	NEW

1.0 OPERATOR ACTIONS:

1.1 **CONFIRM** which steam leak area ambient temperature is high by observation of either an Inverse Video "A" Alarm Flag or an "I" Isolate Flag on NUMAC leak Detection Monitors on H12-P614.

1.2 **OBSERVE** Automatic Functions:

1.2.1 **IF** the RWCU Pump Room A, Pump Room B or HX Room ambient temperature is high, **THEN** a Group 3 isolation signal is generated **AND** the RWCU System isolates.

1.2.2 **IF** the RWCU Outside Pump/HX Room ambient temperature is high, **THEN** a Group 3 isolation signal is generated **AND** the RWCU System isolates.

1.2.3 **IF** the HPCI Equipment Area ambient temperature is high, **THEN** a Group 4 isolation signal is generated **AND** the HPCI System isolates.

1.2.4 **IF** the HPCI Steam Line Area ambient temperature is high, **THEN** a Group 4 isolation signal is generated **AND** the HPCI System isolates.

1.2.5 **IF** the HPCI Steam Line Tunnel ambient temperature is high, **THEN** a Group 4 isolation signal is generated **AND** the HPCI System isolates.

1.2.6 **IF** the RCIC Equipment Area ambient temperature is high, **THEN** a Group 5 isolation signal is generated **AND** the RCIC System isolates.

1.2.7 **IF** the RCIC Steam Line Area ambient temperature is high on NUMAC Steam Leak Detection Monitor, *B21-XY-5949A* Channel A6-4 or *B21-XY-5949B* Channel A4-4, after a 27 minute time delay, **THEN** a Group 5 signal is generated **AND** the RCIC System isolates.

1.2.8 **IF** the RCIC Steam Line Area ambient temperature is high on any of the other NUMAC channels, **THEN** a Group 5 signal is generated **AND** the RCIC System isolates without a time delay.

1.2.9 **IF** the RCIC Steam Line Tunnel ambient temperature is high, after a 27-minute time delay, **THEN** a Group 5 signal is generated **AND** the RCIC System isolates.

CAUTION

An ECCS Room steam leak of approximately 3000 lb/hr to 7300 lb/hr may actuate the local sprinkler head and cool the room below steam leak detection instrumentation setpoints. RCIC isolation may **NOT** occur automatically **AND** manual isolation may be required.

1.3 **PERFORM** Corrective Actions:

1.3.1 **ENTER** 0EOP-03-SCCP, Secondary Containment Control Procedure.

1.3.2 **MONITOR** the steam leak detection temperatures on Control Panel H12-P614 and ERFIS to determine the cause.

NOTE: 0PIC-T/C001, Calibration Check of Type T Thermocouples contains temperature element locations.

2.0 CAUSES:

2.1 RWCU Pump Room A, RWCU Pump Room B or HX room high ambient temperature (140°F).

2.2 RWCU outside pump/HX rooms high ambient temperature (115°F).

2.3 HPCI equipment area high ambient temperature (165°F).

2.4 HPCI steam line area high ambient temperature (190°F).

2.5 HPCI steam line tunnel high ambient temperature (190°F).

2.6 RCIC equipment area high ambient temperature (165°F).

2.7 RCIC steam line area high ambient temperature (190°F).

2.8 RCIC steam line tunnel high ambient temperature (165°F).

2.9 HPCI equipment area high ambient temperature (155°F - no auto action).

2.10 RCIC equipment area high ambient temperature (155°F - no auto action).

2.11 TB main steam tunnel high ambient temperature (170°F - no auto action).

2.12 North RHR area cooler high ambient temperature (175°F - no auto action).

2.13 South RHR area cooler high ambient temperature (175°F - no auto action).

3.0 DEVICES: **SETPOINT:**

NUMAC STEAM LEAK DETECTION MONITOR B21-XY-5948A CHANNEL DATA FOR CHANNELS THAT INPUT TO ANNUNCIATOR A-02 5-7				
CHANNEL NO.	INPUT THERMOCOUPLE TAG	ISO VALVE GROUP	CHANNEL FUNCTION	SET POINT (°F)
A1-1	E41-TE-N030A	4	HPCI STM LK DET AMB T	165
A2-1	E41-TE-N024	NA	HPCI EQUIP AREA AMB T	155
A2-2	E41-TE-3314	4	HPCI STM LK DET AMB T	165
A3-2	E41-TE-3316	4	HPCI STM LK DET AMB T	165
A3-4	B21-TE-N014	NA	TB MAIN STM TUNNEL AMB T	170
A4-2	E41-TE-3318	4	HPCI STM LK DET AMB T	165
A5-1	E51-TE-N025C	4	HPCI STM LK DET AMB T	190
A5-2	E41-TE-3488	4	HPCI STM LK DET AMB T	165
A5-4	E11-TE-N009A	NA	NORTH RHR AREA CLR AMB T	175

NUMAC STEAM LEAK DETECTION MONITOR *B21-XY-5949A* CHANNEL DATA FOR CHANNELS THAT INPUT TO ANNUNCIATOR A-02 5-7				
CHANNEL NO.	INPUT THERMOCOUPLE TAG	ISO VALVE GROUP	CHANNEL FUNCTION	SET POINT (°F)
A1-1	G31-TE-N016A	3	RWCU PUMP RM A AMB T	140
A1-2	G31-TE-5931	3	RWCU OUTSIDE PUMP/HX RMS	115
A1-3	E51-TE-N023A	5	RCIC STM LK DET AMB T	165
A1-4	E51-TE-N011	NA	RCIC EQUIP AREA AMB T	155
A2-1	G31-TE-N016C	3	RWCU PUMP RM B AMB T	140
A3-1	G31-TE-N016E	3	RWCU HX RM AMB T	140
A3-3	E51-TE-N025A	5	RCIC STM LK DET AMB T	190*
A3-4	E51-TE-3319	5	RCIC STM LK DET AMB T	165*
A4-4	E51-TE-3321	5	RCIC STM LK DET AMB T	165
A5-4	E51-TE-3323	5	RCIC STM LK DET AMB T	165
A6-4	E51-TE-3487	5	RCIC STM LK DET AMB T	165*

* 27-minute time delay

NUMAC STEAM LEAK DETECTION MONITOR B21-XY-5948B CHANNEL DATA FOR CHANNELS THAT INPUT TO ANNUNCIATOR A-02 5-7				
CHANNEL NO.	INPUT THERMOCOUPLE TAG	ISO VALVE GROUP	CHANNEL FUNCTION	SET POINT (°F)
A1-1	E41-TE-N030B	4	HPCI STM LK DET AMB T	165
A2-2	E41-TE-3315	4	HPCI STM LK DET AMB T	165
A3-2	E41-TE-3317	4	HPCI STM LK DET AMB T	165
A4-2	E41-TE-3354	4	HPCI STM LK DET AMB T	165
A5-1	E51-TE-N025D	4	HPCI STM LK DET AMB T	190
A5-2	E41-TE-3489	4	HPCI STM LK DET AMB T	165
A5-4	E11-TE-N009B	NA	SOUTH RHR AREA CLR AMB T	175

NUMAC STEAM LEAK DETECTION MONITOR *B21-XY-5949B* CHANNEL DATA FOR CHANNELS THAT INPUT TO ANNUNCIATOR A-02 5-7				
CHANNEL NO.	INPUT THERMOCOUPLE TAG	ISO VALVE GROUP	CHANNEL FUNCTION	SET POINT (°F)
A1-1	G31-TE-N016B	3	RWCU PUMP RM A AMB T	140
A1-2	G31-TE-5932	3	RWCU OUTSIDE PUMP/HX RMS	115
A1-3	E51-TE-N023B	5	RCIC STM LK DET AMB T	165
A2-1	G31-TE-N016D	3	RWCU PUMP RM B AMB T	140
A3-1	G31-TE-N016F	3	RWCU HX RM AMB T	140
A3-3	E51-TE-N025B	5	RCIC STM LK DET AMB T	190*
A4-4	E51-TE-3320	5	RCIC STM LK DET AMB T	165*
A5-4	E51-TE-3322	5	RCIC STM LK DET AMB T	165
A6-3	E51-TE-3355	5	RCIC STM LK DET AMB T	165

* 27-minute time delay

4.0 REFERENCES:

4.1 LL-09364-35

4.2 0EOP-03-SCCP, Secondary Containment Control Procedure

4.3 FP-84127, NUMAC LDM Operations and Maintenance Instructions

4.4 0PIC-T/C001, Calibration Check of Type T Thermocouples

4.5 T.S. 3.3.6.1

4.6 TRM App. B Table 3.3.6.1-1

59. 295037EK206 001

Unit Two (2) is operating at rated power when the Main Turbine trips. Conditions are as follows:

APRM indications	6%
Control Rods	14 NOT FULL IN
Blue scram lights	137 illuminated

SDV HI-HI Level RPS Trip annunciator (A-05 1-6) in alarm

The control rods that are NOT FULL IN are hydraulically stuck due to a common mode failure.

Which ONE of the following identifies the control rod insertion method per EOP-01-LEP-02, Alternate Control Rod Insertion, that would exert the greatest amount of differential pressure across the drive piston of the stuck control rod drive mechanisms?

Control rod insertion:

A.✓ by venting the over piston area.

B. with the individual scram test switches.

C. by increasing the cooling water header pressure.

D. with the Reactor Manual Control System (RMCS).

Feedback

REFERENCE OI-37.1

DISTRACTOR ANALYSIS

A. CORRECT-This action maximizes differential pressure across the drive piston and is a normal practice when control rod drive maintenance is required.

B. INCORRECT - This method would apply the total available differential pressure of the CRD hydraulic system to a single selected control rod. The Scram Discharge Volume vent and drain valves remain open, the maximum differential pressure is applied over the full travel of the control rod. However, this would not be the greatest D/P as venting over piston area to atmosphere is a lower pressure than venting to SDV.

C. INCORRECT - This method increases pressure exerted on the underside of the drive piston but does not vent overpiston pressure.

D. INCORRECT - This method allows maximizing drive pressure by starting both CRD pumps; throttling open Flow Control Valve, C12-F002A (F002B) and, if necessary, throttling closed Drive Pressure Valve,C12-PCV-F003 but does not vent overpiston pressure completely.

Notes

EPE: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

EK2. Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following:
(CFR: 41.7 / 45.8)
EK2.06 CRD mechanisms.. 3.5 3.6

This question matches the k/a in that it measures the RO's knowledge the interrelationship between CRD mechanism construction and ATWS conditions by determining the method of alternate control rod insertion that will exert the greatest D/P.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-300-J*03F
Ref Req'd Y or N:	NO	Technical Ref.:	OI-37.1
? Cognitive Level:	C/A	? Source:	NEW

Alternate Control Rod Insertion

Step: Section 4

Source: PSTG RC/Q-6.1 and RC/Q-6.2

Justification of Difference: Section 4 provides the plant-specific steps required to insert the control rods with the individual scram test switches. The plant-specific steps included in Section 4 are beyond the scope of the PSTG but are required to meet the intent of the PSTG.

Discussion: The purpose of Section 4 is to insert control rods with the individual scram test switches.

Opening individual scram test switches acts on only a single control rod at a time but can be repeated quickly for many rods. If the scram can be reset, this method may be more effective than a full core scram because the total available differential pressure of the CRD hydraulic system is applied to the single selected control rod. Since the Scram Discharge Volume vent and drain valves remain open, the maximum differential pressure is applied over the full travel of the control rod. Also, the rate of water loss from the reactor vessel through the control rod drive mechanism is small.

Step: Section 5

Source: PSTG RC/Q-6.2

Justification of Difference: Section 5 provides the plant-specific steps required to insert control rods with the Reactor Manual Control System (RMCS) defeating RWM interlocks. The plant-specific steps included in Section 5 are beyond the scope of the PSTG but are required to meet the intent of the PSTG.

Discussion: The purpose of Section 5 is to insert control rods with RMCS. This method is best applied when only a few control rods cannot be inserted, alternate methods are being performed which cannot be performed continuously, RPS cannot be reset, or individual control rod scrams are not effective. To assist in driving control rods it is possible to maximize drive pressure by starting both CRD pumps; throttling open Flow Control Valve, C11-F002A (F002B) [C12-F002A (F002B)]; and, if necessary, throttling closed Drive Pressure Valve, C11-PCV-F003 (C12-PCV-F003). Placing the RWM NORMAL/BYPASS switch to "BYPASS" to insert rods defeats the RWM interlocks.

Step: Section 6

Source: PSTG RC/Q-6.2

Justification of Difference: Section 6 provides the plant-specific steps required to insert control rods by venting the over piston area of control rods. The plant-specific steps included in Section 6 are beyond the scope of the PSTG but are required to meet the intent of the PSTG.

Discussion: The purpose of Section 6 is to insert control rods by venting the over piston area of any control rod not inserted to or beyond position 00 (Unit 1 Only) [position 02 (Unit 2 Only)]. This is accomplished for each control rod to be inserted by closing the associated Withdrawal-Riser Isolation Valve, C11-102 (C12-102); removing the vent plug on the riser block for the associated Driver Water Withdrawal Header Vent Valve, C11-F102 (C12-F102); installing a vent hose (routed to floor drain); and opening the associated C11-F102 (C12-F102).

This action maximizes differential pressure across the drive piston and is a normal practice when control rod drive maintenance is required. During an operational event which necessitates use of this method, the discharged liquid could be hot and radioactive. Access is required to the HCU area.

Step: Section 7

Source: PSTG RC/Q-6.1 and RC/Q-6.2

Justification of Difference: Section 7 details the plant-specific steps required to insert control rods by increasing the cooling water header pressure. The plant-specific steps detailed in Section 7 are beyond the scope of the PSTG but are required to meet the intent of the PSTG.

Discussion: The purpose of Section 7 is to insert control rods by increasing the cooling water header pressure. If the scram failed but control rods are otherwise not stuck, this method may be effective by virtue of the increased pressure exerted on the underside of the drive piston. This is accomplished by fully opening the in-service Flow Control Valve, C11-F002A (F002B) [C12-F002A (F002B)], and Drive Pressure Valve, C11-PCV-F003 (C12-PCV-F003).

60. 295038EK201 001

Unit One (1) and Unit Two (2) are operating at rated power. A release of Floor Drain Sample Tank 'A' is in progress per 0OP-06.4, Discharging Radioactive Liquid Effluents to the Discharge Canal. The following alarms and indications are observed:

Radwaste Effluent Rad Hi-Hi annunciator (UA-03 2-8)	IN ALARM
Radwaste Liquid Effluent Isolation Valve (D12-V27A)	OPEN
Radwaste Liquid Effluent Isolation Valve (D12-V27B)	CLOSED

Which ONE of the following describes the operator actions (if any) that are required to immediately isolate the radwaste effluent release per UA-03 2-8?

A. No operator action is required to immediately isolate the release as Radwaste Liquid Effluent Discharge Valve (D12-V27B) is CLOSED.

B. Radwaste Liquid Effluent Isolation Valve (D12-V27A) must be CLOSED and this can be accomplished with the control switch located on the Unit One (1) RTGB ONLY.

C. Radwaste Liquid Effluent Isolation Valve (D12-V27A) must be CLOSED and this can be accomplished with the control switch located in the Radwaste Control Room ONLY.

D✓ Radwaste Liquid Effluent Isolation Valve (D12-V27A) must be CLOSED and this can be accomplished with control switches located on either the Unit One (1) or Unit Two (2) RTGB.

Feedback

REFERENCE - SD-06.3 rev. 1 page 17, APP UA-03 2-8 AND 0OP-06.4
The radwaste overboard valves (release valves) are controlled by main control board key switches. The two valves are designated 1-D12-V27A and 2-D12-V27B. Both unit's RTGB have switches to control either valve. The valves will trip to the closed position on a High-High radiation monitor alarm

DISTRACTOR ANALYSIS
A. INCORRECT - 0OP-6.4 precaution 3.9 states that the discharge and lineup should be secured if a high radiation trip is received and the AUTOMATIC action failed to occur. Release is still in progress through D12-V27A. These are separate flow paths not in series valves.
B. INCORRECT - Both unit's RTGB have switches to control either valve.
C. INCORRECT - No control in Radwaste
D. CORRECT

Notes

EPE: 295038 High Off-Site Release Rate

EK2. Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following:
(CFR: 41.7 / 45.8)
EK2.01 Radwaste... 3.1 3.4

This question matches the k/a in that it measures the RO's knowledge of the interrelationship of rad release from radwaste systems and how to prevent a release.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 3.6	Facility Objective:	CLS-LP-011*08E
Ref Req'd Y or N:	NO	Technical Ref.:	SD-06.3
? Cognitive Level:	C/A	? Source:	NEW

RADWASTE EFFLUENT RAD HI-HI

AUTO ACTIONS

1. If open, the Radwaste Liquid Effluent Discharge Valve, D12-V27A, closes.
2. If open, the Radwaste Liquid Effluent Discharge Valve, D12-V27B, closes.

CAUSE

1. High rad level in the radwaste discharge to the canal.
2. Circuit malfunction.

OBSERVATIONS

1. The radwaste effluent rad monitor indicates greater than last adjusted setpoint on Panel H12-P604.

ACTIONS

1. If open, close the Radwaste Liquid Effluent Discharge Valve, D12-V27A.
2. If open, close the Radwaste Liquid Effluent Discharge Valve, D12-V27B.
3. Close radwaste discharge flow control valves, G16-FCV-189 and G16-FCV-190.
4. Check the detector for crud buildup and clean if needed.
5. Contact E&RC to determine amount of activity and major isotopes.
6. Consult Section 2.1 of Plant Emergency Procedure.
7. If a circuit malfunction is suspected, ensure that a Trouble Tag is prepared.

DEVICE/SETPOINTS

D12-K607C, Relay K1A	setpoint adjusted per E&RC calculation for release

POSSIBLE PLANT EFFECTS

1. Possible release to the environs.
2. An ODCM Required Compensatory Measure may result.
3. This annunciator is required to be operable to support Radwaste Effluent Monitor operability; annunciator inoperability will result in a Required Compensatory Measure.

REFERENCES

1. LL-9353 - 41
2. PEP-02.1
3. ODCM 7.3.1, Table 7.3.1-1 Function 1

The process radiation monitor drawer (Figure 06.3- 6) has some controls of interest to the radwaste operator. The front panel indicating meter is a base-ten log readout with a range from 10^{-1} to 10^{+6} counts per second (cps). The meter has three pens. Two of the pens set the high alarm (blue) and high-high trip/alarm (red) setting. The indicating pointer is black and should correspond to control room recorder indication. During periods where no release is approved, the red pen (trip) is set to slightly above the black pen. This will ensure that any signal (valid or not) will send a trip signal to the radwaste effluent valves. Once investigated, the trip pen can be readjusted.

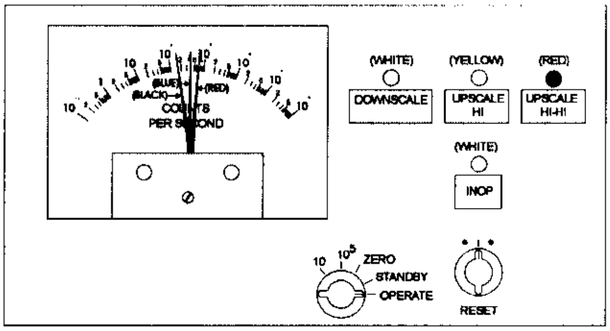


Figure 06.3-6

2.1.7 Radwaste Release Valves

The radwaste overboard valves (release valves) are controlled by main control board key switches. The two valves are designated 1-D12-V27A and 2-D12-V27B. Both unit's RTGB have switches to control either valve. The valves are air operated solenoid valves which will fail closed on a loss of instrument air. **The valves will also trip to the closed position on a High-High radiation monitor alarm.**

2.1.8 Release Monitor Trip Flush Valves

When the Radwaste Monitor receives a HIGH-HIGH radiation signal it sends a trip signal to the Overboard Valves V27A/B. When the monitor trips, flow is secured, but the water which tripped the monitor still remains in the monitor and discharge line. In order not to flush this water out to the discharge canal, manual valves are provided to return this water to the radwaste process system. These valves are the G16-V34, G16-V35, G16-V36, and G16-V37 (see Figure 06.3- 5).

61. 300000A201 001

Unit One (1) is operating at rated power when the following alarm is received.

SERVICE AIR DRYER TROUBLE (UA-01 5-1)

An AO reports that a valve shift failure has occurred on the Service Air Dryer.

Which ONE of the following describes the impact that the valve shift failure will have on the instrument air system and what actions are required to mitigate the abnormal service air dryer operation?

The Service Air Dryer malfunction:

A. could result in low Instrument Air header pressure. The valve shift should be made locally with the valve operator.

B.✓ could result in low Instrument Air header pressure. The Service Air Dryer should be bypassed and removed from service.

C. will not affect the Instrument Air header as it ONLY serves the Service Air header. The valve shift should be made locally with the valve operator.

D. will not affect the Instrument Air header as it ONLY serves the Service Air header. The Service Air Dryer should be bypassed and removed from service.

Feedback

REFERENCE AOP-20 and UA-01 5-1

DISTRACTOR ANALYSIS
A. INCORRECT - No procedural guidance exists to make the valve shift occur.
B. CORRECT - SA Dryer serves the entire air system and this failure has resulted in Low Pressure on IA in an actual event. The AOP directs bypass and Shutdown of SA dryer.
C. and D INCORRECT - SA Dryer serves the entire air system and this failure has resulted in Low Pressure on IA in an actual event.

Notes

A2. Ability to (a) predict the impacts of the following on the .... INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
(CFR: 41.5 / 45.6)
A2.01 Air dryer and filter malfunctions 2.9 2.8

This question matches the k/a in that it measures the RO's ability diagnose the effect of a dryer malfunction on the air system and determine which action is required to mitigate the failure.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.9	Facility Objective:	CLS-LP-046*07F
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-20 UA-01
? Cognitive Level:	M OR FK	? Source:	NEW

SERVICE AIR DRYER TROUBLE

AUTO ACTIONS

NONE

CAUSE

1. Service air dryer dew point high.
2. Service air dryer differential pressure high.
3. Service air dryer temperature high.
4. Service air dryer four-way valve shift failure.
5. Circuit malfunction.

OBSERVATIONS

1. Four-way valve has not fully completed its cycle.
2. Differential pressure gauge at the dryer has an indication of 3 psid or more.
3. High temperature on the regenerating dryer indicated by local thermometer setpoint of 450°F.
4. High dew point indicated by desiccant color change.

ACTIONS

1. Check visual moisture indicator and, if desiccant color change indicates that the problem is due to a high dew point, ensure that a WR/JO is prepared.
2. If a high differential pressure is observed on the pre-filter or after-filter dP gauge at the dryer, ensure that a WR/JO is prepared.
3. If a high temperature is observed at the dryer, check the thermostat to ensure that it is set at 275E - 300°F.
4. If necessary, Bypass Air Dryer, and place Instrument Air Dryer in service.
5. If a circuit or equipment malfunction is suspected, ensure that a WR/JO is prepared.

DEVICE/SETPOINTS

Relay 4CR, 5CR, 6CR, TDR	Energized

POSSIBLE PLANT EFFECTS

1. 9527-LL-9352 - 8
2. SOER 88-01, Recommendation 1

3.0 OPERATOR ACTIONS

NOTE: Service Air System pre-filter or after-filter differential pressure should **NOT** exceed 15 psid.

NOTE: In service air compressor high discharge pressure (U1 [≥125 psig] U2 [≥130 psig]) or relief valves lifting could be an indication of air dryer high differential pressure.

7. **IF** *SERVICE AIR DRYER TROUBLE* (UA-01 5-1)) is in alarm **OR** local observation indicates valve shift failure or filter high differential pressure, **THEN PERFORM** the following:

   a. **OPEN** *AIR DRYER BYPASS VALVE, SA-V864.* ☐

   b. **OPEN** *AIR DRYER BYPASS VALVE, SA-V865.* ☐

   c. **SHUT DOWN** the service air dryer in accordance with 0OP-46. ☐

R1

CAUTION

The instrument air dryer provides a low dew point pneumatic source to downstream components when the Service Air Dryer is out of service or degraded. A low dew point is necessary to insure long term reliability of these components. The time the dryer is bypassed should be minimized when the Service Air Dryer is out of service or degraded.

8. **IF** *INSTRUMENT AIR DRYER TROUBLE* (UA-01 6-4) is in alarm **OR** local observation indicates filter high differential pressure, **THEN OPEN** *INSTRUMENT AIR DRYER BYPASS VALVE, SA-V79.* ☐

9. **ENSURE** proper TBCCW cooling water flow to affected compressors. ☐

10. **CHECK** the breakers for compressors that are tripped **AND RESET** thermal or magnetic trips as required. ☐

11. **IF** interruptible instrument air is lost to the Fuel Pool Gate Seals, **THEN SUPPLY** the seals with nitrogen bottles in accordance with 0OP-46. ☐

62. 400000K104 001

Unit Two (2) is at rated power when the following annunciators ALARM concurrently.

RBCCW HEAD TANK LEVEL HI/LO	UA-03 1-5
RBCCW LIQUID PROCESS RAD HIGH	UA-03 1-6

Which ONE of the following identifies the probable cause of these two alarms?

The probable cause is a tube rupture in:

A.✓ a Reactor Building sample cooler.

B. a Drywell Cooler Heat Exchanger.

C. the RBCCW/Service Water Heat Exchanger.

D. the Drywell Equipment Drain Tank Heat Exchanger.

Feedback

MODIFIED BANK QUESTION LOI-CLS-LP-021-A*005 001
Original Question
Unit Two (2) is at rated power with the following annunciators in ALARM.

RBCCW HEAD TANK LEVEL HI/LO	UA-03 1-5
RBCCW LIQUID PROCESS RAD HIGH	UA-03 1-6

The probable cause is a tube rupture in:

A. a Drywell Cooler Heat Exchanger.

B. the RBCCW/Service Water Heat Exchanger.

C. the Drywell Equipment Drain Tank Heat Exchanger.

D. the Reactor Recirc Pump seal cooling Heat Exchanger. CORRECT

References: 2-APP UA-03 1-5/R38, 2-APP UA-03 1-6/R38 SD-21/R1, AOP-16/R15

DISTRACTOR ANALYSIS
A. CORRECT - Reactor water at high pressure cooled by RBCCW
B.,C., and D. INCORRECT - all at lower pressure than RBCCW

Notes

SYSTEM: 400000 Component Cooling Water System (CCWS)
K1. Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following:
(CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.04 Reactor coolant system, in order to determine source (s)
of RCS leakage into CCWS 2.9 3.1

This question matches the k/a in that it measures the RO's ability diagnose the origin of RCS leakage into the RBCCW system.

Categories

Tier:	TIER 2	Group:	GROUP 1
Importance Rating:	RO 2.9	Facility Objective:	CLS-LP-021*005
Ref Req'd Y or N:	NO	Technical Ref.:	UA-03
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

RBCCW HEAD TANK LEVEL HI/LO

AUTO ACTIONS

NONE

CAUSES

1. High level may be caused by any of the following:
   a. Inleakage from the RBCCW surge tank makeup valve.
   b. Inleakage from components serviced by the RBCCW System.
   c. Heat-up of water in the system.
   d. Improper valve lineup.
2. Low level may be caused by any of the following:
   a. System leakage.
   b. Piping failure.
   c. RBCCW surge tank makeup valve malfunction.
   d. Loss of makeup water pressure.
   e. Improper valve lineup.
3. Circuit malfunction.

OBSERVATIONS

1. Local RBCCW Surge Tank Level Glass, RCC-LG663, indicates actual level, high or low.
2. If gross leakage or piping failure exists, then RBCCW PUMP DISCH HEADER PRESS LOW (UA-03 2-5) should alarm.
3. If high level alarm exists due to inleakage from one or more of the following, then RBCCW LIQUID PROCESS RAD HIGH (UA-03 1-6) may alarm:
   a. RWCU System.
   b. Reactor Recirculation Pump seal coolers.
   c. Fuel Pool Cooling and Cleanup System.
   d. Reactor Building sample coolers.
4. If loss of makeup water exists, then DEMIN WATER XFR PUMP HDR PRESS LOW (UA-14 5-1) should alarm.

ACTIONS

1. If low level exists, determine the cause as follows:
   a. If DEMIN WATER XFR PUMP HDR PRESS LOW alarms, refer to APP UA-14 5-1.
   b. Check that RBCCW Surge Tank Makeup Valve, RCC-LV662, is open.
   c. Throttle open LV662 Bypass Valve, RCC-V314, to return level to 20 inches below top of tank and when that level is achieved, close the valve.
   d. Check RBCCW System for leakage, including drywell chiller area.
   e. Check RBCCW System valve lineup per OP-21, RBCCW System.

ACTIONS (Continued)

2. If high level exists, determine the cause as follows:
   a. Open RBCCW Surge Tank Drain Valve, RCC-V107, to drain to normal level and when level is normal, close the valve.
   b. Check closed LV662 Bypass Valve, 2-RCC-V314.
   c. Close LV662 Inlet Isolation Valve, 2-RCC-V312.
   d. If RBCCW surge tank inleakage is suspected, then refer to OP-21, RBCCW System, for manual level control.

> **NOTE:** If surge tank level increases with makeup water isolated, then inleakage from components serviced by RBCCW may exist. If surge tank level remains constant, then makeup water inleakage may exist.

   e. If inleakage is suspected, refer to AOP-16, RBCCW System Failure.

3. If the level in the RBCCW surge tank cannot be recovered, then refer to AOP-16, RBCCW System Failure.
4. If a circuit malfunction is suspected, then ensure that a Trouble Tag is prepared.

DEVICE/SETPOINTS

Level Switch RCC-LS 664	High: 5 inches from the top of the surge tank. Low: 35 inches from the top of the surge tank.

POSSIBLE PLANT EFFECTS

1. If level in the RBCCW surge tank is not recovered, then subsequent loss of cooling to components serviced by RBCCW may result.
2. Manual Reactor Scram and Recirculation Pump Trip (AOP-16).

REFERENCES

1. LL-9353 - 35
2. OP-21, RBCCW System
3. APP UA-14 5-1, DEMIN WATER XFR PUMP HDR PRESS LOW
4. AOP-16, RBCCW System Failure

RBCCW LIQUID PROCESS RAD HIGH

AUTO ACTIONS

NONE

CAUSES

1. High radiation in RBCCW System due to inleakage from one or more of the following:
   a. RWCU System.
   b. Reactor recirculation pump seal coolers.
   c. Fuel Pool System.
   d. Reactor Building sample coolers.
2. Circuit malfunction.

OBSERVATIONS

1. RBCCW System radiation level increasing as indicated on the RBCCW Rad Monitor (D12-P604) on XU-3.
2. If RBCCW inleakage exists, then RBCCW HEAD TANK LEVEL HI/LO (UA-03 1-5) may alarm.

ACTIONS

1. If inleakage is suspected, then refer to AOP-16, RBCCW System Failure.
2. If a circuit malfunction is suspected, then ensure that a Trouble Tag is prepared.

DEVICE/SETPOINT

D12-K607C, Relay K10	As set by PT-78.4

POSSIBLE PLANT EFFECTS

1. Increase in general area radiation levels.
2. If RBCCW and SW leak occurs, then unmonitored radioactive release.

REFERENCES

1. LL-9353 - 37
2. FP-50054 - 7
3. AOP-16, RBCCW System Failure

63. 500000EA202 001

During an accident on Unit Two (2), following emergency depressurization and reactor water level restoration containment conditions are as follows:

Drywell pressure	33 psig
Drywell average temp	295°F
Suppr Pool water level	+5 inches
Suppr Pool temp	182°F
Suppr Chamber press	32 psig
Drywell hydrogen	4.5% (ERFIS)
Drywell oxygen	3.5% (ERFIS)
Suppr Pool hydrogen	5.5% (ERFIS)
Suppr Pool oxygen	3.5% (ERFIS)

The SRO has directed that the H_2/O_2 monitors CAC-AT-4409 and 4410 be isolated.

Which ONE of the following is the bases for isolating the H_2/O_2 monitors?

The monitor sample pumps are isolated so they will be available for use later in the accident and not damaged by:

A.✓ high drywell pressure conditions.

B. high suppression pool water level.

C. suppression pool hydrogen concentration.

D. drywell average air temperature conditions.

Feedback

REFERENCE 0OI-37.8 REV. 4 PAGE 32

The H2/O2 monitors are designed for pressures up to 30 psig. To preclude damage to these sample pumps and the subsequent radioactive release to secondary containment, the sample pumps are isolated when drywell pressure exceeds 30 psig. The EOPs are written for beyond design basis accidents so it is appropriate to isolate these sample pumps so that they are not damaged and will be able to be used later in the accident when drywell pressure is lower.

DISTRACTOR ANALYSIS
A. CORRECT
B. +5 FEET 4409/4410 torus sample lines must be isolated and shifted to the drywell
C. No requirements to isolate with high hydrogen concentration
D. No requirements to isolate on high drywell temperature.

Notes

EPE: 500000 High Containment Hydrogen Concentration

EA2 Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS:

(CFR: 41.10 / 43.5 / 45.13)
EA2.02 Oxygen monitoring system availability 3.0 3.5

This question matches the k/a in that it measures the RO's ability to interpret the oxygen monitoring equipment availability under conditions where signifcant hydrogen may be produced.

Categories

Tier:	TIER 1	Group:	GROUP 2
Importance Rating:	RO 3.0	Facility Objective:	CLS-LP-024*07F
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-37.8
? Cognitive Level:	M OR FK	? Source:	NEW

STEP PC/P-17

30 PSIG

IF DRYWELL PRESS IS ABOVE 30 PSIG

THEN SECURE AND ISOLATE CAC-AT-4409 AND 4410

PC/P-17

STEP BASES:

The CAC-AT-4409 and CAC-AT-4410 H_2/O_2 analyzers were designed for pressures up to 30 psig. To preclude damage to these sample pumps and the subsequent radioactive release to secondary containment, the sample pumps are isolated when drywell pressure exceeds 30 psig. The EOPs are written for beyond design basis accidents so it is appropriate to isolate these sample pumps so that they are not damaged and will be able to be used later in the accident when drywell pressure is lower.

64. 600000AK101 001

You are a control room operator. A person discovering a fire reports to you that a Class B fire exists in the Unit Two (2) Turbine Building 20' elevation.

Which ONE (1) of the following describes the type of fire that the fire brigade will likely encounter?

A. combustible metals.

B. wood, paper and cloth.

C. energized electrical equipment.

D✓ flammable liquids, greases, and gases.

Feedback

REFERENCE - SD-41 Rev. 4 Page 7

MODIFIED CLS-LP-306-A*04B 001

ORIGINAL QUESTION
You are a control room operator. A person discovering a fire reports to you that a Class C fire exists in the Cable Spread Room. Which ONE (1) of the following describes the fire that the fire brigade will likely encounter?

A. combustible metals.
B. wood, paper and cloth.
C. energized electrical equipment. CORRECT
D. flammable liquids, greases, and gases.

DISTRACTOR ANALYSIS
A.B. and C. INCORRECT - Class B is flammable liquids, greases, and gases.

Notes

APE: 600000 Plant Fire On Site

AK1 Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site:
AK1.01 Fire Classifications by type 2.5 2.8

This question matches the k/a in that it measures the RO's knowledge of fire class.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 2.5	Facility Objective:	CLS-LP-306-A*04A
Ref Req'd Y or N:	NO	Technical Ref.:	SD-41
? Cognitive Level:	M OR FK	? Source:	MOD. LOI BANK

1.3 General System Description

1.3.1 General

1. Fire Theory

There are four classes of fires. Each class requires specific actions and equipment to extinguish it. The classes of fires are:

Class	Description
"A"	Fires involving ordinary combustible materials such as wood, paper and cloth where the "cooling-quenching" effects of water is most effective. Typically the class "A" material leaves an ash residue.
"B"	Fires involving flammable liquids, greases, and gases where the "smothering-blanketing" effect of oxygen excluding agents is most effective. Other extinguishing methods include removal of fuel, temperature reduction, and inhibiting the chemical chain reaction.
"C"	Fires involving energized electrical equipment. De-energizing the power source is the most important element in suppressing the fire. Once the equipment is de-energized, the burning materials are usually Class A or B.
"D"	Fires involving combustible metals such as magnesium, zirconium, sodium, potassium, and titanium. Special extinguishing agents are required for each combustible metal.

2. Fire Suppression Agents

The elements required for a fire to exist are commonly referred to as the fire tetrahedron (Figure 41-1), they are:

- Heat
- Fuel
- Oxidizing agent (typically oxygen)
- Chemical chain reaction

65. 600000AK304 001

A fire in the Control Building requires Control Room evacuation and entry into 0ASSD-02.

0ASSD-02 directs the Diesel Generator Operator to trip the Unit Two (2) RPS MG Set output breakers and open the DC supply breakers to Distribution Panels 4A and 4B.

Which ONE of the following identifies a reason why this procedural action is required?

Failure to perform this action could result in:

A. misoperation of RCIC.

B. a loss of drywell cooling.

C. the inability to operate SRVs.

D✓ uncontrolled injection from HPCI.

Feedback

REFERENCE CLS-LP-304 REV. 1 PAGE 23
Opening specified breakers disables HPCI flow control circuitry to prevent uncontrolled injection, removes power from SRV solenoids (but does not prevent operation from alternate circuitry @ RSDP) and MSIV solenoids to establish high/low pressure interface. Will remove power from normal RCIC circuitry, but alternate power @ RSDP, has no impact on drywell coolers
MODIFIED BANK Q# LOI-CLS-LP-304-A*25A 001
ORIGINAL QUESTION
A fire in the Control Building requires Control Room evacuation and entry into ASSD-02.

ASSD-02 directs the Diesel Generator Operator to trip the Unit Two (2) RPS MG Set output breakers and open the DC supply breakers to Distribution Panels 4A and 4B.

Failure to perform this action could result in:
A. misoperation of RCIC.
B. loss of drywell cooling.
C. inability to operate SRVs.
D. spurious operation of MSIVs. **CORRECT**

DISTRACTOR ANALYSIS
A. INCORRECT - Will remove power from normal RCIC circuitry, but alternate power @ RSDP
B. INCORRECT - has no impact on drywell coolers
C. INCORRECT - removes power from SRV solenoids (but does not prevent operation from alternate circuitry @ RSDP)
D. CORRECT - Opening specified breakers disables HPCI flow control circuitry to prevent uncontrolled injection and MSIV solenoids to establish high/low pressure interface.

Notes

APE: 600000 Plant Fire On Site
AK3 Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:
AK3.04 Actions contained in the abnormal procedure for plant fire on site 2.8 3.4

This question matches the k/a in that it measures the RO's knowledge of reasons for actions taken in ASSD (Fire) procedures.

Categories

Tier:	TIER 1	Group:	GROUP 1
Importance Rating:	RO 2.8	Facility Objective:	CLS-LP-304*25C
Ref Req'd Y or N:	NO	Technical Ref.:	CLS-LP-304
? Cognitive Level:	C/A	? Source:	MOD. LOI BANK

During the cooldown:

- SRVs opening sequence E, G, and B is used to evenly distribute heat in the suppression chamber

- Reactor Vessel Water Level is maintained greater than 20 inches by coordinating RCIC injection and SRV opening.

- The cooldown rate is determined using reactor pressure and a RPV pressure to temperature saturation curve.

- If suppression Pool level increases to -2 feet <u>and</u> suppression pool temperature is less than 140°F, the SCO directs the MCC Operator to transfer RCIC suction from the CST to the Suppression Pool. RCIC suction is transferred to from the CST to the Suppression Pool due to possible Stress imposed on the SRV tail pipes that could occur if a SRV actuation occurred with Suppression pool level greater than -2 feet.

The next action for the Unit SCO and MCC operator is to place RHR in suppression pool cooling. This action requires co-ordination with, and actions by the Diesel Generator Operator, the Emergency Switchgear Operator and the Service Water Building Operator.

Prior to reporting to the Diesel Building, the Emergency Switchgear and Diesel Generator Operators must complete important actions in the Battery Rooms.

The actions to prevent spurious operation of MSIVs, SRVs and HPCI are actions that must be performed soon after the decision to abandon the Control Room is made. (Spurious Operation and High/Low Pressure Interface) The MSIVs may spuriously open during a Control Room fire, resulting in a loss of reactor vessel water inventory. The SRVs may spuriously open during a Control Room fire resulting in a loss of reactor vessel inventory. HPCI may auto start or spuriously start and then fail to trip on high reactor vessel level. This could result in flooding RCIC, HPCI, and SRV steam lines and cause severe damage to RCIC, HPCI, and the SRVs.

The HPCI systems are disabled and the MSIVs and the SRVs are prevented from spurious operation by the following actions performed by the Emergency Switchgear Operator for Unit 1, and the Diesel Generator Operator for Unit 2:

- Power is removed from the Unit 1 MSIVs, SRVs, and HPCI by opening the output breakers for Unit 1 RPS MG Sets A&B and opening the supply breakers to 125V DC distribution panels 3A and 3B located on 125/250 switchboards 1A and 1B.

66. GRO2.1.23 001

Unit One (1) is at rated power with all equipment OPERABLE and in a Standby Lineup.

Which ONE of the following meets the criteria of a simple evolution per 0AP-013, Plant Equipment Control?

A. Shifting CRD Stabilizing Valve Sets.

B✓ Alternating Condensate Transfer Pumps.

C. Adjusting Reactor Feed Pump Bias Settings.

D. Transfer of Feedwater Control Mode Select Switch.

Feedback

REFERENCE 0AP-013 REV. 10 PAGE 7

Shifting equipment (i.e., swapping pumps/fans) if the equipment is in its standby line-up, and no special information is required to start equipment other than turning the switch. Standard practices such as pre-start checks will be done prior to starting the equipment

DISTRACTOR ANALYSIS
A. INCORRECT - Requires valve lineup prior to transferring stabilizing valve select switch. Special documentation is required (IV) for a Continuous Use procedure.
B. CORRECT - Meets requirement in AP-013 listed above. No special info is required to shift pumps.
C. INCORRECT - Procedure is Continuous Use and is more involved than a simply making an adjustment to a monitored parameter. Procedure contains a CAUTION and implications of improper performance could result in a plant transient.
D. INCORRECT - Procedure is reference use but is too involved to be considered simple. It modifies the feedwater control mode and must only be performed if plant conditions have been steady for at least 5 minutes when transferring to 3-ELEM.

Notes

2.1.23 Ability to perform specific system and integrated plant procedures during different modes of plant operation.
(CFR: 45.2 / 45.6)
IMPORTANCE RO 3.9 SRO 4.0

This question matches the k/a in that it measures the RO's knowledge of standard operating practices for simple evolutions.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.9	Facility Objective:	CLS-LP-201-C*16D
Ref Req'd Y or N:	NO	Technical Ref.:	AP-013
? Cognitive Level:	M OR FK	? Source:	NEW

5.2 Simple Evolutions

5.2.4 Evaluation of a proposed operation and its designation as a simple evolution will be the responsibility of the Shift Superintendent/Unit SCO. While this decision must be made on a case by case basis, examples of simple evolutions include the following:

1. Changing recorder chart paper

2. Changing the point monitored, range, or chart speed on a monitoring instrument

3. Blowing down an air receiver by throttling open and then closing the drain

4. Filling or draining cooling system head tanks

5. Shifting equipment (i.e., swapping pumps/fans) if the equipment is in its standby line-up, and no special information is required to start equipment other than turning the switch. Standard practices such as pre-start checks will be done prior to starting the equipment

6. The stroking of individual valves for PMTs or ISI data acquisition may be considered a simple evolution if the Shift Superintendent/Unit SCO determines that there will be no adverse impact on the plant. If the manipulation requires Independent Verification of restoration to service, then the requirements of OPS-NGGC-1303 shall be met. A log entry shall be made in the CO log detailing which valve was operated, the reason, results, and the as left position of the valve

7. The person that operates a valve or breaker is responsible for ensuring appropriate log entries are made in the CO's log.

8.4 Shifting CRD Stabilizing Valve Sets

C
Continuous
Use

8.4.1 Initial Conditions

Date/Time Started _____

Initials

1. CRD Hydraulic System is in operation in accordance with Section 5.1. _____

8.4.2 Procedural Steps

1. **OPEN** *STABILIZING VALVE SET 2A(2B) INLET ISOLATION VALVE, C12-F067A(F067B),* for the out-of-service stabilizing valve set. ___/___ Ind.Ver.

2. **OPEN** *STABILIZING VALVE SET 2A(2B) OUTLET ISOLATION VALVE, C12-F068A(F068B),* for the out-of-service stabilizing valve set. ___/___ Ind.Ver.

3. **PLACE** *STABILIZING VLV SELECT* switch on Panel P603 in *A(B)* for the stabilizing valve set to be placed in service. _____

4. **CLOSE** *STABILIZING VALVE SET 2A(2B) INLET ISOLATION VALVE, C12-F067A(F067B),* for the original in-service stabilizing valve set. ___/___ Ind.Ver.

5. **CLOSE** *STABILIZING VALVE SET 2A(2B) OUTLET ISOLATION VALVE, C12-F068A(F068B),* for the original in-service stabilizing valve set. ___/___ Ind.Ver.

6. **MONITOR** operation of the CRD Hydraulic System in accordance with Section 6.0. _____

Date/Time Completed _______________

Performed By (Print)	Initials
______________________	_____
______________________	_____
______________________	_____
______________________	_____

Reviewed By: ______________________
Unit SCO

R
Reference
Use

8.9 Alternating Pumps

8.9.1 Initial Conditions

1. Condensate Storage and Transfer System in service in accordance with Section 5.2, **AND** alternating condensate transfer pumps is required. ☐

OR

2. Demineralized Water Storage and Transfer System in service in accordance with Section 5.1, **AND** alternating demineralized water transfer pumps is required. ☐

8.9.2 Procedural Steps

1. **PERFORM** the following for the condensate transfer pumps **OR** the demineralized water transfer pumps:

   a. **PLACE** the pump in position *AUTO 1* in *ON.* ☐

   b. **PLACE** the pump that has been running in *OFF.* ☐

   c. **ENSURE** discharge pressure is stable. ☐

   d. **PLACE** the pump in *AUTO 2* in *AUTO 1.* ☐

   e. **PLACE** the pump in *OFF* in *AUTO 2.* ☐

8.23 Adjusting Reactor Feed Pump BIAS Settings

C
Continuous Use

8.23.1 Initial Conditions

1. It is desired to adjust reactor feed pump BIAS settings. ☐

8.23.2 Procedural Steps

1. **DEPRESS** *SEL* pushbutton on *RFPT A SP CTL, C32-SIC-R601A*, until *A BIAS* is displayed. ☐

2. **DEPRESS** *SEL* pushbutton on *RFPT B SP CTL, C32-SIC-R601B*, until *B BIAS* is displayed. ☐

CAUTION

Momentarily depressing the raise or lower pushbuttons on *C32-SIC-R601A(B),* will cause pump demand to change in increments of 0.1%. Continuously depressing the raise or lower pushbuttons will cause pump demand to change at an exponential rate.

3. **SLOWLY ADJUST** *BIAS* settings as necessary until suction flows are approximately equal. ☐

4. **IF** desired, **THEN DEPRESS** *SEL* pushbutton on *RFPT A SP CTL, C32-SIC-R601A*, until *PMP A DEM* is displayed. ☐

5. **IF** desired, **THEN DEPRESS** *SEL* pushbutton on *RFPT B SP CTL, C32-SIC-R601B*, until *PMP B DEM* is displayed. ☐

8.2 Transfer of Feedwater Control Mode Select Switch

R
Reference
Use

8.2.1 Initial Conditions

1. Total feedwater flow is greater than 2.0 X 10^6 lbm/hr. ☐

2. Feedwater control is in master automatic. ☐

8.2.2 Procedural Steps

1. **IF** switch is to be transferred to *3 ELEM*, **THEN PERFORM** the following:

   a. **ENSURE** steam flow and feedwater flow have negligible mismatch using multiple indications (i.e., Process Computer; Steam Flow/Feed Flow Recorder, C32-R607; ERFIS). ☐

   b. **ENSURE** vessel level has been at steady state for approximately 5 minutes. ☐

2. **DEPRESS** *SEL* button on *SULCV, FW-LIC-3269*, until *LVL ERROR* is displayed **AND CHECK** level error is less than approximately 2 inches. ☐

3. **SHIFT** *FEEDWATER CONTROL MODE SELECT* control switch to the desired position. ☐

67. GRO2.1.3 001

Unit One (1) and Unit Two (2) are operating at rated power. The Shift Superintendent is leading the crew shift turnover briefing.

Which ONE of the following identifies the minimum number of personnel designated to monitor the RTGB during the turnover briefing?

A. One Control Operator monitoring both units.

B.✓ One Control Operator monitoring each unit.

C. One Control Operator monitoring each unit and one Senior Control Operator monitoring both units.

D. One Control Operator and one Senior Control Operator monitoring each unit.

Feedback

REFERENCE 0OI-01.02 REV. 34 PAGE 12

One Control Operator for each unit shall be designated to monitor the RTGB during the brief.

A. INCORRECT - Each unit requires a dedicated RTGB monitor
B. CORRECT
C. INCORRECT - ONLY CO required to be designated
D. INCORRECT - ONLY CO required to be designated

Notes

2.1.3 Knowledge of shift turnover practices.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.0 SRO 3.4

This question matches the k/a in that it measures the RO's knowledge of turnover practices.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.0	Facility Objective:	CLS-L-201-D*05F
Ref Req'd Y or N:	NO	Technical Ref.:	0OI-01.02
? Cognitive Level:	M OR FK	? Source:	NEW

5.6 Crew Shift Briefing

NOTE:	The crew shift turnover briefing may be held separately, with the briefing for the non-operating unit held outside the Control Room and at a different time, for cases such as Refueling or Forced Outages to reduce Control Room distractions and accommodate dual unit personnel attendance to both.

5.6.1 The crew shift turnover briefing should begin promptly at 0710 and 1910.

5.6.2 The crew shift briefing will be led by the Shift Superintendent or designee in the Main Control Room.

1. One Control Operator for each unit shall be designated to monitor the RTGB during the brief.
2. The Shift Superintendent is responsible for minimizing distractions in the Control Room during crew shift briefings, these include non-critical phone calls, changing/initialing chart paper, hanging clearances, and scheduled RTGB activities.
3. The crew shift briefing shall contain the following (as a minimum):

    a. Current plant mode and power level

    b. Status of Technical Specification equipment

    c. Abnormal plant conditions/equipment status

    d. Major work in progress

    e. Limitations and other information needed for plant operations

    f. Work priorities

    g. Requests for comments

    h. Review of applicable Nuclear Condition Reports (NCRs) generated since the crew last stood watch

    i. A review and discussion of any Standing Instruction issued since the crew last stood watch

    j. Planned activities related to Reactivity Management

68. GRO2.2.30 001

A core reload sequence is in progress in accordance with 0FH-11, Refueling. The initial loading of fuel bundles around each SRM centered 4-bundle cell was completed with all four SRMs fully inserted and reading 50 cps.

After several fuel assemblies were loaded the RO observes that all have increased steadily. Current readings are as follows:

SRM A	120 cps	SRM C	260 cps
SRM B	130 cps	SRM D	140 cps

Which ONE of the following identifies the ROs required responsibilities at this time?

The RO should:

A.✓ immediately direct suspension of fuel movement.

B. continue monitoring SRMs during fuel movement as readings are not unusual.

C. report to the SRO that SRM C is INOPERABLE and recommend bypassing SRM C until I&C can investigate. Fuel movement may continue.

D. consult with the Reactor Engineer to determine if the SRM readings are consistent with expected values for this particular reload sequence. Fuel movement may continue.

Feedback

REFERENCE FH-11 Rev. 77 pages 7 and 8
FH-11 requires suspending fuel movement if SRM increase by factor of 5 relative to initial base-line SRM reading (or doubles with any single bundle). Also, malfunctioning of any SRM channel shall be reason to terminate refueling operations until TS compliance can be determined.

DISTRACTOR ANALYSIS
A. CORRECT - Only requires one SRM to increase to 5X above baseline require suspension of fuel movements.
B. INCORRECT - Fuel movement is required to be suspended.
C. INCORRECT - SRM C should not be declared INOP based upon this deviation from the other channels until I&C can investigate. FH-11 requires termination of fuel movement for malfunction of SRM in any account.
D. INCORRECT - Fuel movement is required to be suspended

It is often confused that the procedural requirement of a factor five with five doubles. If applicant does not recognize that SRM C increased by a factor of 5 then all other choices may seem reasonable.

Notes

2.2.30 Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area / communication with fuel storage facility / systems operated from the control room in support of fueling operations / and supporting instrumentation.
(CFR: 45.12)
IMPORTANCE RO 3.5 SRO 3.3

This question matches the k/a in that it measures the RO's knowledge duties regarding SRM and criticality monitoring during fuel handling.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.5	Facility Objective:	CLS-LP-305-C*025
Ref Req'd Y or N:	NO	Technical Ref.:	FH-11
? Cognitive Level:	C/A	? Source:	NEW

4.0 PRECAUTIONS AND LIMITATIONS

R7 4.35 If attaching tools, such as a jet pump grapple or control blade latching tool, to either the monorail or frame mounted hoist, verify proper thread engagement/size by ensuring there is no play in the connection prior to thread engagement of three (3) full turns. The correct tool and coupling thread size is 7/16-14 UNC. Additionally, a 1/2-13 UNC bolt will not fit into a proper size tool (7/16-14 UNC); thus, this check may be performed if practical. Failure to detect mis-matched thread sizes will significantly reduce the load capacity of the tool/hoist.

4.36 Valve F15-V5000 may be closed as necessary, to isolate the pneumatic air system unloader valve upon failure of the unloader valve.

4.37 Indication of criticality observed on the SRM indicators during functional, subcritical, or shutdown margin rod checks shall be reason to terminate fuel loading until a complete evaluation of the cause of the criticality indication is determined.

4.38 Fuel movement should be suspended and the Reactor Engineer contacted if either of the following occur:

4.38.1 An SRM reading increases by a factor of two upon insertion of any single bundle after the initial loading of fuel bundles around each SRM is complete.

4.38.2 An SRM increases by an overall factor of five relative to the baseline SRM count rate obtained after the initial loading of fuel bundles around each SRM centered 4-bundle cell is complete.

4.39 Indications of criticality observed on the SRM indicators during any fuel movements shall be cause to terminate fuel movements until a complete evaluation of the cause of the criticality indication is determined.

4.40 The loss of communications between the Control Room and the refueling floor shall be reason to terminate refueling operations.

4.41 The malfunctioning of any nuclear instrumentation channel shall be reason to terminate refueling operations until Tech Spec compliance has been determined.

69. GRO2.2.4 001

The Fire Brigade requests that the Motor-Driven fire pump be started.

Which ONE of the following identifies where this action can be performed?

The Motor-Driven fire pump can be started locally:

A. ONLY.

B✓ and at the Unit One (1) RTGB ONLY.

C. and at the Unit Two (2) RTGB ONLY.

D. and at both the Unit One (1) and Unit (2) RTGBs.

Feedback

REFERENCE - 0OP-41 REV. 75 PAGE 21.

DISTRACTOR ANALYSIS
A. - INCORRECT - Control pushbutton on Unit One RTGB can be used per 0OP-41
B. - CORRECT
C. and D. - INCORRECT - Plausible that controls would be exist on Unit Two however they do not. This is a difference between Units

Notes

2.2.4 (multi-unit) Ability to explain the variations in control board layouts / systems / instrumentation and procedural actions between units at a facility.
(CFR: 45.1-45.13)
IMPORTANCE RO 2.8 SRO 3.0

This question matches the k/a in that it measures the RO's knowledge of unit differences regarding fire pump controls.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 2.8	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	OP-41
? Cognitive Level:	M OR FK	? Source:	NEW

8.3 Manual Starting of Motor-Driven Fire Pump

R
Reference Use

8.3.1 Initial Conditions

1. All applicable prerequisites as listed in Section 4.0 are met. ☐

2. The Fire Protection System is in operation in accordance with Section 5.1. ☐

8.3.2 Procedural Steps

CAUTION

The Motor-Driven Fire Pump must be shut down manually after an automatic initiation.

1. **IF** Motor-Driven Fire Pump is to be started from the control room, **THEN MOMENTARILY DEPRESS** the *START* push button. ☐

2. **IF** Motor-Driven Fire Pump is to be started locally, **THEN MOMENTARILY DEPRESS** the local *START* push button. ☐

Unit One Side only

The motor driven fire pump may also be started from the Fire Protection panel XU-69 in the Control Room. The remote start push button starts the pump without supervision of the pressure switch and the pump will run until manually stopped at the local panel.

Alarms are provided on Annunciator Panel UA-37:

- MWT BLDG MTR DRV FIRE PUMP RUNNING - initiated by (PS-1871 at 105 psig ± 10 psig and lowering)
- MOTOR-DRIVEN FIRE PUMP FAIL TO RUN

Electrical power for the motor driven fire pump P-2 is supplied from bus E-2 (CB-AH7) and from bus E-4 (CB-AL3). The two feeders terminate at Transfer Switch (LG-5) located adjacent to the Water Treatment Building. Upon loss of power from the normal source (E-2), a manual transfer switch is available to transfer to the alternate source (E-4).

3.2.3 Diesel Driven Fire Pump

The Diesel Driven Fire Pump, P-1, local control panel is equipped with a five-position selector switch (OFF, MANUAL START A, MANUAL START B, TEST, AUTO) and two manual push buttons (MANUAL START and RESET).

With the selector switch in AUTO, the pump will start when system pressure drops to approximately 90 psig. Once started, the engine will continue to run until manually stopped or automatically stopped by engine overspeed. There are no other automatic shutdowns.

The pump may also be started by placing the control switch to MANUAL A or MANUAL B and depressing the MANUAL START pushbutton. The selection of MANUAL A or MANUAL B selects the battery that will crank the diesel until it fires. The engine will not crank when the MANUAL START pushbutton is depressed if either the MANUAL A or the MANUAL B mode switch position is not selected.

The pump will receive an auto start signal on station blackout or loss of power to the local control panel which controls the battery charging circuit and automatic controls.

70. GRO2.3.10 001

Unit Two (2) operators are performing a downpower to support repair of a condenser water box leak. The maintenance planning ticket has an item for operations to reduce Hydrogen Water Chemistry System hydrogen injection rate.

Which ONE of the following identifies the reason why operations would need to reduce hydrogen injection rate during the downpower?

Reduction of hydrogen injection rate will:

A.✓ reduce dose rates in the condenser water box area during downpower.

B. eliminate the possibility of an explosive atmosphere existing in the area of the leak.

C. establish the proper concentrations of hydrogen and oxygen to support vessel chemistry corresponding to the reduced reactor power level.

D. ensure a suitable breathing environment exists in confined spaces, such as the condenser water box pit area, prior to entry.

Feedback

REFERENCE SD-59 PAGE 8 REV. 9 - Injection of hydrogen shifts the N-16 distribution ratio increasing gamma dose carried over in steam. Dose levels near condenser water box area are highly impacted by hydrogen injection.

DISTRACTOR ANALYSIS

A. CORRECT - Good ALARA practice when balanced with Chemistry needs.
B. INCORRECT - Not likely to have an explosive atmosphere in this location as hydrogen is injected downstream at low concentrations.
C. INCORRECT - Correct ratio is automatic and will adjust during a downpower without operator action. Actually reducing hydrogen injection will impact vessel chemistry in a negative way.
D. INCORRECT - Not likely to affect breathing conditions by reducing hydrogen injection rate. However, distractor is plausible as it is a gas not suitable for breathing when in large concentrations that could collect in a confined space.

Notes

2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

(CFR: 43.4 / 45.10)

IMPORTANCE RO 2.9 SRO 3.3

This question matches the k/a in that it measures the RO's knowledge of procedure performed to reduce radiation levels during plant maintenance.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 2.9	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	SD-59
? Cognitive Level:	M OR FK	? Source:	NEW

This entire process is monitored very closely by E&RC Chemistry, which is responsible for tracking Hydrogen and Oxygen injection. Since the discovery of shroud cracking caused by IGSCC, Chemistry tracks time In-Service for Hydrogen/Oxygen injection and should be notified anytime HWC is removed or placed into service.

1.3.2 Radiological Implications Of Hydrogen Water Chemistry Control

The primary source of background radiation levels during reactor operation, near steam lines outside the Primary Containment, is attributed to the decay of Nitrogen-16 (N^{16}). N^{16} has a half-life of 7.1 seconds and decays with the emission of a high-energy gamma (6.1 Mev).

The major sources of Nitrogen in a BWR are from Oxygen-16 and from the leakage of nitrogen based chemical compounds from the RWCU and Condensate demineralizers. Oxygen-16 forms Nitrogen-16 via a neutron-proton reaction.

($O^{16} + \eta \text{-->} N^{16} + \rho$)

When using normal water chemistry methods (i.e., without H_2 injection), a major portion of the Nitrogen-16 present in the reactor coolant combines with the free Oxygen to form water-soluble Nitrites (NO_2) and Nitrates (NO_3). These compounds are circulated through the reactor coolant systems and are ultimately removed by the RWCU System. A smaller fraction of the Nitrogen-16 is carried over in the steam in the form of Nitrogen gas (N_2) and Ammonia (NH_3) and is the predominate contributor to background radiation levels.

The implementation of Hydrogen Water Chemistry (H_2 injection) alters the Nitrogen-16 carryover ratio. The net production of Nitrogen-16 is not influenced by Hydrogen injection.

The excess Hydrogen injected into the reactor coolant creates the driving force to shift the Nitrogen-16 distribution ratio, resulting in a larger fraction of the Nitrogen-16 forming volatile Ammonia and a smaller fraction forming Nitrites and Nitrates. This additional volatile Ammonia is then carried over in the reactor steam resulting in higher background radiation levels. Any increase in Hydrogen injection rates will result in a proportional increase in background radiation levels and vise-versa.

71. GRO2.3.2 001

A clearance requires that a Reactor Water Cleanup (RWCU) system drain valve in the RWCU heat exchanger room be independently verified to be in the closed position. It has been estimated that in order to complete the independent verification (IV) an operator would have to spend approximately 10 minutes in the general area of the valve where the dose rate is 90 mRem/hr.

Which ONE of the identifies the appropriate IV requirements per OPS-NGGC-1303, Independent Verification, procedure while performing the work described above?

Per OPS-NGGC-1303, the expected radiation exposure does:

A. NOT meet the excessive exposure criteria to waive the IV and therefore must be performed.

B✓ meet the excessive exposure criteria to waive the IV. The IV can be waived with authorization by the Control Room Supervisor (CRS).

C. meet the excessive exposure criteria to waive the IV. However, the IV cannot be waived since the excessive exposure criteria does not apply valves in a locked room that is not routinely monitored.

D. meet the excessive exposure criteria criteria to waive the IV. However, the IV cannot be waived since the excessive exposure criteria does not apply to valves that are part of a Technical Specification required system.

Feedback

REFERENCE - OPS-NGGC-1303 REV. 3 PAGE 16 - IV requirements can be waived if exposure is greater than 10 mrem. Waiver requires authorization from the respective supervisor (page 6).

DISTRACTOR ANALYSIS
A. INCORRECT - Calculates to 15mR which does meet requirements to consider waiver.
B. CORRECT - Exposure does meet waiver requirements and CRS can authorize waiver of IV as procedure does not make any other additional stipuations.
C. INCORRECT - Exposure does meet waiver requirements and CRS can authorize waiver of IV as procedure does not stipuate that the IV can NOT be waived for valves in locked rooms.
D. INCORRECT - Exposure does meet waiver requirements and CRS can authorize waiver of IV as procedure does not stipulate that the IV can only be waived for valves that are not Tech Spec.

Notes

2.3.2 Knowledge of facility ALARA program.
(CFR: 41.12 / 43.4 / 45.9 / 45.10)
IMPORTANCE RO 2.5 SRO 2.9

This question matches the k/a in that it measures the RO's knowledge of plant ALARA practices.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 2.5	Facility Objective:	CLS-LP-201-C*10B
Ref Req'd Y or N:	NO	Technical Ref.:	OPS-NGGC-1303
? Cognitive Level:	C/A	? Source:	NEW

9.5 Exceptions to Independent Verification

9.5.1 Systems and components operated from the Main Control Board or Waste Processing Control Board and other control panels in the Main Control Room or Radwaste Control Room on a daily basis to support normal plant operation do not require INDEPENDENT VERIFICATION. Attachment 4 lists the RTGB components at RNP that are exempt from INDEPENDENT VERIFICATION only for operating evolutions performed from the RTGB. Maintenance evolutions on the components listed in Attachment 4 will require Independent Verification as applicable.

9.5.2 Reterminations in Q-list and Technical Specification-related systems require INDEPENDENT VERIFICATION using drawings or a Work Order or procedure where there is identification of the reterminations. [At BNP - this does not apply to momentary wire lifts as defined in 0AI-59, Jumpering and Wire Removal]. QC requirements for reterminations are contained in NUA-NGGC-1530.

9.5.3 INDEPENDENT VERIFICATION requirements may be waived if:

1. Excessive radiation exposures would result. As a guideline, an exposure of greater than 10 mrem to conduct the INDEPENDENT/CONCURRENT VERIFICATION would be considered excessive. Individual situations should be determined on a case-by-case basis by the respective supervisor. In these situations, an alternate means such as FUNCTIONAL VERIFICATION not involving radiation exposure (such as observing process parameters) should be utilized.

2. Entry into any area where personnel safety is compromised or jeopardized due to the presence of extreme temperatures (greater than 120°F), or other hazards potentially dangerous to health are present.

3. Manipulated equipment have required positions controlled by valve and equipment lineup sheets and current plant operational conditions do not require the system to be operable. In these situations, prior to the time operability is required, valve and equipment lineup check sheets with INDEPENDENT VERIFICATION shall be completed.

72. GRO2.3.4 001

A Progress Energy employee, upon employment, has received non-Progress Energy occupational dose for the current year, but the amount has not been determined.

He has received 450 mRem (TEDE) Progress Energy dose for the year.

Which ONE of the following identifies the maximum additional dose this worker may receive for the year without exceeding any administrative limit as specified in NGGM-PM-0002, Radiation Control and Protection Manual?

A✓ 50 mRem.

B. 550 mRem.

C. 1550 mRem.

D. 3550 mRem.

Feedback

REFERENCE - NGGM-PM-0002 REV. 33 PAGE 11

Normal admin limit is 2 rem Progress Energy and 4 rem total, however if non-Progress Energy occupational dose for the year has not been determined, admin limit of 0.5 rem is imposed

Randomly selected from the bank LOI-CLS-LP-102-A*005 001

DISTRACTOR ANALYSIS
A. CORRECT - adds up to 0.5 rem limit imposed
B. INCORRECT - adds up to 1 rem, not a limit
C. INCORRECT - adds up to 2 rem Progress Energy admin limit
D. INCORRECT - adds up to 4 rem, Progress Energy total dose if non-Progress Energy dose is determined

Notes

2.3.4 Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized.
(CFR: 43.4 / 45.10)
IMPORTANCE RO 2.5 SRO 3.1

This question matches the k/a in that it measures the RO's knowledge of radiation exposure limits.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 2.5	Facility Objective:	CLS-LP-102-A*005
Ref Req'd Y or N:	NO	Technical Ref.:	NGGM-PM-0002
? Cognitive Level:	C/A	? Source:	BANK LOI

4.5 Radiation Control Training

4.5.1 Radiation Control training shall be given to ensure that each person who enters the radiation control area (RCA), or who may be involved with radiological activities, understands their responsibility to minimize their dose and understands the associated risks.

4.5.2 Special briefings and training, including the use of mockups where applicable, should be considered for work involving higher than usual exposures to radiation, or difficult to master skills and techniques.

4.5.3 Radiation Control Technicians and their supervisors should review theoretical and practical training. Training or briefing(s) shall also be given to applicable procedures, equipment, and programs. This training shall be performed in accordance with the appropriate training program procedure.

4.5.4 Respiratory Protection training is required for persons who may perform work under the respiratory protection program.

4.6 Exposure Control

4.6.1 Annual Administrative Dose Limits

The Progress Energy goal is that no individual shall exceed the following annual administrative limits for total effective dose equivalent (TEDE):

1. 0.5 rem Progress Energy dose if non-Progress Energy occupational dose for the current year has not been determined (no dose extension permitted).
2. 2 rem Progress Energy dose and 4 rem total dose if non-Progress Energy occupational dose for the current year has been determined.
3. Site Vice President approval is required to exceed the annual administrative dose limits.

4.6.2 Lifetime Administrative Dose Limit

1. Progress Energy personnel cumulative lifetime TEDE in rem shall not exceed the individual's age in years as of the end of the year.
2. Progress Energy personnel annual administrative limit shall be reduced as necessary to avoid exceeding the lifetime dose limit unless an annual dose limit extension is authorized by the Site Vice President or designee.

73. GRO2.4.13 001

Which ONE of the following describes the Control Operator (CO) reponsibilities regarding immediate actions during EOP flowchart use?

The CO is expected to perform immediate actions:

A. from memory. It is mandatory that the immediate actions be performed prior to entering the EOP-01 Scram flowchart.

B.✓ from memory. It is NOT mandatory that immediate actions be performed prior to entering the EOP-01 Scram flowchart.

C. by marking off steps on the Scram Card instructional aid. It is mandatory that these steps be performed prior to entering the EOP-01 Scram flowchart.

D. by marking off steps on the Scram Card instructional aid. It is NOT mandatory that these steps be performed prior to entering the EOP-01 Scram flowchart.

Feedback

REFERENCE - EOP-UG REV. 43 PAGE 21-23

DISTRACTOR ANALYSIS
A. INCORRECT - Although scram IA are expected to be memorized it is not mandatory that the actions be completed prior to entering the scram flow chart as these steps are contained in the flow chart.
B. CORRECT
C&D - INCORRECT - Scram Card contains steps other than IA in the scram procedure that can be performed without direct instruction from the SCO and which would not create a problem with sequence of steps.

Notes

2.4.13 Knowledge of crew roles and responsibilities during EOP flowchart use.
(CFR: 41.10 / 45.12)
IMPORTANCE RO 3.3 SRO 3.9

This question matches the k/a in that it measures the RO's knowledge of EOP immediate actions requirements.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.3	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	EOP-UG
? Cognitive Level:	M OR FK	? Source:	NEW

C. OPERATOR ACTIONS

1. Control Operator Immediate Actions

The control operator immediate actions are those actions which may be performed following a reactor scram prior to entering the scram procedure (EOP-01). These actions are not mandatory and shall not conflict with entering the scram procedure. All the control operator immediate actions are located in the scram procedure flowchart. There are no control operator immediate actions in EOP-02 through EOP-04. In the event the actions are not performed prior to entering the scram procedure, the scram procedure shall take precedence. The control operator immediate actions which should be memorized by control operators, are defined as follows:

a. Unit 2 Only: After steam flow is less than 3 x 10^6 lb/hr, PLACE the reactor mode switch to SHUTDOWN.
Unit 1 Only: PLACE the reactor mode switch to SHUTDOWN.

b. IF reactor power is below 2% (APRM downscale trip), THEN TRIP the main turbine.

c. ENSURE the master reactor level controller setpoint is +170".

d. IF two reactor feed pumps are running, AND reactor vessel level is above +170" AND rising, THEN TRIP one.

The EOP actions are those which are contained within EOP-01 through EOP-04. In the event the control operator immediate actions are not performed prior to entering EOP-01, these actions become EOP actions.

Since the EOP actions are readily available to the control operator, there is no need to memorize them.

The operator is not required to have the Operating Procedures in hand while executing the EOPs, but may use any other procedure as necessary.

The following guidance applies to referencing of supporting material that is not included in the procedure but provides information required in the performance of a step (ERFIS, instructional aids, Users' Guide, etc.).

a. If the information is available from several sources and a specific source is preferred, then that source is explicitly referenced at the point it is needed.

b. If the information is provided by a source which is not readily recognized by the operator, then the source is explicitly referenced at the point it is needed.

4. Flowchart Execution

When executing these procedures the SRO normally handles the flowchart; i.e., reads the steps and marks them off as they are executed while one or more ROs perform the control board verifications/manipulations. The person using the flowchart should read those steps aloud that are necessary to give adequate guidance to the persons operating the RTGB to execute the procedure.

The steps of the flowchart should be marked off as they are executed or circled if execution is not accomplished. A circled step should be readdressed at a later time as plant conditions permit. An exception to the circling of steps allowance exists for WHEN steps since the definition of a WHEN step (see III.B.1) does not allow performance of subsequent operator actions until the identified value or condition exists. Overhead projector transparency water base pens are preferable to grease pencils because they are easier to write with and easier to read.

It is very important for the Control Room personnel to maintain a constant vigil of the entire plant's condition. The EOPs enhance the operator's ability to perform the correct actions consistently and in a timely manner, during many varied situations. However, the operators must not "lock in" on the procedures and become unaware of important parameters and conditions that may change during rapidly developing transients and plant evolutions.

5. Periodically, the SRO should review critical steps that have been activated and read out loud those critical steps that the RO needs to monitor.

6. Control Operator Actions for Scram Card Instructional Aid

The following actions have been taken from the Reactor Scram Procedure, 1(2)EOP-01-RSP. These actions are items that the Control Operator should be able to perform without direct instructions from the Senior Control Operator and which would not create problems with the sequence of steps in the reactor scram procedure. If these steps are not completed using the instructional aid, then the reactor scram procedure will direct the actions.

- Ensure scram valves are open by manual scram or ARI trip
- Control reactor pressure between 800 and 1000 psig
- Control reactor vessel level between +170 and +200 inches
- Insert nuclear instrumentation
- Place recirc pump speed controller to 10%
- Ensure heater drain pumps are tripped
- Ensure turbine oil system operating

74. GRO2.4.15 001

Following the announcement of a fire in the South RHR room, the Shift Superintendent has announced "In the Control Room Shift Brief".

Which ONE of the following should NOT be communicated during the brief?

A. Status of fire.

B. EAL classifications.

C. Status of evacuations.

D✓ Orders to the fire brigade.

Feedback

Selected from NRC BSEP 2001 exam Q# GEN 2.4.15

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2. Actions Main Control Room
a. The Shift Superintendent should provide a brief to Control Room personnel when plant status has changed significantly, periodically to ensure the entire crew is aware of plant status and direction or during a lull in activities...
d. Orders should NOT be given during a brief.

DISTRACTOR ANALYSIS
A. INCORRECT - Status should be communicated
B. INCORRECT - EAL classifications should be communicated
C. INCORRECT - Evacuation status should be communicated
D. CORRECT - Orders are specifically noted as items that should not be communicated during a brief.

Notes

2.4.15 Knowledge of communications procedures associated with EOP implementation.
(CFR: 41.10 / 45.13)
IMPORTANCE RO 3.0 SRO 3.5

This question matches the k/a in that it measures the RO's knowledge of communications performed during EOPs.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.0	Facility Objective:	NONE
Ref Req'd Y or N:	NO	Technical Ref.:	0AP-50
? Cognitive Level:	M OR FK	? Source:	BANK NRC

d. Unless otherwise directed, personnel not involved in the response are to continue their normal activities with the exception of FIRE emergencies. During FIRE emergencies all Hot Work shall be discontinued and placed in a safe condition. All personnel are to report to their assigned work area until the ALL CLEAR is sounded. In the event that the assigned work area is involved in the FIRE emergency, personnel are to report to the lunchroom in the O&M Building. All personnel should listen to the PA for any instructions related to them. They shall not go to the scene of the emergency or to any command station unless specifically requested or required to do so by procedure.

5.2.2 Conduct of Emergency Activity (From Initiation Announcement to Termination Announcement)

1. Main Control Room

The Shift Superintendent is in charge of plant response. The Shift Superintendent will normally be located in the main Control Room, although he or she may leave to respond as required by the emergency.

2. Actions - Main Control Room

a. The Shift Superintendent should provide a brief to Control Room personnel when plant status has changed significantly, periodically to ensure the entire crew is aware of plant status and direction or during a lull in activities. The brief should normally be conducted as follows:

- The Shift Superintendent will announce "In the Control Room - Shift Brief." General information, such as EAL classifications, status of evacuations, or fires may then be provided.

- The Unit SRO will provide plant status. This will normally include where we have been, where we are, and where we are going.

- The Shift Superintendent will close the brief by reporting status of evolutions in progress, e.g., status of repair efforts. Feedback from the crew will be requested to ensure understanding of the information provided and to gain any additional information available.

- The briefing will be closed by announcing "End of Brief."

- Each crew member will acknowledge the brief with "I understand."

b. The Shift Superintendent or Unit SRO may alter the above process as needed based on plant conditions.

c. Annunciators that alarm during briefs should normally be reported to the Unit SRO. For cases where multiple annunciators are being received, the reports may be delayed until after the brief at the discretion of the Unit SCO. If the reporting is delayed, the Unit SCO should ensure the annunciators are directly associated with a recognized event or condition and the reason for the alarm is understood.

d. Orders should **NOT** be given during a brief.

e. All indications that are not understood must be communicated to and acknowledged by the Unit SCO, then investigated.

f. Operators should not talk simultaneously to the Unit SCO, but must interrupt the Unit SCO when critical information must be transmitted immediately (i.e., EOP entry condition is observed, or a reactor scram is imminent).

75. GRO2.4.34 001

The Unit Two (2) SCO has directed that RHR Loop B be placed in Suppression Pool Cooling on Unit Two (2) per AOP-32, Plant Shutdown From Outside the Control Room.

RHRSW pump 2B has been started and an operator at MCC 2XB is standing by to throttle OPEN the RHR Heat Exchanger B Service Water Discharge valve, E11-F068B.

Which ONE of the following describes how the E11-F068B is adjusted properly?

The operator at MCC 2XB will throttle OPEN E11-F068B until informed by an operator:

A. locally at E11-F068B that the desired valve position has been established.

B.✓ at E4 that the desired RHRSW Pump 2B amperage has been established.

C. at the Remote Shutdown Panel (RSDP) that the desired RHRSW flow rate has been established.

D. in the Service Water Building that the desired Nuclear Service Water Discharge Pressure has been established.

Feedback

REFERENCE - 0AOP-32 Rev. 37 page 11

DISTRACTOR ANALYSIS
A. INCORRECT - AOP-32 contains no provision to adjust SW flow in this manner although it could be a plausible method.
B. CORRECT - F068B is opened until RHRSW pump amperage reaches 80 amps.
C. INCORRECT - RHRSW flow indication is not available at RSDP. However RHR flow is available and AOP-32 coordination for RHR flow is similar to as described here.
D. INCORRECT - AOP-32 contains no provision to adjust SW flow in this manner although it could be a plausible method.

Notes

2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications.
(CFR: 43.5 / 45.13)
IMPORTANCE RO 3.8 SRO 3.6

This question matches the k/a in that it measures the RO's knowledge of tasks performed outside the control room.

Categories

Tier:	TIER 3	Group:	RO
Importance Rating:	RO 3.8	Facility Objective:	CLS-LP-302-E*007
Ref Req'd Y or N:	NO	Technical Ref.:	AOP-32
? Cognitive Level:	M OR FK	? Source:	NEW

3.0 OPERATOR ACTIONS

b. Station 1, **CLOSE** the following valves:

- *RHR VITAL SERVICE WATER HEADER WELL WATER SUPPLY VALVE, SW-V141* ☐
- *SERVICE WATER HEADER WELL WATER SUPPLY VALVE, SW-V143.* ☐

c. Station 2, **ENSURE** *RHR HEAT EXCHANGER B SERVICE WATER DISCHARGE VALVE, E11-F002B*, is open at MCC 1(2)XB Compt DN9, Row G4. ☐

d. Station 4, **ENSURE** both Nuclear Service Water Pumps to the affected unit are operating:

NSW Pump	Location	
1A	4160 Bus E1, AF9	☐
1B	4160 Bus E2, AH6	☐
2A	4160 Bus E3, AJ3	☐
2B	4160 Bus E4, AL1	☐

e. Station 4, **START** *RHR SERVICE WATER BOOSTER PUMP B* or *D:*

RHR SW Pump	Location	
1B	4160 Bus E4, AK9	☐
1D	4160 Bus E2, AH4	☐
2B	4160 Bus E4, AK4	☐
2D	4160 Bus E2, AG8	☐

f. Station 4, **MONITOR** the amperage on the Service Water Booster Pump that was started. ☐

g. Station 2, **THROTTLE OPEN** *RHR HEAT EXCHANGER B SERVICE WATER DISCHARGE VALVE, E11-F068B*, at MCC 1(2)XB Compt DN1, Row K4, until the amperage on the running RHR service water pump reaches 80 amps. ☐