SONGS October 2009 NRC Written Examination
Senior Reactor Operator
Answer Key

1.	A	26.	C	51.	A	76.	C
2.	B	27.	A	52.	D	77.	B
3.	B	28.	B	53.	D	78.	C
4.	A	29.	B	54.	A	79.	A
5.	D	30.	C	55.	C	80.	C
6.	A	31.	D	56.	D	81.	D
7.	A	32.	D	57.	A	82.	A
8.	D	33.	B	58.	C	83.	C
9.	B	34.	C	59.	D	84.	D
10.	B	35.	D	60.	C	85.	D
11.	D	36.	A	61.	A	86.	A
12.	C	37.	D	62.	B	87.	B
13.	A	38.	D	63.	B	88.	A
14.	D	39.	B	64.	D	89.	D
15.	C	40.	C	65.	D	90.	A
16.	A	41.	B	66.	A	91.	B
17.	B	42.	D	67.	D	92.	A
18.	C	43.	A	68.	B	93.	B
19.	A	44.	D	69.	C	94.	A
20.	C	45.	C	70.	D	95.	C
21.	B	46.	D	71.	B	96.	C
22.	C	47.	B	72.	A	97.	B
23.	B	48.	C	73.	A	98.	C
24.	B	49.	C	74.	B	99.	A
25.	C	50.	D	75.	C	100.	A

Answer Key Breakdown:

A = 26
B = 24
C = 25
D = 25

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003 G 2.4.11	
Importance Rating	4.0	

Reactor Coolant Pump System: Emergency Procedures/Plan: Knowledge of abnormal condition procedures

Proposed Question: Common 1

Which ONE (1) of the following is an entry condition listed in SO23-13-6, Reactor Coolant Pump Seal Failure?

A. Controlled Bleed-Off temperature is ABOVE normal.

B. Individual seal cavity temperature is RISING during heatup.

C. Controlled Bleed-Off header pressure is LOWERING.

D. Individual seal cavity pressure is RISING during heatup.

Proposed Answer: A

Explanation:

A. Correct. As identified in SO23-13-6, Reactor Coolant Pump Seal Failure Entry Conditions.
B. Incorrect. Plausible because increasing seal cavity temperatures could be an indication of a failure, however, this is not an Entry Condition for SO23-13-6.
C. Incorrect. Plausible because Controlled Bleed-Off header pressure increasing is an Entry Condition for a Reactor Coolant Pump Seal Failure.
D. Incorrect. Plausible because rising seal cavity pressures could be considered an Entry Condition, however, not during a heatup as this would be an expected condition.

Technical Reference(s) SO23-13-6, Entry Conditions Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 94469 / 94468	ANALYZE normal and abnormal operations of the Reactor Coolant System. INTERPRET instrumentation and controls utilized in the Reactor Coolant System.

Question Source: Bank # 75538 (See Comments)
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam SONGS 2000 (replaced Distractor B)

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: From SO23-13-6 Entry Conditions	Revision # 5

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ABNORMAL OPERATING INSTRUCTION
REVISION 5

SO23-13-6
PAGE 2 OF 10

REACTOR COOLANT PUMP SEAL FAILURE

PURPOSE

Specify actions to mitigate the effects of a degraded or failed RCP, RCP Seal, or RCP support system.

ENTRY CONDITIONS

This event is identified by one or more of the following alarms or indications:

1. Individual RCP middle, upper, and vapor seal cavity pressure indications above or below normal.
2. Individual RCP controlled bleed-off temperature indications above normal.
3. Individual RCP controlled bleed-off flow indications above or below normal.
4. RCP controlled bleed-off header pressure increasing on PI-0215.
5. 56B57, RCP BLEED-OFF FLOW HI/LO.
6. 56B58, RCP BLEED-OFF PRESSURE HI.
7. 56C24, RCP P001 SEAL PRESSURE HI/LO.
8. 56C26, RCP P003 SEAL PRESSURE HI/LO.
9. 56C28, RCP P004 SEAL PRESSURE HI/LO.
10. 56C30, RCP P002 SEAL PRESSURE HI/LO.

Comments / Reference: SONGS Exam Bank #75538	Revision # N/A

Modified Distractor B from the following:

(1) Controlled Bleed-Off relief valve temperature constant.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003 K1.10	
	Importance Rating	3.0	

Reactor Coolant Pump System: Knowledge of the physical connections and/or cause-effect relationships between the RCPs and the following systems: RCS

Proposed Question: Common 2

Given the following conditions with the Unit in MODE 5 with loops filled:

- Reactor Coolant System (RCS) temperature is 150°F.
- Steam Generator (SG) temperature is 260°F.
- No Reactor Coolant Pumps are running.
- The Pressurizer is solid.

Which ONE (1) of the following will result if the FIRST Reactor Coolant Pump is started with the conditions listed?

A. Excessive core lift.

B. An RCS pressure transient.

C. Excessive RCP stator temperatures.

D. Exceed RCS heatup limits.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because core lift is a concern when starting the 4th Reactor Coolant Pump, however, in this condition the 1st RCP is being started.

B. Correct. With RCS temperature less then the temperature listed in the Pressure Temperature Limits Report (PTLR) the Steam Generator to RCS ΔT must be less than 100°F. This prevents an overpressure transient on the RCS when the RCP is started and the RCS heats up.

C. Incorrect. Plausible because RCP motor current would be high, however, starting current would only be lowered if the RCP were started at a higher temperature.

D. Incorrect. Plausible because RCS temperature would rise, however, there is insufficient Steam Generator mass to exceed a heatup limit.

Technical Reference(s)		
	SO23-3-1.7, L&S 2.1	Attached w/ Revision # See Comments / Reference
	SO23-5-1.03, Step 6.18.7	
	Tech Spec LCO 3.4.7.4	
	SO23-5-1.3, L&S 4.1 and 4.2	

Proposed references to be provided during examination: None

Learning Objective: 94469 ANALYZE normal and abnormal operations of the Reactor Coolant System.

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5, 14

55.43

Comments / Reference: From SO23-3-1.7, L&S 2.1	Revision # 16

NUCLEAR ORGANIZATION
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OPERATING INSTRUCTION
REVISION 35
ATTACHMENT 16

SO23-3-1.7
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RCP OPERATION LIMITATIONS AND SPECIFICS (Continued)

1.0 **REACTOR COOLANT PUMPS** (Continued)

1.23 To protect CNTMT integrity during an RCP overcurrent event coincident with a loss of DC Control Power, the transformer breaker control power source is normally aligned to the opposite Unit. (Ref. 2.1.4)

1.24 The Reactivity Affecting activities in this procedure which involve only RCS temperature changes are already encompassed by the maintenance of shutdown margin requirements. Consequently, a Reactivity Brief is not required.

2.0 **REACTOR COOLANT SYSTEM**

2.1 In the first few days after entering Shutdown Cooling conditions, the Steam Generators remain hotter than the RCS, which can cause a rapid re-pressurization of the RCS when starting an RCP. The T_{sat} values are used as a convenient way to obtain S/G temperatures by converting from S/G pressure. In the case of low pressures in the S/Gs while on Shutdown Cooling, the S/G temperatures may be substituted for the T_{sat} values used in Step 6.1.16 of the main body. The values used in the step are more conservative than those required in LCO 3.4.6, LCS 3.4.106, LCO 3.4.7, and LCS 3.4.107. (Ref. 2.2.1)

Comments / Reference: From SO23-5-1.3, Step 6.18.7	Revision # 32

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6.0 PROCEDURE (Continued) — PERF. BY INITIALS

6.18 Enter Mode 3

6.18.1 Requisite step completed: 6.17 ________

6.18.2 REVIEW the heatup guidelines and plotting requirements of Attachments 8 and 10. ________

NOTE

The remaining steps in this section may be completed concurrently or in any order.

GUIDELINE

UNIT 2 ONLY: When the RCS is >340°F, then plant heatup rate is controlled at a nominal 20°F/HR. (LS-4.2)

6.18.3 START a third RCP per SO23-3-1.7, Section for Starting A Reactor Coolant Pump. [LS-9.1, LS-9.13] [Mark N/A if already running.] ________

6.18.4 CONTINUE plant heatup to <500°F.

.1 Log **Mode 3 entry** (RCS at 350°F):

DATE ___________ TIME _____ ________

6.18.5 If DSS or DEFAS are INOPERABLE, then Review LCS 3.3.113 and 3.3.114. (Mark N/A if ATWS/DSS and DEFAS are operable.) ________

6.18.6 INITIATE SO23-5-1.3.1, Attachment for Turbine Generator Pre-Roll Checklist. ________

6.18.7 When RCS temperature is > 400°F, then START the 4th RCP per SO23-3-1.7, Section for Starting a Reactor Coolant Pump. ________

Comments / Reference: From Tech Spec LCO 3.4.7.4	Amendment # 203

RCS Loops — MODE 4
B 3.4.6

BASES (continued)

Note 2 requires that either of the following two conditions be satisfied before an RCP may be started with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR.

a. Pressurizer water volume is < 900 ft^3; or

b. Secondary side water temperature in each SG is < 100°F above each of the RCS cold leg temperatures.

Satisfying the above condition will preclude a large pressure surge in the RCS when the RCP is started.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE and has the minimum water level specified in SR 3.4.6.2.

Comments / Reference: From SO23-5-1.3, L&S 4.1 and 4.2	Revision # 32

NUCLEAR ORGANIZATION
UNITS 2 AND 3

INTEGRATED OPERATING INSTRUCTION
REVISION 32
ATTACHMENT 12

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3.0 SAFE SHUTDOWN OPERABILITY REQUIREMENTS

3.1 LIMIT: Exiting a 10CFR50 Appendix R Action Statement by entering a non-applicable Mode prior to the end of the 60-day Action will allow for termination of the compensatory measures. However, reentry into an applicable Mode without restoring the specific component/feature to OPERABLE status will cause the Action Statement to resume at the point in the 60-day period when it was exited.

3.2 LIMIT: Steam Generator Pressure Indication Channels A and B are required for Safe Shutdown. (Tech. Spec. LCS 3.7.113-1)

3.3 Steam Generator Pressure Indication may still be used to meet Safe Shutdown requirements while in bypass or tripped.

3.4 When requesting I&C to make "Live" Channel A or B Steam Generator Pressure Instruments, then any other "Live" pressure channel other than A or B will be simulated per the I&C procedure.

4.0 RCS HEATUP LIMITATIONS

4.1 The RCS **HEATUP** Administrative guideline is 50°F/hr. when Tc ≥70°F (this guideline is more conservative than Tech. Spec. LCO 3.4.3 and LCS 3.4.103 limit of 60°F/hour).

4.2 **UNIT 2 ONLY:** Due to a high rate of S/G tube failure, RCS nominal heatup is 20°F/HR when the RCS is >340°F. This is based on a theory that S/G sleeve collapse is caused by differential expansion rates between the sleeves and the tubes. When water is trapped in this gap, it causes deformation of the sleeve with resultant flow loss. This expansion rate is greater at higher temperatures. Consequently, controlling the heatup rate to a nominal 20°F/HR when the RCS is >340°F should reduce this failure rate. (AR 060102028)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004 A2.25	
	Importance Rating	3.8	

Chemical and Volume Control System: Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Uncontrolled boration or dilution

Proposed Question: Common 3

Given the following conditions:

- Unit 2 is operating in MODE 1 with all systems aligned for normal automatic control when the following alarms are received:
  - 58A04 - VCT LEVEL HI/LO.
  - 58A14 - BORIC ACID PUMPS AUTO START FAILURE.
  - 58A06 - BORIC ACID TO VCT FLOW HI/LO.
- All Reactor Coolant System parameters are stable at normal values.

Which ONE (1) of the following:

1.) Identifies the impact on the Chemical and Volume Control System?

2.) What action must be taken to mitigate the situation?

A. 1.) LT-0226, VCT Level Transmitter has failed high and the Boric Acid Pumps have stopped to prevent over boration of the RCS.
2.) Place FIC-0210Y, Boron Makeup Flow Controller in MANUAL to ensure boration remains secured.

B. 1.) LT-0226, VCT Level Transmitter has failed low and the selected Boric Acid Pump failed to start.
2.) Place HS-0210, Makeup Mode Selector in MANUAL to stop the uncontrolled dilution.

C. 1.) LT-0227, VCT Level Transmitter has failed high and the Boric Acid Pumps have stopped to prevent over boration of the RCS.
2.) Place FIC-0210Y, Boron Makeup Flow Controller in MANUAL to ensure boration remains secured.

D. 1.) LT-0227, VCT Level Transmitter has failed low and the selected Boric Acid Pump failed to start.
2.) Place HS-0210, Makeup Mode Selector in MANUAL to stop the uncontrolled dilution.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the VCT HI / LO alarm would come in on a high failure of LT-0226 and it could be thought that a boration evolution would need to be stopped if in progress and placing FIC-0210Y in MANUAL would prevent normal boration.
B. Correct. VCT Level transmitter LT-0226 failing low causes Auto Makeup to initiate to the VCT and the failure of the selected Boric Acid Pump to start results in the pump failure and low flow alarms and is an indication that the only makeup flow is a dilution.
C. Incorrect. Plausible because the VCT HI / LO alarm would come in on a high failure of LT-0227 and placing FIC-0210Y in MANUAL could prevent an advertent boration.
D. Incorrect. Plausible because the Makeup Mode Selector should be placed in MANUAL, however, the failure of transmitter LT-0227 causes the Charging Pump suction to shift from the VCT to the RWST.

Technical Reference(s): SO23-15-58.A, 58A04; SO23-15-58.A, 58A06; SO23-15-58.A, 58A14; SD-SO23-390, Sect 2.2.21 & Figure I-1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 102549 DESCRIBE the configuration and operational characteristics of Chemical and Volume Control System components.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments / Reference: From SO23-15-58.A, 58A04	Revision # 10

58A04 VCT LEVEL HI/LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	WHITE	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)LSH-0226 2(3)LSL-0226	2(3)MT-077 Level Switch High 2(3)MT-077 Level Switch Low	HI 80% LO 35%	2(3)LI-0226B 2(3)LR-0226 2(3)LI-0226A 2(3)LI-0227	L226 L227	686/708

1.0 **REQUIRED ACTIONS**:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS: (Continued)

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
LEVEL LOW	
2.3 VCT auto level control failure (If VCT is < 32% and no auto makeup has occurred).	2.3 Manually raise VCT level per SO23-3-2.2, Section for Manual Blended Makeup Mode.
2.4 Loss of RCS inventory	2.4 If VCT level has dropped to 32%, then verify auto makeup is in progress. 2.4.1 RESTORE Pressurizer level to normal operating band per SO23-3-1.10, Section for Normal Pressurizer Level Control. 2.4.2 If unable to maintain VCT level, then refer to SO23-13-14, Reactor Coolant Leak.
2.5 LT-0226 failed low	2.5 Consider placing Makeup Mode Selector Switch HS-0210, to MANUAL.

Comments / Reference: From SO23-15-58.A, 58A06	Revision # 10

NUCLEAR ORGANIZATION
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ALARM RESPONSE INSTRUCTION
REVISION 10
ATTACHMENT 2

SO23-15-58.A
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58A06 BORIC ACID TO VCT FLOW HI/LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	N/A	58A14

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)FSHL-0210Y	Reactor Make Up Water-Boric Acid Make Up Flow Deviation Alarm	± 1 gpm of controller setpoint (100 sec Time Delay) [1]	2(3)FQI-0210Y	F210Y	688/710

1.0 REQUIRED ACTIONS:

1.1 If unable to obtain required boric acid flow during automatic or manual operation of 2(3)FIC-210Y, then SECURE flow to prevent unplanned boration or dilution of the Reactor Coolant System.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
	NOTE: If in the manual make up mode, then the boric acid pump has to be manually started.
2.1 Boric Acid Pump not running	2.1 ENSURE that a Boric Acid Pump is running during make up per SO23-3-2.2, Makeup Operations.
2.2 2(3)FIC-0210Y, Makeup Flow Controller fails to control 2(3)FV-0210Y, Boric Acid Makeup Discharge Flow Control Valve.	2.2 ENSURE 2(3)FIC-0210Y: • Controller energized • AUTO is selected • "OOS" is not displayed on controller 2.2.1 If "OOS" is displayed on the controller, then notify the I&C Department to reprogram the controller.

[1] Alarm is disabled when 2(3)HS-0210, Handswitch for Demin Water and Boric Acid Make Up to the VCT, is in the DILUTE POSITION.

Comments / Reference: From SO23-15-58.A, 58A14	Revision # 10

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ALARM RESPONSE INSTRUCTION REVISION 10 ATTACHMENT 2 | SO23-15-58.A PAGE 38 OF 131

58A14 BORIC ACID PUMPS AUTO START FAILURE

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	WHITE	NO	58A46, 58A47

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)PSL-0206	2(3)P-175 Discharge	85 PSIG	NONE	NONE	696/718 697/719 698/720
2(3)PSL-0208	2(3)P-174 Discharge	85 PSIG			

1.0 REQUIRED ACTIONS:

1.1 PLACE 2(3)HS-0210, Makeup Mode Selector, in MANUAL. |

1.1.1 Terminate the makeup evolution per the in-use section of SO23-3-2.2. |

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Failure of the Selected BAMU Pump to Auto Start	2.1 TRANSFER to the other BAMU Pump per SO23-2-19, Section for Rotation of Auxiliary Plant Equipment - Unit 2(3).

3.0 ASSOCIATED RESPONSES:

3.1 POSITION 2(3)HS-0210, Makeup Mode Selector to AUTO.

3.2 Notify the CRS/SM and the STA to review Tech. Specs. LCO 3.1.9, LCO 3.1.10, and initiate an EDMR/LCOAR as required.

Comments / Reference: From SD-SO23-390, Page 53	Revision # 17

NUCLEAR ORGANIZATION SYSTEM DESCRIPTION SD-SO23-390
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PART I LETDOWN AND CHARGING

2.0 DESCRIPTION (Continued)

2.2 Components (Continued)

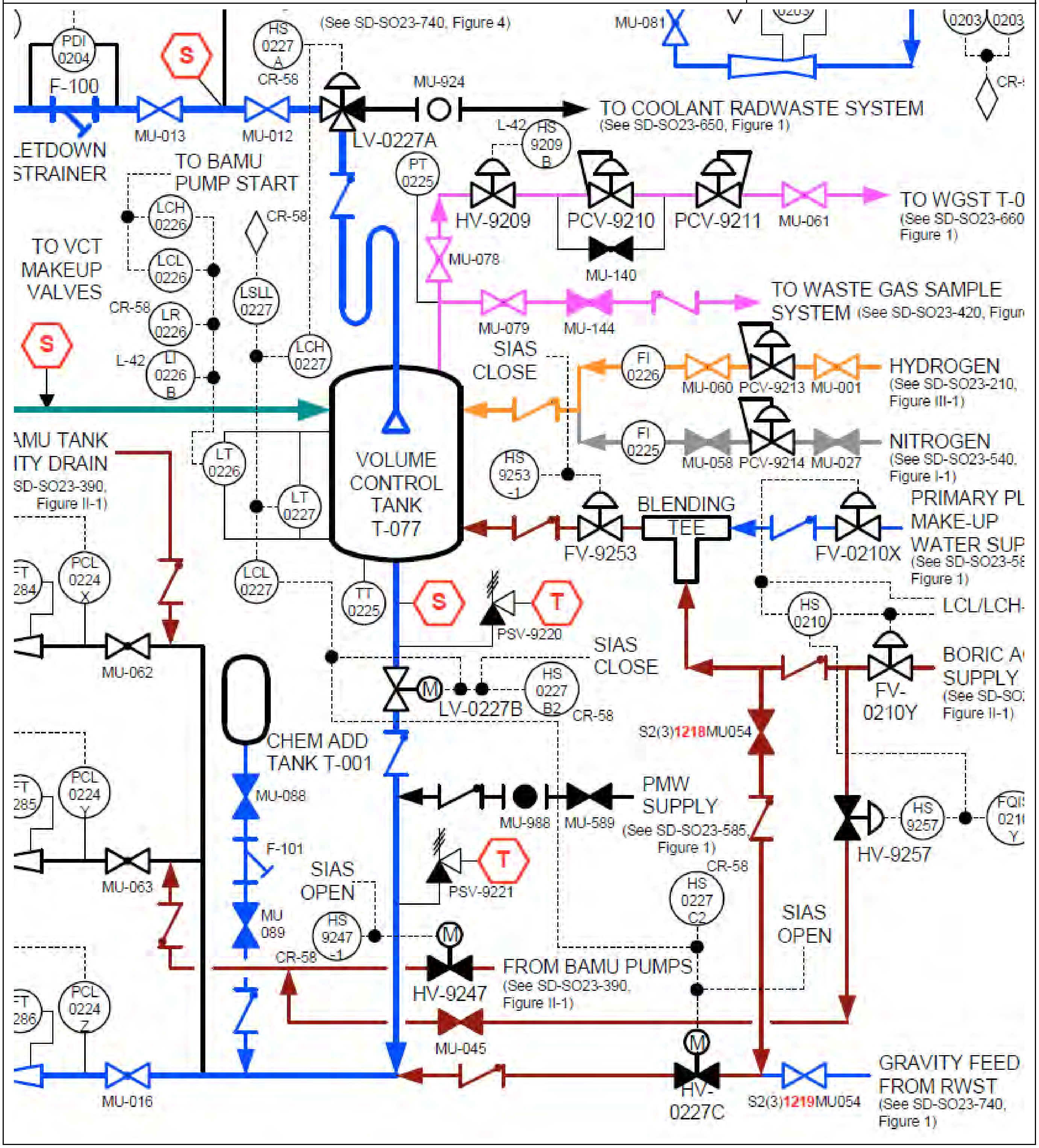
2.2.21 **Volume Control Tank (VCT), 2(3)T-077** (Figures I-1 & 13) (Continued)

.10 Differential Pressure Level Instruments provide VCT, 2(3)T-077 Level Indication, 2(3)LI-0226A, Level Recorder, 2(3)LR-0226 on 2(3)CR-58, and controls for Automatic Makeup System.

.10.1 2(3)L1-0226A is also provided on Evacuation Shutdown Panel 2(3)L-042.

.10.2 Differential Pressure Level Instrument, 2(3)LT-0227, provides a signal to operate VCT Valves 2(3)LV-0227A, 0227B, and 0227C.

Comments / Reference: From SD-SO23-390, Figure I-1	Revision # 17



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005 K5.03	
Importance Rating	2.9	

Residual Heat Removal System: Knowledge of the operational implications of the following concepts as they apply to the RHRS: Reactivity effects of RHR fill water

Proposed Question: Common 4

Given the following conditions during Refueling:

- Unit 3 is in MODE 6 and core re-load has been completed.
- While filling and venting a portion of the out-of-service Shutdown Cooling Train, an inadvertent dilution occurred.
- This lowered Refueling Cavity and RCS boron concentration from 2672 ppm to 2587 ppm.

Which ONE (1) of the following describes the indications observed as a result of this dilution?

A. Slight rise in cavity level and a small rise in source range count rate.

B. Slight rise in Core Exit Thermocouple temperature and a doubling in source range count rate.

C. Small rise in dose rates around Shutdown Cooling Equipment and a small rise in source range count rate.

D. Small drop in dose rates around Shutdown Cooling Equipment and a doubling in source range count rate.

Proposed Answer: A

Explanation:

A. Correct. The only level change will be associated with the amount of water added and the count rate changes would be small given that the previously existing shutdown margin was > 5% (Keff ≤ .95) and the IBW would be about 105 ppm/% at BOL. There also would be no significant increase in N-16 gamma production to cause coolant dose rates to rise.

B. Incorrect. Plausible because it could be thought that the boron change was large enough to cause a doubling and cause an increase in nuclear heat, however, there would be no CET change until the Reactor was at the POAH.

C. Incorrect. Plausible because it could be thought that the change in core count rate would also be reflected in the coolant dose rate. There also would be no significant increase in N-16 gamma production to cause coolant dose rates to rise.

D. Incorrect. Plausible because dilution of the RCS could slightly reduce RCS specific activity and it could be thought that the boron change was large enough to cause a doubling.

Technical Reference(s) Operation Physics Summary, Fig 3.2-4 Attached w/ Revision # See Comments / Reference
LP 2AO711, Pages 13 & 14

Proposed references to be provided during examination: None

Learning Objective: 52680 EXPLAIN how and why each of the following will change with successive equal positive reactivity insertions into a subcritical reactor: Factor by which count rate increases.

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

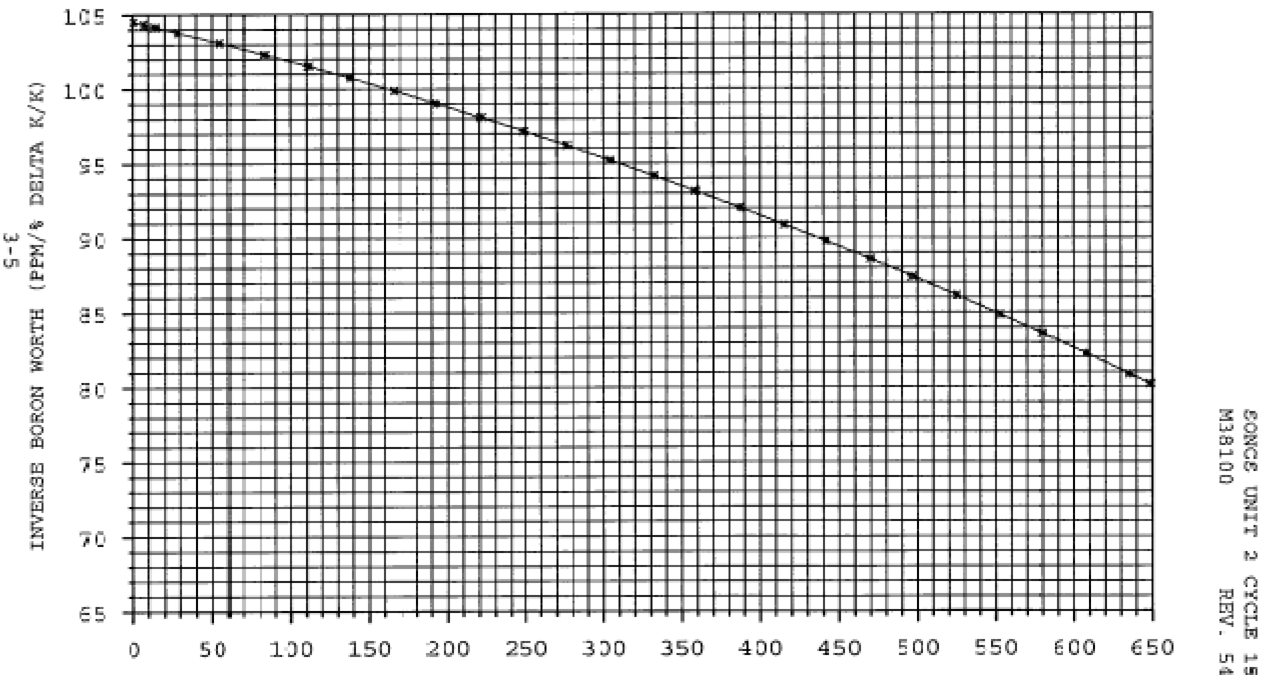
Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 1, 5
55.43

Comments / Reference: From Operation Physics Summary, Fig 3.2-4	Revision # 54

FIGURE 3.2-4
SONGS UNIT 2 CYCLE 15
INVERSE BORON WORTH (ZP, 68 F)



Comments / Reference: From LP 2AO711, Page 13	Revision # 5-4

6.1.2 Inadvertent Dilution

.1 Procedure entry conditions:

.1.1 Unexplained rise in reactor power

.1.2 Unexplained rise in RCS temperature

.1.3 Unexpected lowering of RCS boron concentration

.1.4 Unexplained increase in count rate when reactor is shut down

.2 Relatively rapid boron dilution can occur if one of the following conditions are present:

.2.1 Blended makeup to the VCT with BAMU pumps unavailable or the VCT Blend setpoints aren't properly set

.2.2 Placing an IX in service without being fully saturated to RCS boron concentration (or placing the deborating IX in service).

.2.3 Failure of Letdown Temperature controller to maintain temperature such that temperature decreases. This causes an increased affinity for boron absorption leading to a dilution.

.2.4 PMW system (pump/valve) failure

.3 Gradual boron dilutions

Comments / Reference: From LP 2AO711, Page 14	Revision # 5-4

.3.1 SOER 94-2 delineated 7 events had occurred during a one year period due to:

.3.1.1 Preparing demineralizers for service

.3.1.2 Routine boron dilution activities near full power

.3.2 Gradual Boron Dilutions during shutdown are of particular concern.

.3.2.1 RCS may be in a (reduced inventory) condition

.3.2.2 Plant activities increase methods/opportunity of diluting RCS.

.3.2.3 New fuel with higher enrichment.

.3.2.4 Safety systems may be disabled e.g. (RPS trips)

.4 Actions

.4.1 If Refueling in Progress

.4.1.1 Suspend core alterations or positive reactivity additions

.4.1.2 If cavity is being filled, stop fill and verify Boron Concentration per requirements

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005 K2.01	
Importance Rating	3.0	

Residual Heat Removal System: Knowledge of bus power supplies to the following: RHR pumps

Proposed Question: Common 5

Given the following conditions:

- Unit 2 is at 100% power.
- Unit 3 is cooling down in MODE 4.
- 230 kV Switchyard is in a normal alignment.
- All 4160 VAC 1E Bus voltages have lowered to 3950 V.

After 30 seconds has elapsed, which ONE (1) of the following is the power supply to Unit 3 Shutdown Cooling Pump 3P016?

A. Reserve Auxiliary Transformer 2XR2 via Bus 2A06.

B. Emergency Diesel Generator 2G003 via Bus 2A06.

C. Emergency Diesel Generator 3G003 via Bus 3A06.

D. Reserve Auxiliary Transformer 3XR2 via Bus 3A06.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Reserve Auxiliary Transformer on the opposite Unit (2XR2) is the Preferred Power Source, however, when a degraded voltage exists on both Units (Switchyard in a normal alignment) this transfer would not take place.

B. Incorrect. Plausible because this power source would be available, however, it requires manual actions by the operator including operation of 50.54.X switches in the associated Switchgear Room. Additionally, Bus 3A06 EDG (3G003) would need to be INOPERABLE to take this action.

C. Incorrect. Plausible because had this condition lasted for longer than 110 seconds the Emergency Diesel Generator would be the source of power, however, power to Shutdown Cooling Pump 3P016 remains on the Reserve Auxiliary Transformer.

D. Correct. This is the power supply to Shutdown Cooling Pump 3P016 until the degraded voltage condition lasts for longer than 110 seconds.

Technical Reference(s)	SD-SO23-120, Page 109	Attached w/ Revision # See Comments / Reference
	SD-SO23-120, Figure III-1	
	SD-SO23-120, Page 154	
	SO23-15-63.C, 63C05	

Proposed references to be provided during examination: None

Learning Objective: 79744 — DESCRIBE the operation of the Pumps, Tanks, and remotely operated Valves of the CIS, CSS, and SIS. Include the controls, function, location, and specific features such as type, capacity, and power supplies where applicable.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	7, 8
	55.43	

Comments / Reference: From SD-SO23-120, Page 109	Revision # 19

PART III 1E 4.16 kV AND 480 V ELECTRICAL DISTRIBUTION SYSTEM

2.0 DESCRIPTION (Continued)

2.1.4 **General Control Scheme**

1. A description of the automatic transfer capability that exists between the 4.16 kV 1E buses of each Unit is discussed below. For simplicity, this discussion covers only one load group (2A04). However, similar operations take place on the redundant load group and the load groups associated with the other unit.

2. Bus 2A04 is normally supplied from Reserve Auxiliary Transformer (2XR1). If power from the Reserve Auxiliary Transformer (2XR1) to bus 2A04 is lost, the following actions take place:

3. The LOVS or SDVS (Sustained Degraded Voltage System) also sends a signal to start Diesel Generator 2G002.

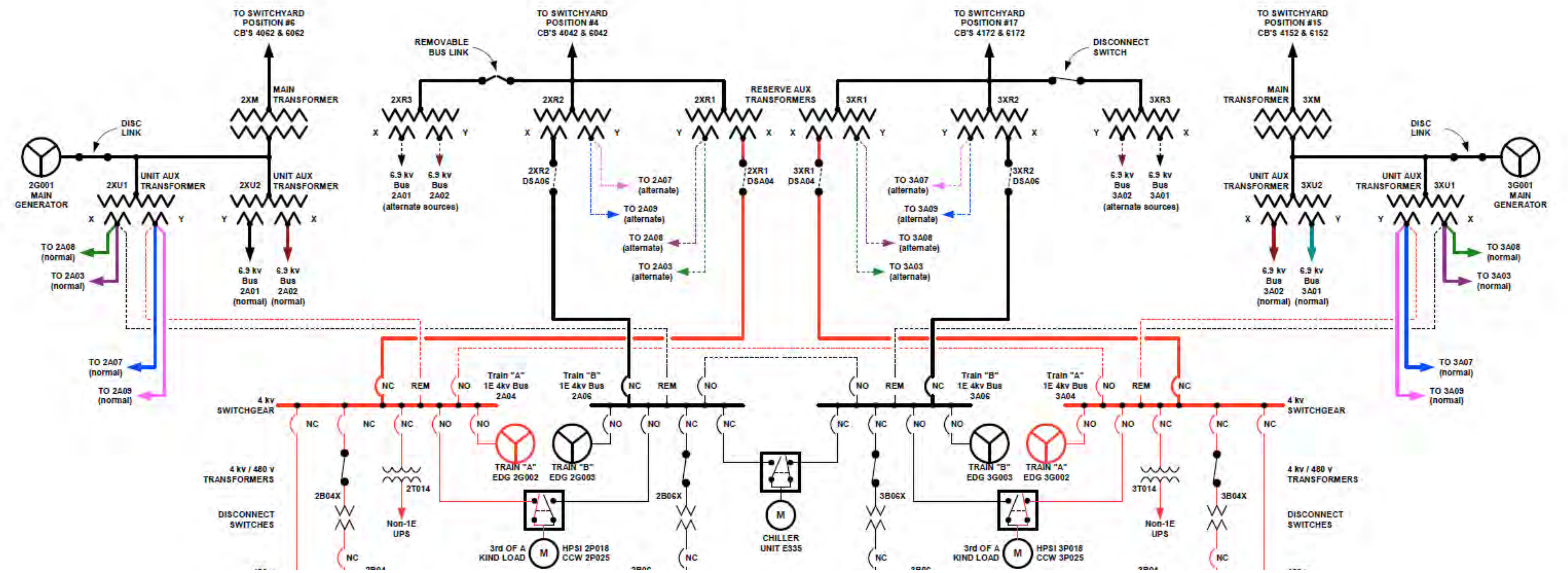
4. After the residual voltage at bus 2A04 has decayed to approximately 25%, as detected by bus 2A04 residual voltage relays, the LOVS or SDVS signals the Unit 3 bus tie circuit breaker 3A04-16 to close provided bus 3A04 has normal voltage and is being powered from its respective Reserve Auxiliary Transformer 3XR1 or Unit Auxiliary Transformer 3XU1.

5. After 3A04-16 closes, Unit 2 bus tie circuit breaker 2A04-17 closes and bus 2A04 will be powered from Reserve Auxiliary Transformer 3XR1, through bus 3A04.

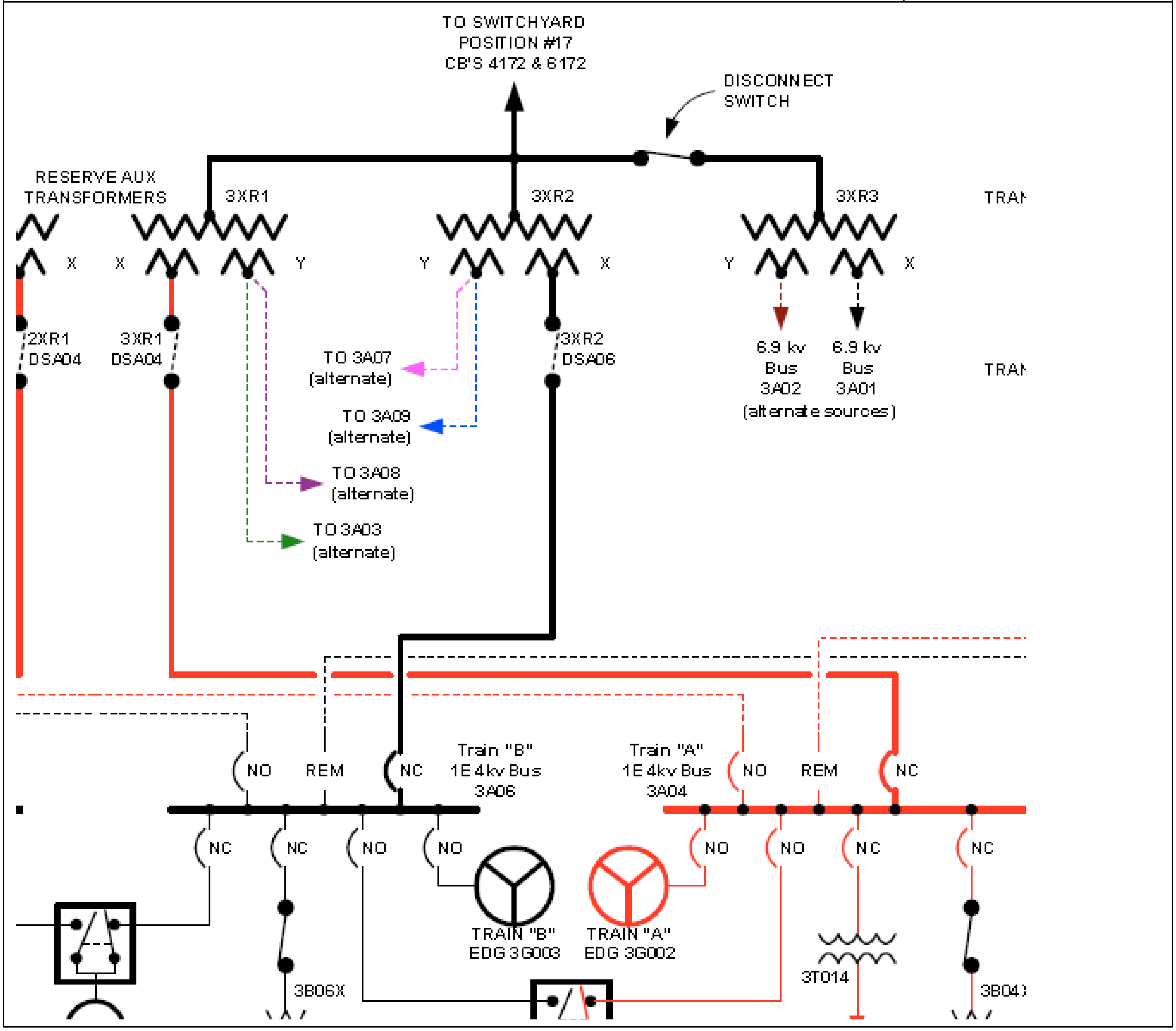
6. If bus 3A04 is not being supplied by Reserve Auxiliary Transformer 3XR1 as detected by Reserve Auxiliary Transformer breaker (3A04-18) not being closed or if bus 3A04 has no voltage, the transfer will not be permitted and bus tie breaker 3A04-16 will not close. Additionally an interlock prevents crosstie circuit breaker 3A04-16 from closing if the Unit 3 Diesel Generator (3G002) circuit breaker 3A04-13 is closed and the Diesel Generator is supplying its designated load group. This prevents the Diesel Generator from supplying two load groups and overloading the Diesel Generator, since the Diesel Generator is only rated to carry the loads associated with one load group. However, if Diesel Generator (3G002) is paralleled with the Reserve Auxiliary Transformer (3XR1) during a periodic load test, a LOVS or SDVS at Bus 2A04 will initiate a transfer to Bus 3A04 and the Unit 3 Diesel Generator breaker 3A04-13 will be tripped.

Comments / Reference: From SD-SO23-120, Figure III-1	Revision # 19

FIGURE III-1: 1E 4.16 kV ELECTRICAL DISTRIBUTION SYSTEM



Comments / Reference: From SD-SO23-120, Figure III-1	Revision # 19



Comments / Reference: From SD-SO23-120, Page 154	Revision # 19

PART III 1E 4.16 kV AND 480 V ELECTRICAL DISTRIBUTION SYSTEM

3.0 OPERATIONS (Continued)

3.3.6 **Cross-Tie to the Opposite Unit's Diesel Generator** (Continued)

7. In this condition, to establish the cross-tie connection between bus 2A04 and bus 3A04, an operator must manually select 2HV-5054XA1 and 2HS-5054XB1 at Exposure Fire Isolation Panel 2L412 in the Unit 2 1E switchgear room at El. 50' to the "50.54X" position. In the Unit 3 1E switchgear room, the operator must also manually select 3HS-5054XA1 and 3HS-5454XB1 at Exposure Fire Isolation Panel 3L412 to the "50.54X" position.

8. The following functions are achieved upon placing the 2HS-5054XA1, 3HS-5454XB1, 3HS-5054XA1, and 3HS-5054SB1 in the "50.54X" position:

Comments / Reference: From SO23-15-63.C, 63C05	Revision # 8

NUCLEAR ORGANIZATION UNIT 2 — ANNUNCIATOR RESPONSE INSTRUCTION REVISION 8 ATTACHMENT 2 — SO2-15-63.C PAGE 14 OF 136

63C05 2A06 VOLTAGE LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	RED [1]	N/A	63C15, 63C21, 63C25, 63C28, 63C35

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK #
127 Device	Undervoltage Device	N/A	2EI-1641	EY8183	1704

1.0 **REQUIRED ACTIONS:**

1.1 Notify the Unit 2 CRS/CO of the low voltage alarm.

1.2 Verify voltage indication and perform the following:

1.2.1 If 2A06 Control Room voltage indication is lost, and loads are still operating, then a C-A Phase PT failure may have occurred. (AR 031000894-2)

VOLTAGE IS > 4154

1.2.2 Locally inspect 2A0617 (PT Cubicle) for any Degraded Voltage Relays (127D-1, 2, 3 & 4) tripped (light ILLUMINATED).

.1 Notify the CRS/SM and the STA to review Tech. Spec. LCO 3.3.7, and INITIATE corrective actions, as required.

.2 Notify Electrical Test Department.

VOLTAGE IS > 3796 and ≤ 4154

NOTE

When the Diesel Generator Output breaker is closed, then the SDVS (Sustained Degraded Voltage Signal) circuitry is defeated. If the Diesel Generator is in parallel with the preferred power source and confirmed degraded voltage condition exists, then the Diesel Generator Output breaker must be opened to allow the SDVS timing relays to auto sequence (110 ± 22 sec.) to protect Class 1E equipment.

1.2.3 Degraded Grid Voltage condition exists. (110 ± 22 second timer starts when the alarm annunciates.)

.1 Unload the Diesel Generator.

.2 ENSURE OPEN Diesel Generator Output Breaker 2A0613.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 K5.07	
	Importance Rating	2.7	

Emergency Core Cooling System: Knowledge of the operational implications of the following concepts as they apply to ECCS: Expected temperature levels in various locations of the RCS due to various plant conditions

Proposed Question: Common 6

Given the following conditions:

- A Large Break Loss of Coolant Accident (LOCA) has occurred.
- The break location is a Reactor Coolant System Hot Leg.

Which ONE (1) of the following would be the result if the steps of SO23-12-11, EOI Supporting Attachments, Attachment 11, Simultaneous Hot/Cold Leg Injection were performed immediately following the LOCA diagnosis?

A. Inadequate core cooling due to the loss of Safety Injection flow out the break.

B. Safety Injection flow through the core would be stopped.

C. Boration flow would be inadequate to maintain core SHUTDOWN MARGIN.

D. High Pressure Safety Injection Pumps would reach run out conditions.

Proposed Answer: A

Explanation:

A. Correct. Failure to wait two hours prior to initiating Simultaneous Hot/Cold Leg Injection would result in an Inadequate Core Cooling situation. This is due to the fluid injected at the Hot Leg getting entrained in the steam leaving the break in the early period of the LOCA.
B. Incorrect. Plausible because Safety Injection flow from the Hot Leg would be impeded due to steam leaving the core area, however, Cold Leg flow would still be ensured.
C. Incorrect. Plausible because performing Attachment 11 would result in effectively reducing the boration rate, however, boration flow would still exceed the minimum required to maintain SDM.
D. Incorrect. Plausible because under certain core conditions the HPSI Pumps can exceed runout. Attachment 11, Simultaneous Hot/Cold Leg Injection controls the flow rate through the Cold and Hot Leg Injection Valves to prevent this condition from occurring. This issue is addressed in SO23-12-11, Attachment 13.

Technical Reference(s)		
	SO23-14-11, Attachment 11, Step 1a Bases	Attached w/ Revision # See Comments / Reference
	SO23-12-11, Attachment 11, Step 3 Caution	
	SO23-14-11, Attachment 13, Bases	

Proposed references to be provided during examination: None

Learning Objective: 54933 — Per the LOCA procedure SO23-12-3 DESCRIBE: The basis for each step, caution or note.

Question Source: Bank # 75372
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 5, 10
55.43 ______

Comments / Reference: From SO23-14-11, Attachment 11, Step 1a Bases	Revision # 2

EOI SUPPORTING ATTACHMENTS BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.14 ATTACHMENT 11, SIMULTANEOUS HOT / COLD LEG INJECTION

Simultaneous injection into both hot and cold legs is used as a flushing mechanism to prevent the precipitation of Boric Acid in the Reactor Vessel.

Due to plant specific Safety Injection flow instrument design features, and Safety Injection system design features, there are four different Hot/Cold Leg Injection flowpaths within this attachment. These four flowpaths can be defined by the existing RCS pressure and the number of HPSI pumps that are operating. These flowpaths are:

1) Two HPSI pumps are operating and RCS pressure is greater than 500 PSIA
2) Two HPSI pumps are operating and RCS pressure is 500 PSIA or less
3) One HPSI pump is operating and RCS pressure is greater than 500 PSIA
4) One HPSI pump is operating and RCS pressure is 500 PSIA or less.

Injecting to both sides of the Reactor Vessel ensures that fluid from the Reactor Vessel (where the Boric Acid is being concentrated) flows out of the break regardless of the break location and is replenished with a dilute solution of borated water from the other side of the Reactor Vessel.

More than 2 hours and less than 4 hours must have elapsed following SIAS initiation to initiate this attachment, as directed by FS-25, MONITOR Need for Simultaneous Hot/Cold Leg Injection. The realignment to hot/cold leg injection is unnecessary, and you will not even be directed to this attachment, if entry into SDC is expected prior to 4 hour after SIAS initiation.

Step 1a.: Verifies at least 2 hours have elapsed since SIAS actuation. Prior to 2 hours, the fluid injected into the hot leg may be entrained in the steam being released from the break and possibly diverted from reaching the Reactor Vessel. After 2 hours, Core decay heat has dropped sufficiently such that there is insufficient steam velocity to entrain the fluid being injected into the hot leg. The action is taken no later than 4 hours after SIAS in order to ensure that the buildup of Boric Acid is terminated well before the potential for boron precipitation occurs.

Comments / Reference: From SO23-12-11, Attachment 11, Step 3 Caution	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 6
ATTACHMENT 11

SO23-12-11 ISS 2
PAGE 146 OF 278

EOI SUPPORTING ATTACHMENTS

SIMULTANEOUS HOT / COLD LEG INJECTION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 VERIFY Entry Conditions:	
a. ENSURE time elapsed from SIAS actuation – greater than 2 hours.	
b. VERIFY FS-7, VERIFY SI Throttle/Stop Criteria – NOT satisfied.	b. IF FS-7, VERIFY SI Throttle/Stop Criteria – met, OR IF Injection valves THROTTLED to maintain SI Throttle/Stop criteria, THEN EXIT this attachment.
2 ENSURE SDC Valves Closed:	
a. ENSURE SDC To LPSI Pump Suction Isolation valves – closed: HV-9337 HV-9377 HV-9339 HV-9378.	

CAUTION

An operating HPSI Pump flow in excess of 910 GPM per pump may exceed pump run-out.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3 VERIFY HPSI Operability:	
a. VERIFY both trains of HPSI – operating.	a. GO TO step 11.

Comments / Reference: From SO23-14-11, Attachment 13, Bases	Revision # 2

EOI SUPPORTING ATTACHMENTS BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.15 ATTACHMENT 12, MINIMUM REQUIRED SI FLOWRATES DURING COLD LEG INJECTION

This attachment shows the minimum required HPSI flow per injection path and the minimum required total LPSI flowrate that will ensure the system design basis is met during the Cold Leg Injection mode. The table was done per injection point to better match Control Board indications. (LPSI flow is measured at pump discharge common header.)

As a minimum, one HPSI Train (one HPSI Pump and its four injection valves), and one LPSI Train (one LPSI Pump and two injection valves), operating are required to meet the system design basis.

Worst case single failure analysis has shown that for a small break LOCA, loss of one Emergency Diesel Generator (EDG) is most limiting. With the failure of one EDG, the plant will lose one HPSI pump, one LPSI pump and two LPSI injection valves. Thus, LPSI is injecting to two legs and HPSI is injecting to four legs. The assumption is made that the break is on the discharge piping of one RCP with an open injection path. Total flow to the Core is approximately 75% of total HPSI and approximately 50% of total LPSI.

During a RAS Actuation the LPSI pumps are secured. A single HPSI pump will meet the decay heat removal requirements at the start of recirculation.

The table was developed using the best estimate values for SI flow provided by CE. These values were placed on a spreadsheet and scales changed to provide values in 50 PSI increments. The wider range give operators more data points to evaluate expected SI flow.

A comparison has been made between this new SI flow table and the similar table in the FSAR. The SI flow table in the EOI is conservative at each point to UFSAR table 6.3-5. (Ref. 2.1.3)

4.16 ATTACHMENT 13, MINIMUM REQUIRED HPSI FLOWRATES DURING HOT / COLD LEG INJECTION

This table is to be used during the Simultaneous Hot/Cold Leg Injection Mode as part of post-LOCA long-term cooling. The table provides the flow required two hours post-LOCA for Core flushing to avoid the precipitation of boric acid in the Core. Additional considerations are the decay heat two hours post-LOCA and a maximum indicated flow of 910 GPM per pump to avoid pump runout.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	007 K4.01	
Importance Rating	3.3	

Pressurizer Relief / Quench Tank System: Knowledge of the PRTS design feature(s) and/or interlock(s) which provide for the following: Quench tank cooling

Proposed Question: Common 7

Given the following condition:

- Annunciator 50A21 - QUENCH TANK TEMP HI has just alarmed.

Which ONE (1) of the following alignments is used to cool the Quench Tank?

The Quench Tank is...

A. drained to the Reactor Coolant Drain Tank and refilled with Primary Makeup Water.

B. vented to the Waste Gas System while the contents cool to ambient temperature.

C. drained to the Containment Sump and refilled with Primary Makeup Water.

D. vented and drained to the Radwaste Primary Tank using the Reactor Coolant Drain Tank Pumps.

Proposed Answer: A

Explanation:

A. Correct. This is the guidance provided in Annunciator Response Procedure 50A21. There is no internal method used to cool the Quench Tank.
B. Incorrect. Plausible because this is the required action to address a high Quench Tank pressure, however, a high temperature requires drain and refill of the Quench Tank.
C. Incorrect. Plausible because the Primary Makeup Water Pump is used, however, the effluent is directed to the Reactor Coolant Drain Tank vice Containment Sump.
D. Incorrect. Plausible because this is where the coolant is pumped from the Reactor Coolant Drain Tank, however, this does not address the high temperature and cooling of the Quench Tank.

Technical Reference(s): SO23-15-50.A1, 50A21; SD-SO23-360, Figure I-6; SD-SO23-650, Figure 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 94467 / 94465

DESCRIBE the configuration and operational characteristics of the Reactor Coolant System components.
IDENTIFY Reactor Coolant System flowpaths, components and locations including being able to draw and label system diagrams.

Question Source: Bank # 127252
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 3, 10
55.43 ______

Comments / Reference: From SO23-15-50.A1, 50A21 — Revision # 8

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ALARM RESPONSE INSTRUCTION REVISION 8 ATTACHMENT 2 — SO23-15-50.A1 PAGE 48 OF 64

50A21 QUENCH TANK TEMP HI

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	WHITE	N/A	50A31

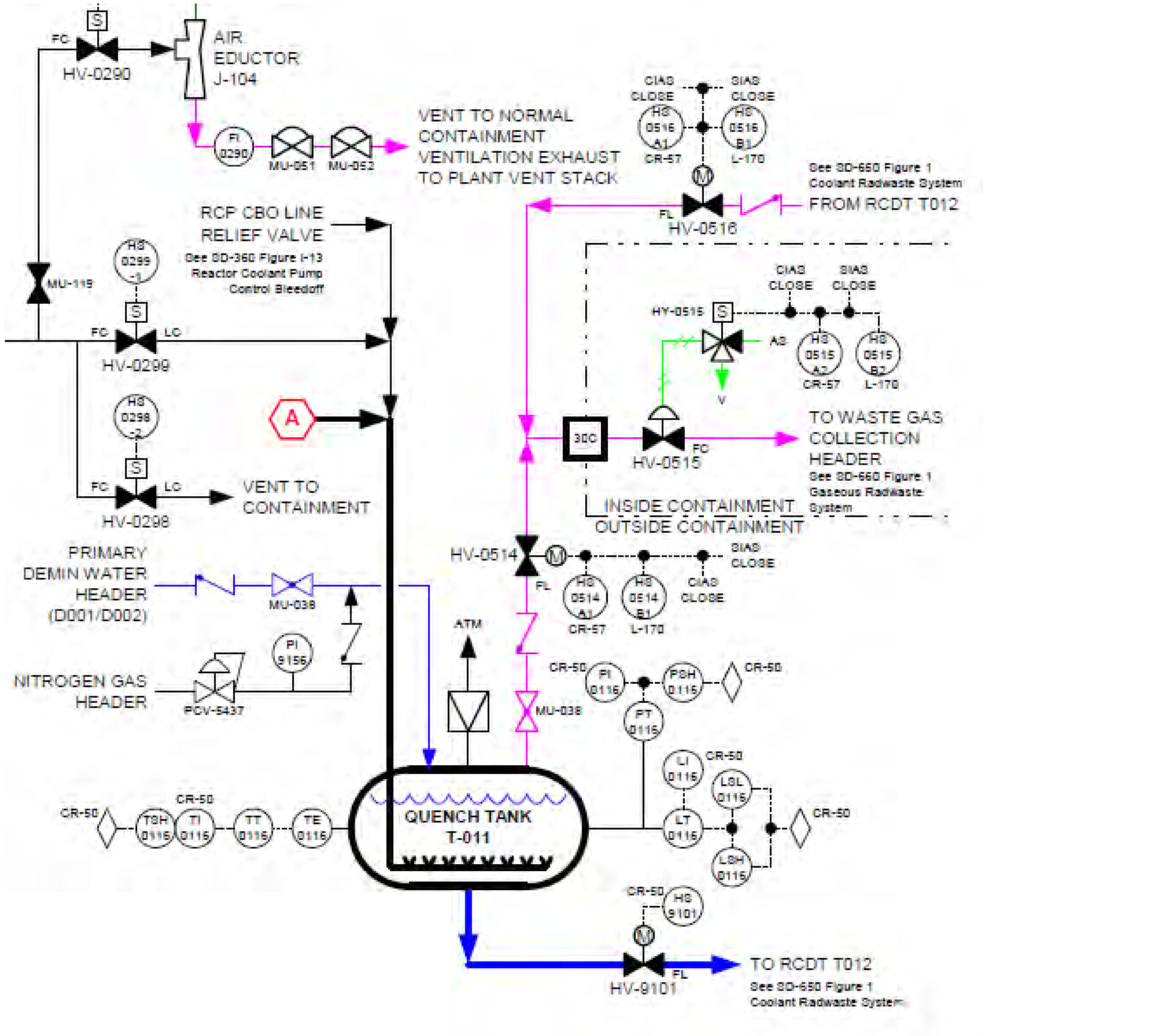
INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)TSH-0116	Quench Tank Temperature High	200°F	2(3)TI-0116	T116	631/653

1.0 REQUIRED ACTIONS:

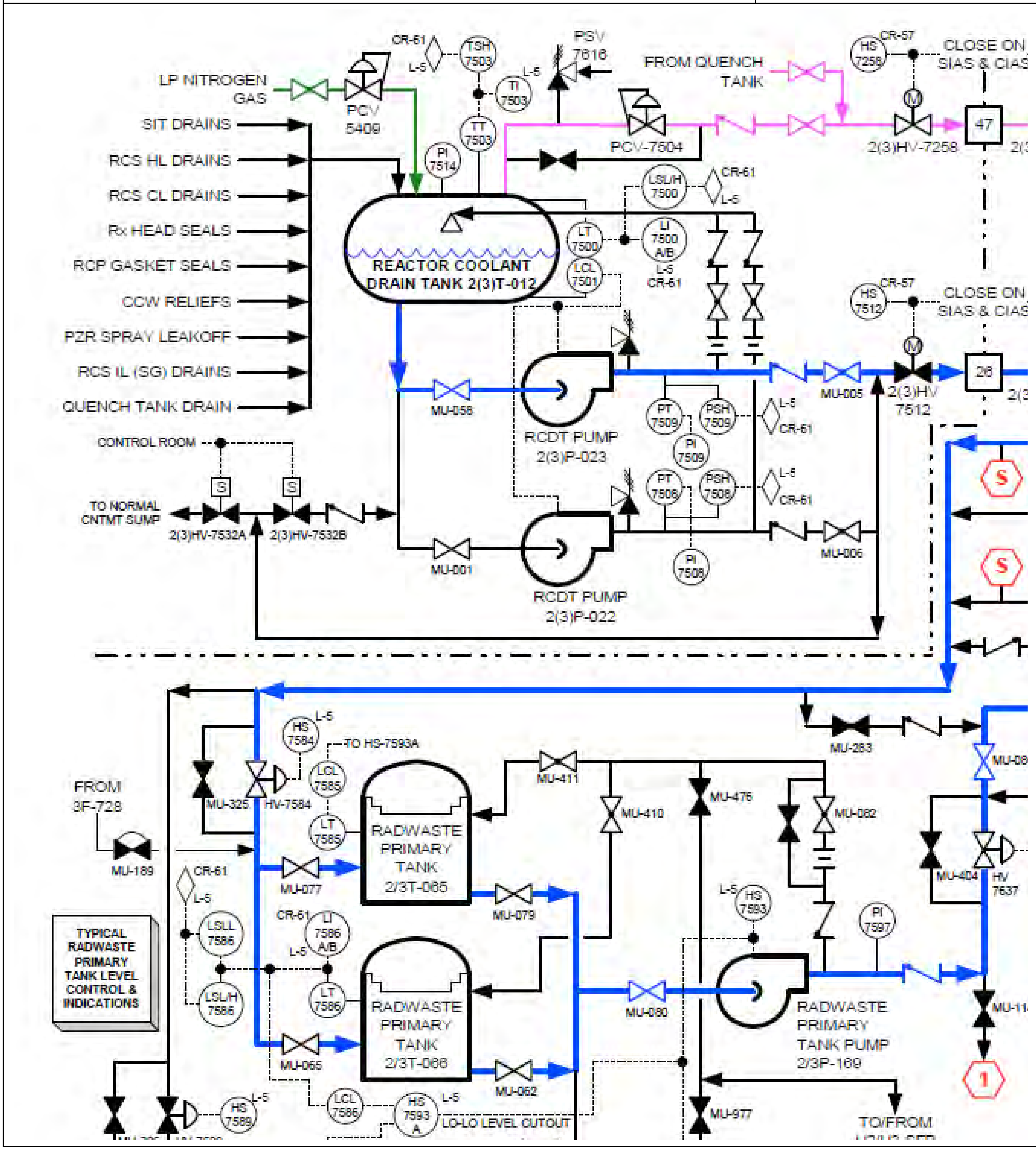
1.1 Cool the Quench Tank by concurrently Draining the Quench Tank to the RCDT through HV-9101, and Making Up through S2(3)1901MU321.

1.1.1 Control Quench Tank pressure by Venting the Quench Tank to the Waste Gas System through 2(3)HV-9100.

Comments / Reference: From SD-SO23-360, Figure I-6	Revision # 17



Comments / Reference: From SD-SO23-650, Figure 1	Revision # 12



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008 A2.04	
Importance Rating	3.3	

Component Cooling Water System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PRMS alarm

Proposed Question: Common 8

Given the following conditions on Unit 2:

- All systems on both Units are aligned for normal MODE 1 conditions.
- Annunciator 57C50 - PROCESS/EFFLUENT/AREA RADIATION HI has alarmed from high radiation on 2RE-7819, CCW Non-Critical Loop Radiation Monitor.

Which ONE (1) of the following:

1.) Identifies a possible cause for the alarm?

2.) What action will be taken to mitigate the situation?

A. 1.) Waste Gas Compressor leakage into the Component Cooling Water System.
2.) Swap Waste Gas Compressors and isolate the CCW supply and return valves to the Waste Gas Compressors.

B. 1.) Radwaste Condensate Return Sample Cooler leakage into the Component Cooling Water System.
2.) Have Chemistry secure any sampling through the Radwaste Condensate Return Sample Cooler.

C. 1.) Control Element Drive Mechanism Cooler E404 leakage into the Component Cooling Water System.
2.) Shift Control Element Drive Mechanism Cooling Fans and isolate the CCW supply and return to E404.

D. 1.) Letdown Heat Exchanger leakage into the Component Cooling Water System.
2.) Secure Letdown and Charging and isolate the CCW supply and return to the Letdown Heat Exchanger.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a Waste Gas Compressor can leak into CCW and the isolation actions are correct, however, the compressors are normally aligned to Unit 3 Non-Critical Loop.
B. Incorrect. Plausible because the Radwaste Condensate Return Sample Cooler can leak into CCW and the isolation actions are correct, however, the Radwaste Condensate Return Sample Cooler is normally aligned to Unit 3 Non-Critical Loop.
C. Incorrect. Plausible because E404 is supplied by the Unit 2 Non-Critical Loop and the isolation actions are correct, however, the cooler is a CCW to low pressure air heat exchanger and could not leak into the CCW System.
D. Correct. The Letdown Heat Exchanger is a viable source based on pressure and activity potential and the actions are correct per SO23-13-7.

Technical Reference(s): SO23-13-7, Step 9; SD-SO23-400, Figures 3 & 4; SD-SO23-400, Page 10 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81027 / 81030 — EXPLAIN the interfaces between the CCW System and other plant systems. ANALYZE normal and abnormal operations of the CCW System.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 10
- 55.43 ____

Comments / Reference: From SO23-13-7, Step 9	Revision # 13

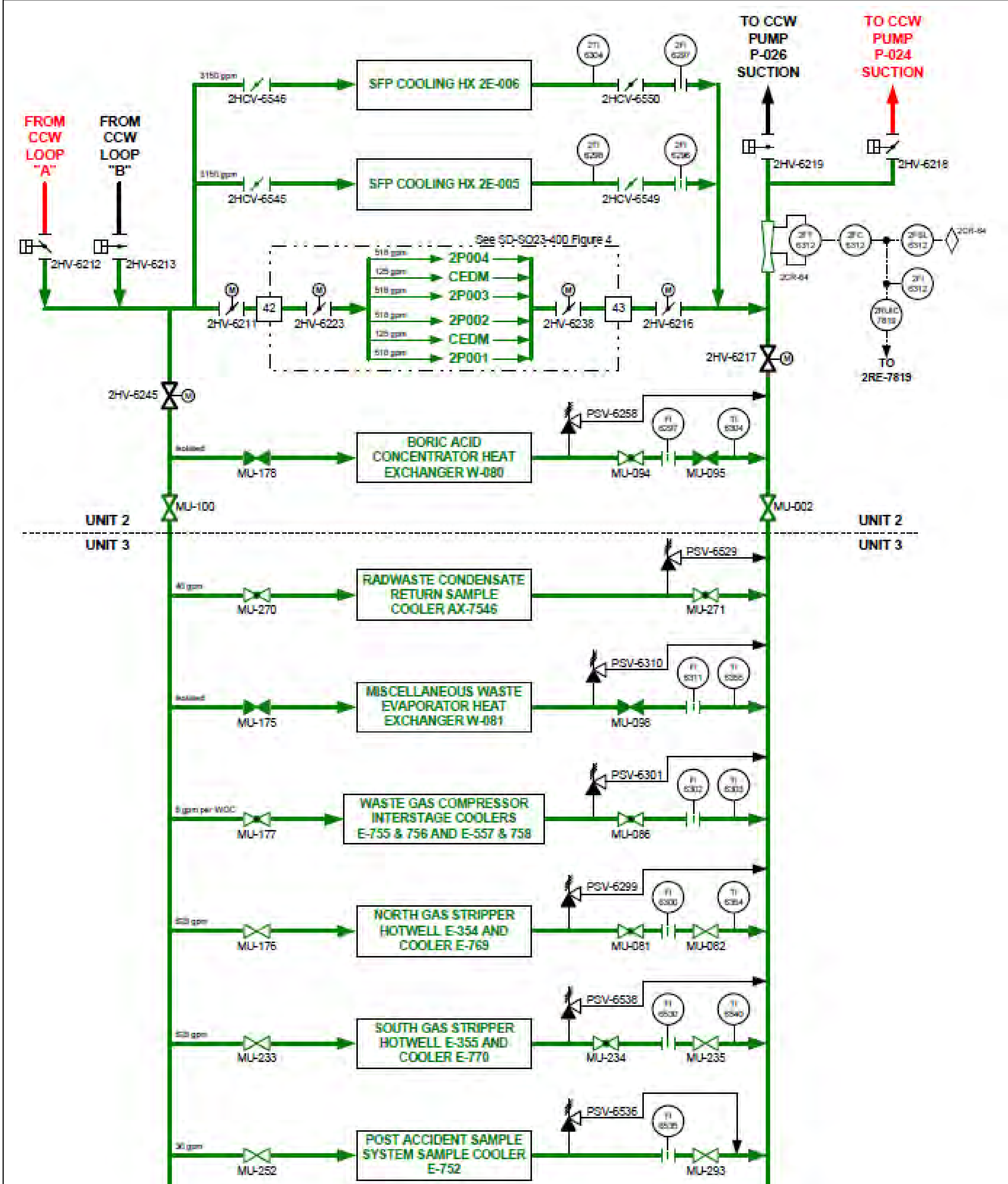
NUCLEAR ORGANIZATION ABNORMAL OPERATING INSTRUCTION SO23-13-7
UNITS 2 AND 3 REVISION 13 PAGE 16 OF 110

LOSS OF COMPONENT COOLING WATER (CCW)/SALTWATER COOLING (SWC)

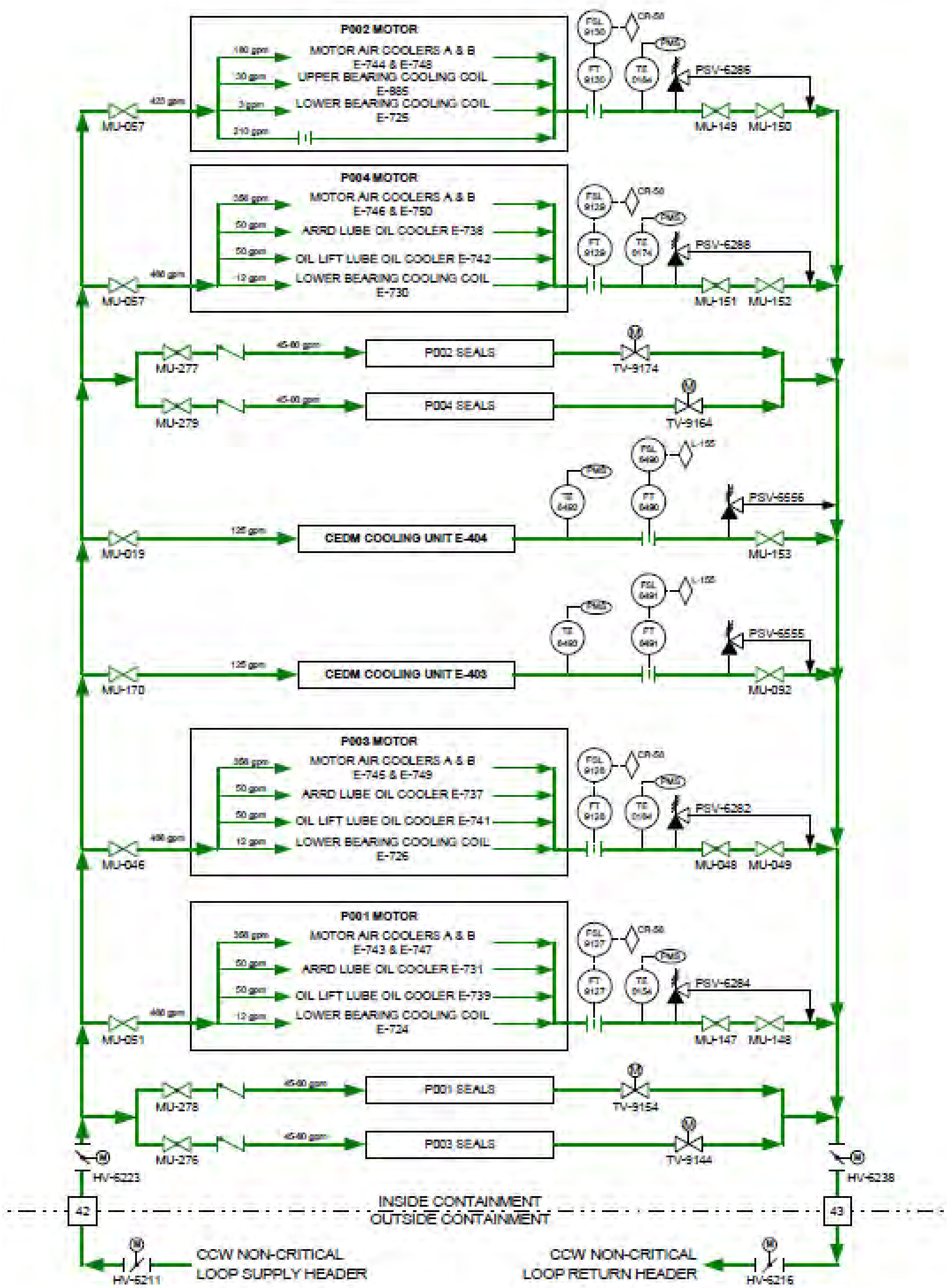
OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9 **CCW High Activity**	
☐ a. NOTIFY the 70' HP Control Point.	
b. CHECK system parameters:	
☐ • CCW return temperatures	
☐ • CCW flow rates	
☐ • Pipe radiation levels (with hand-held radiation detector)	
At possible leak locations:	
☐ • E-062, Letdown Heat Exchanger TI-6295 and FI-6294 (Loop A) TI-6389 and FI-6388 (Loop B)	
☐ • Shutdown Cooling Heat Exchangers TI-6332 and 2(3)FISL-6331 (ME-003) TI-6251 and 2(3)FISL-6250 (ME-004)	
☐ • Spent Fuel Pool Heat Exchangers TI-6248 and FI-6297 (ME-005) TI-6304 and FI-6296 (ME-006)	
☐ • AX-7546, Radwaste Condensate Return sample cooler	
☐ c. MONITOR RIC-7819, CCW Process Radiation Monitor, for leakage trend.	
☐ d. DETERMINE the source of the activity.	☐ d. EVALUATE the magnitude of leakage and activity per SO23-13-14.
☐ 1) ISOLATE the leak	
☐ 2) INITIATE necessary repairs.	
☐ e. GO TO Step 19.	

Comments / Reference: From SD-SO23-400, Figure 3	Revision # 18



Comments / Reference: From SD-SO23-400, Figure 4	Revision # 18



Comments / Reference: From SD-SO23-400, Page 10	Revision # 18

NUCLEAR ORGANIZATION SYSTEM DESCRIPTION SD-SO23-400
UNITS 2 AND 3 REVISION 18 PAGE 10 OF 70

2.0 DESCRIPTION (Continued)

2.1 **Main Flow Paths** (See Figure 1) (Continued)

.3.5 In summary, a single failure of a Surge Tank Normal Makeup Water Valve could result in overpressurizing both trains of CCW.

.4 A Process Radiation Monitor is installed to detect leakage of radioactive fluid into the Component Cooling Water System.

.4.1 The radiation monitor monitors the radiation level of the cooling water in the return header of the Non-Critical Loop.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008 G 2.4.46	
Importance Rating	4.2	

Component Cooling Water System: Emergency Procedures/Plan: Ability to verify that the alarms are consistent with plant conditions

Proposed Question: Common 9

Given the following conditions with both Units operating at 100% power:

- Annunciator 64A26 - CCW SURGE TANK TRAIN A LEVEL HI/LO is in alarm on both Units.
- The following Component Cooling Water (CCW) parameters are reported:
  - Unit 2 Train A Loop CCW Pump discharge pressure is 118 psig and CCW Surge Tank level is lowering.
  - Unit 3 Train A Loop CCW Pump discharge pressure is 113 psig and CCW Surge Tank level is rising.

Which ONE (1) of the following would cause these alarms and indications?

A. Unit 2 and Unit 3 Train A CCW loops are cross-connected via Post Accident Cleanup Unit E370.

B. Unit 2 and Unit 3 Train A CCW loops are cross-connected via Emergency Chiller E336.

C. Unit 2 CCW Surge Tank fill valve is closed with a tube leak in the Unit 3 Letdown Heat Exchanger.

D. Unit 3 CCW Surge Tank fill valve is closed with a tube leak in the Unit 2 Letdown Heat Exchanger.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it may be thought that the Spent Fuel Building CCW loads are common and may be swapped between Units similar to the Emergency Chiller, however, E370 is a Unit specific Train A load.

B. Correct. Emergency Chiller E336 can be supplied from either Unit 2 or Unit 3 Train A CCW. There are no check valves and inter-unit leakage is a real potential.

C. Incorrect. Plausible because the pressure difference across the Unit 3 Letdown Heat Exchanger would cause a leak into CCW and Unit 3 level would rise, however, the fill valve being closed does not by itself explain Unit 2 level lowering.

D. Incorrect. Plausible because with or without a leak in the Letdown Heat Exchanger, the pressures listed would drive flow to the Unit 3 side causing the level transients described, however no cross-tie path is provided to explain this.

Technical Reference(s) SO23-13-7, Step 11a
SO23-2-17.1, Attachment 1
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81028 / 81030
DESCRIBE the configuration and operational characteristics of CCW System components.
ANALYZE normal and abnormal operations of the CCW System.

Question Source: Bank # 135049
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7, 10
55.43 ______

Comments / Reference: From SO23-13-7, Step 11a	Revision # 13

LOSS OF COMPONENT COOLING WATER (CCW)/SALTWATER COOLING (SWC)

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11 **Cross Unit CCW Leakage**	
☐ a. VERIFY Emergency Chillers CCW Supply and Return Valves - NOT recently realigned.	☐ a. ENSURE Emergency Chillers CCW Supply and Return Valves are aligned correctly per SO23-2-17, Attachment for Transferring Emergency Chiller E-335 and/or E-336 CCW Supply.
☐ b. VERIFY Unit 3 supplying Unit 2 and 3 Noncritical Loops in Radwaste.	☐ b. ENSURE Unit 2 and 3 Noncritical Loops in Radwaste are aligned correctly for current plant conditions.
☐ 1) ENSURE CLOSED SA1203MU100, Radwaste Return Header Unit Crosstie.	
☐ 2) ENSURE CLOSED SA1203MU002, Radwaste Return Header Unit Crosstie.	
☐ c. REQUEST Maintenance Engineering to evaluate cause of cross-Unit leakage.	
☐ d. GO TO Step 19.	

Comments / Reference: From SO23-2-17.1, Attachment 1	Revision # 15

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2.0 PROCEDURE (Continued)

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	INITIALS PERF/ IND VER
2.1.7	**SDC HX ME-004**				
.1	2(3)HCV-6548	Shutdown Cooling Train A Heat Exchanger ME-004 CCW Supply Throttle Valve		LOCKED (A) $7^3/8$ ($7^5/8$) TURNS OPEN	______ ______ **CV**
.2	2(3)HV-6501	Shutdown Cooling Train A Heat Exchanger ME-004 CCW Return Isolation Valve	[1]	CLOSED/ OPEN	______ ______
.3	N/A	2(3)HV-6501 Auto/Manual Positioner [LS-2.7]		AUTO	______ ______
.4	S2(3)2417MR171	Shutdown Cooling Train A HX ME-004 CCW Return Iso HV-6501 Inst Air Supply Block Valve		OPEN	______ ______
2.1.8	**Post Accident Cleanup Unit ME-370**				
.1	S2(3)1203MU063	Fuel Handling Bldg Train A PACU ME-370 CCW Return Throttle Valve		LOCKED (A) $2^1/10$ (2) TURNS OPEN	______ ______ **CV**
.2	S2(3)1203MU015	Fuel Handling Bldg Train A PACU ME-370 CCW Supply Isolation Valve		LOCKED (A) OPEN	______ ______
.3	S2(3)1203MU087	Fuel Handling Bldg Train A PACU ME-370 CCW Return Isolation Valve		LOCKED (A) OPEN	______ ______

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	010 K3.02	
Importance Rating	4.0	

Pressurizer Pressure Control System: Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RPS

Proposed Question: Common 10

Given the following conditions at 100% power:

- HS-0100A, Pressurizer Pressure Channel Select Switch is selected to Channel X.
- PT-0100X, Pressurizer Pressure Control System pressure transmitter has failed low.

Assuming no operator actions, which ONE (1) of the following identifies the FIRST Reactor Protection System trip that will be actuated?

A. Low Departure from Nuclear Boiling Ratio.

B. High Pressurizer Pressure.

C. High Local Power Density.

D. Low Pressurizer Pressure.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because a low Pressurizer pressure is associated with a low DNBR, however, when this pressure transmitter fails low it turns all Heaters on and Pressurizer pressure will rise.
B. Correct. Failing this transmitter low energizes all Pressurizer Heaters, raising pressure until the high pressure trip setpoint is reached.
C. Incorrect. Plausible because low DNBR and high Local Power Density (LPD) trips are generated by Core Protection Calculators which have inputs including pressurizer pressure, however, pressure only affects DNBR trip setpoint, not LPD.
D. Incorrect. Plausible if thought that the pressure transmitter failing low would cause a low Pressurizer pressure trip.

Technical Reference(s) SD-SO23-360, Figure III-5; SO23-SO23-710, Figures 1 & 8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56622 — DESCRIBE the response of the Plant Protection System to failures and alarms, including possible causes, effects on the system or overall plant, and operator actions to mitigate the effects.

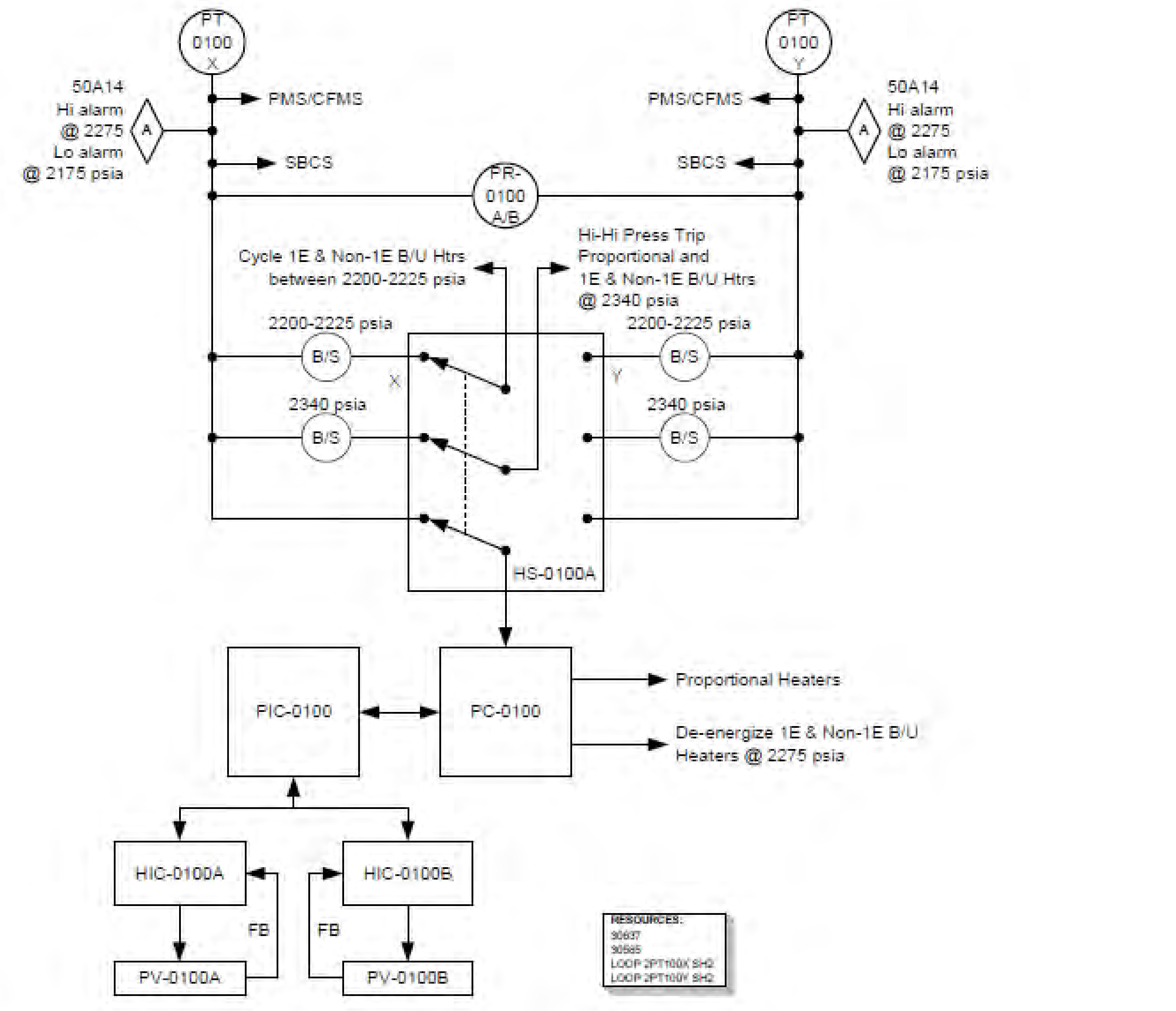
Question Source: Bank # 128108
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

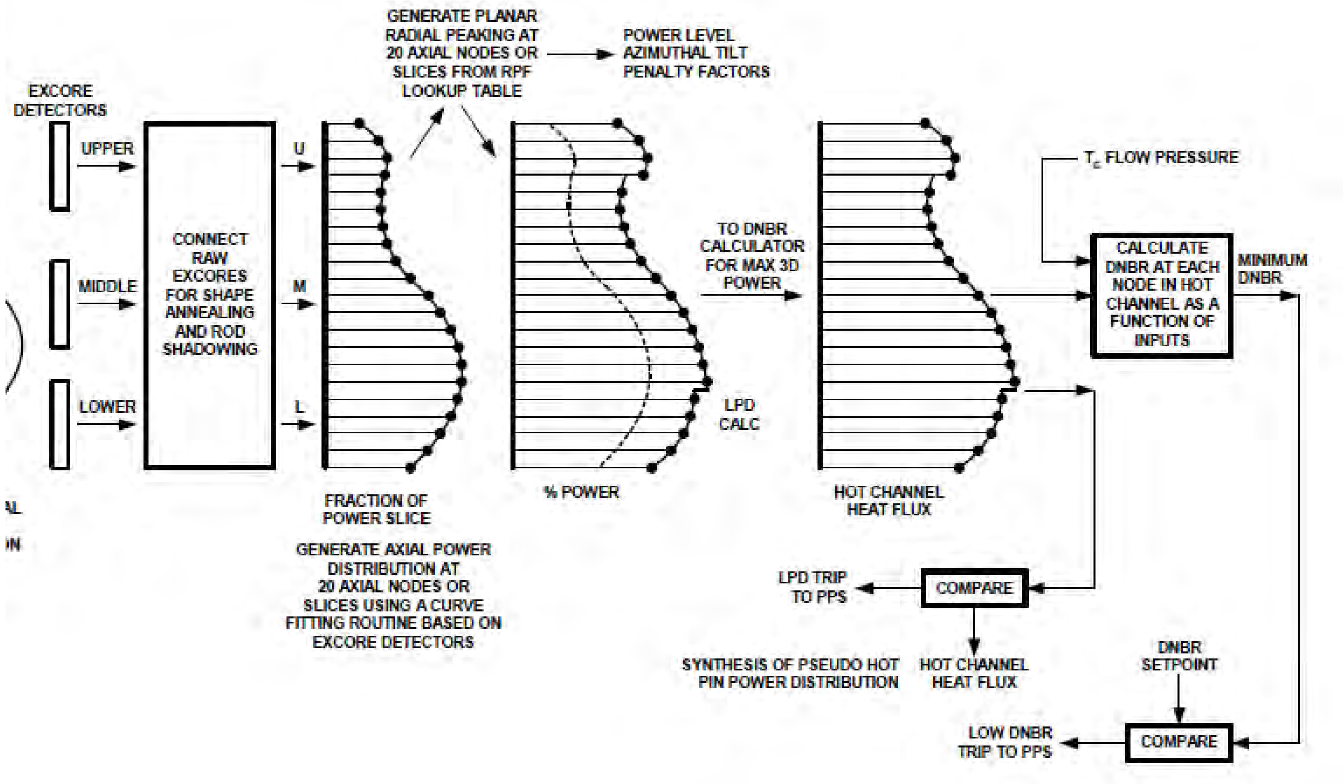
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ______

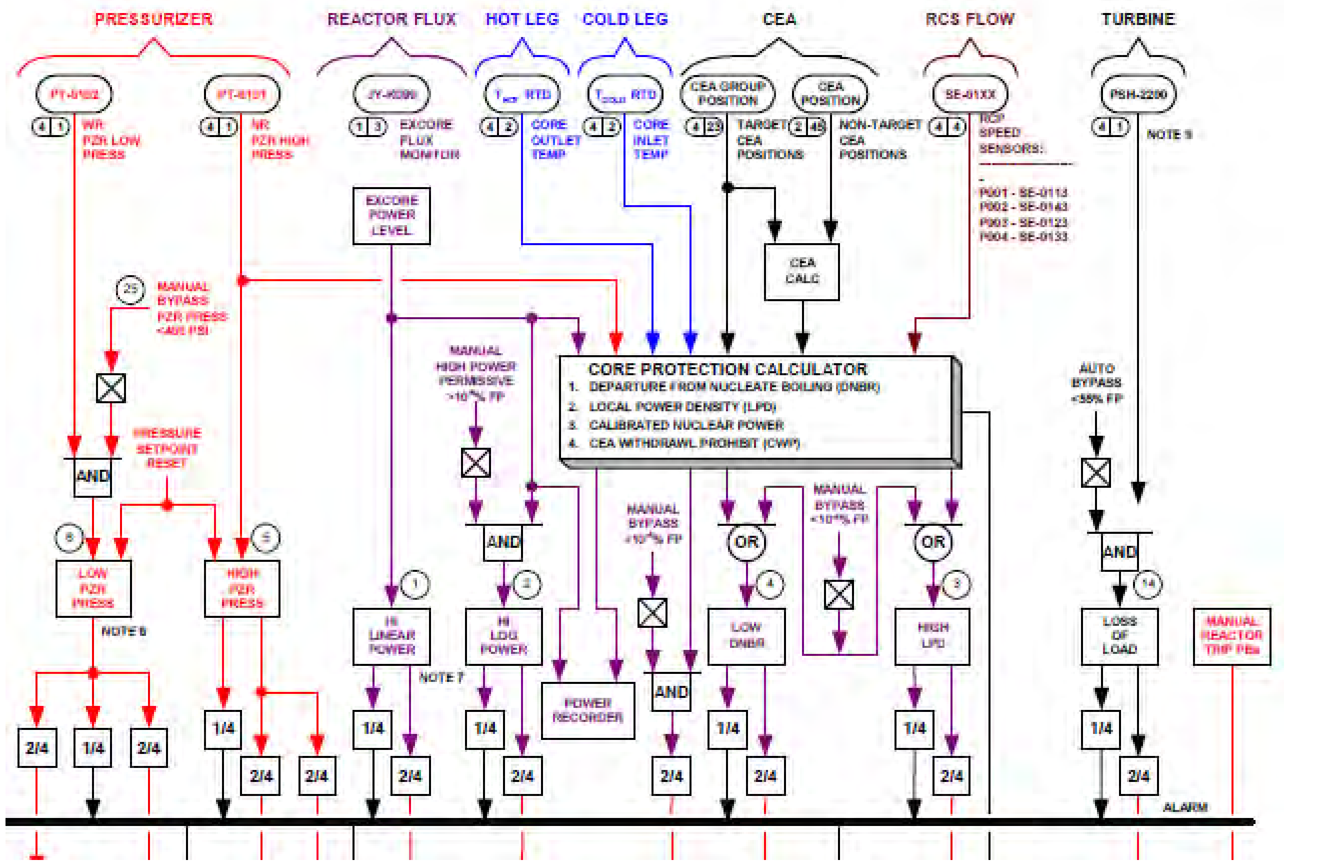
Comments / Reference: From SD-SO23-360, Figure III-5	Revision # 17



Comments / Reference: From SD-SO23-710, Figure 8	Revision # 7



Comments / Reference: From SD-SO23-710, Figure 1	Revision # 7



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K2.01	
	Importance Rating	3.3	

Reactor Protection System: Knowledge of bus power supplies to the following: RPS channels, components, and interconnections

Proposed Question: Common 11

Given the following conditions:

- Unit 2 is at 100% power.
- Pressurizer Level Control is selected to Channel X.
- Pressurizer Pressure Control is selected to Channel X.

If Vital 120 VAC Instrument Bus 2Y02 is deenergized, which ONE (1) of the following occurs?

A. All Pressurizer Proportional and Backup Heaters energize.

B. A Reactor trip will occur.

C. All three (3) Charging Pumps automatically start.

D. Control Element Assembly Calculator #1 fails.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because all Pressurizer Heaters will energize upon a loss of Vital 120 VAC Instrument Bus 2Y01 with Channel X in service.

B. Incorrect. Plausible because a Core Protection Calculator Auxiliary Trip will generate a Reactor trip, however, two channels are required. Only Channel B trips on the loss of Instrument Bus 2Y02.

C. Incorrect. Plausible because all Charging Pumps will start upon a loss of Vital 120 VAC Instrument Bus 2Y01 with Channel X in service.

D. Correct. Loss of Vital 120 VAC Instrument Bus 2Y02 will cause Control Element Assembly Calculator #1 failure.

Technical Reference(s) SO23-13-18, Attachment 2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56418 — DESCRIBE the operation of Pressurizer Level Control System components, instrumentation, controls and alarms including function, location, interlocks, capacity and power supplies where applicable.

Question Source: Bank # 127140
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam SONGS 2006

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

10 CFR Part 55 Content: 55.41 7, 10
55.43 ______

Comments / Reference: From SO23-13-18, Attachment 2	Revision # 8

2.0 PROCEDURE (Continued)

2.2 **EFFECTS AND ACTIONS ON LOSS OF VITAL BUS Y02.** (Continued)

AFFECTED EQUIPMENT	INDICATIONS AND ASSOCIATED ACTIONS
.7 Atmospheric Dump Valve/Controller: HV-8421/PIC-8421-2	● HV-8421, Atmospheric Dump Valve, FAILS CLOSED. HV-8421, Atmospheric Dump Valve, may be operated locally.
.8 Atmospheric Dump Valve/Controller: HV-8419/PIC-8419-1	● HV-8419, Atmospheric Dump Valve, will lose its Pressure input but can be operated from the Controller in MANUAL using PIC-8419-1.
.9 EFAS Trip Paths 2 & 4 Valves: HV-4712 HV-4705 HV-4715 HV-4731 HV-4716	● Valves Open. ● The affected Unit is in a *4 hour* Action Statement (Tech. Spec. LCO 3.7.5) since these valves will not close on a MSIS signal.
.10 Status indicating lamps for both RPS (RTCB) Status Panels	● Extinguished (Control Room and Hallway indication).
.11 RX Trip Paths 1 and 2 Actuated	☐ INITIATE local verification that RTCBs 1, 2, 5, and 6 are OPEN. ☐ INITIATE local verification that RTCBs 3, 4, 7 and 8 are CLOSED.
.12 Channel B CPC	● Tripped.
.13 CEAC 1	● Failed.
.14 PPS HI Log Power	● Tripped.

Comments / Reference: From SO23-13-18, Attachment 2	Revision # 8

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LOSS OF VITAL BUS Y02

CONTINUOUS USE

1.0 PREREQUISITES

None.

2.0 PROCEDURE

2.1 Review Tech. Spec. impacted LCO 3.4.9.b, 3.8.1, 3.8.2, 3.8.3, 3.8.7, 3.8.9 and 3.7.5.

2.2 **EFFECTS AND ACTIONS ON LOSS OF VITAL BUS Y02.**

2.2.1 Perform the following:

AFFECTED EQUIPMENT	INDICATIONS AND ASSOCIATED ACTIONS
.1 PPS B status lights extinguished	☐ VERIFY protection system bistables NOT TRIPPED on PPS Channels A and C ROMs.
.2 Channels 1-4 Red ESFAS Function lights along the bottom of the ROM extinguished.	☐ VERIFY all ESFAS function lights EXTINGUISHED on PPS Channels A, B, C, & D ROMs. [1]
.3 Channel B Lumigraphs on CR56 extinguished	☐ VERIFY Safety Channel indications providing input to PPS Channels A, C, and D do not indicate that a Plant Protection Trip Setpoint has been exceeded.
.4 Charging Pumps P-190, P-191, and P-192.	☐ Operate Charging Pumps as necessary to control PZR level.
.5 PZR Pressure and Level Control	☐ ENSURE PZR Level Channel X is SELECTED. ☐ If LIC-0110 is selected to setpoint LS2, then transfer Pressurizer level setpoint to LS1 per SO23-3-1.10, Attachment for Transferring Pressurizer Level and Pressure Controls. ☐ If an ACTUAL Pressurizer LO-LO level exists, then ENSURE all heaters DE-ENERGIZED. (AR 020900184)
.6 Vital Bus Inverter Y002 de-energized	☐ ENSURE SO23-6-17, Attachment for Re-energizing Vital Bus Y02 from the Alternate Source, in progress. (Tech. Spec. LCO 3.8.7 and LCO 3.8.9)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013 K6.01	
Importance Rating	2.7	

Engineered Safety Features Actuation System: Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors

Proposed Question: Common 12

Given the following conditions:

- Unit 2 is in MODE 1.
- PT-351A, Containment Narrow Range Pressure Transmitter has failed HIGH.

Which ONE (1) of the following describes the effects on the Plant Protective System if a second Containment Narrow Range Pressure Transmitter failed HIGH?

A. CIAS only actuates.

B. CIAS, SIAS and CSAS actuate.

C. CIAS, SIAS and CCAS actuate.

D. CIAS, SIAS, CSAS and CCAS actuate.

Proposed Answer: C

Explanation:

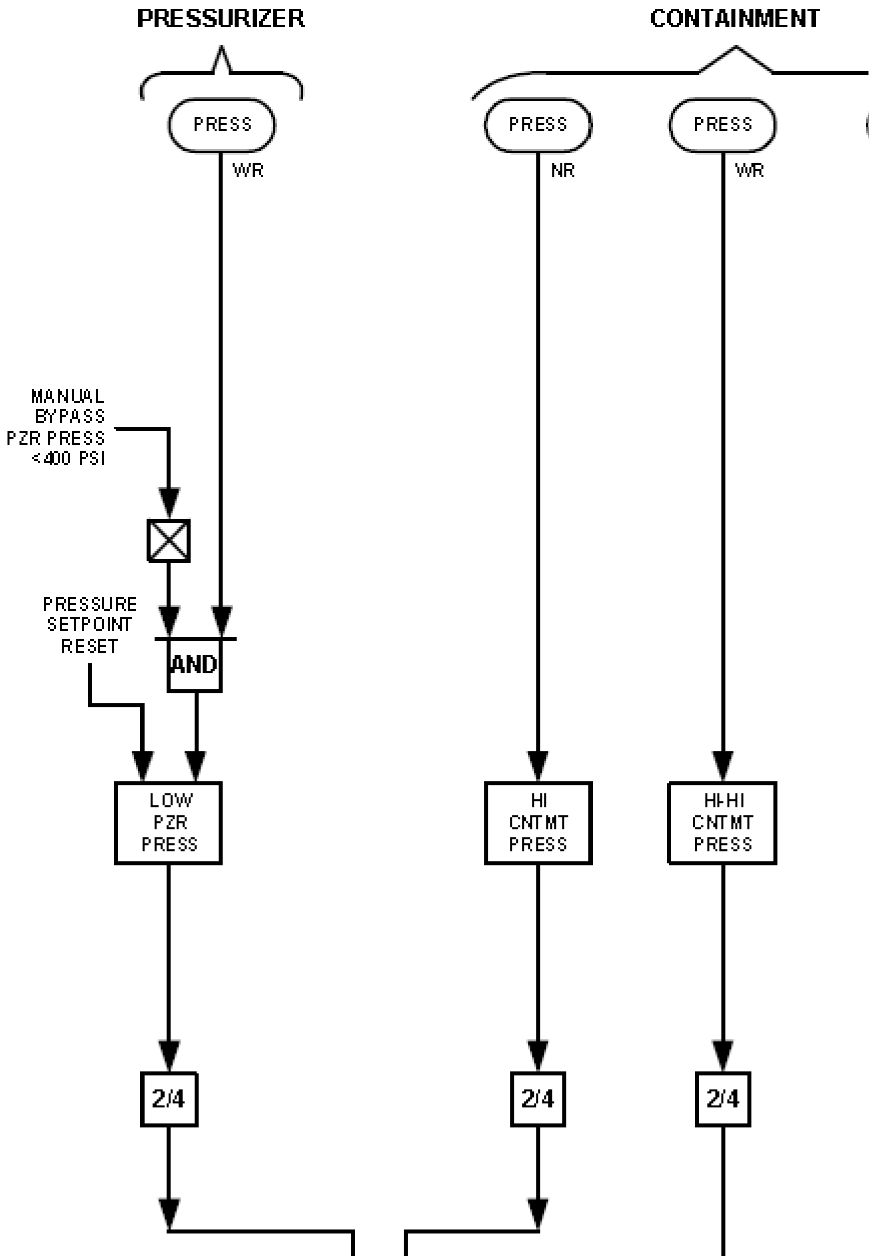
A. Incorrect. Plausible because it could be thought that only low Pressurizer Pressure actuated SIAS and that CCAS and CSAS actuation require SIAS actuation.
B. Incorrect. Plausible because it could be thought that with high Containment pressure and SIAS that CSAS would actuate and that the CCAS required a 2/4 signal from the wide range Containment pressure transmitters.
C. Correct. SIAS and CIAS will actuate on 2/4 narrow range Containment pressures at the high setpoint and when SIAS actuates then CCAS will actuate.
D. Incorrect. Plausible because it could be thought that with a failed high Containment pressure and SIAS, that CSAS and CCAS also would actuate however CSAS requires 2/4 from the wide range Containment pressure transmitters.

Technical Reference(s) SD-SO23-720, Figure 1 Attached w/ Revision # See Comments / Reference

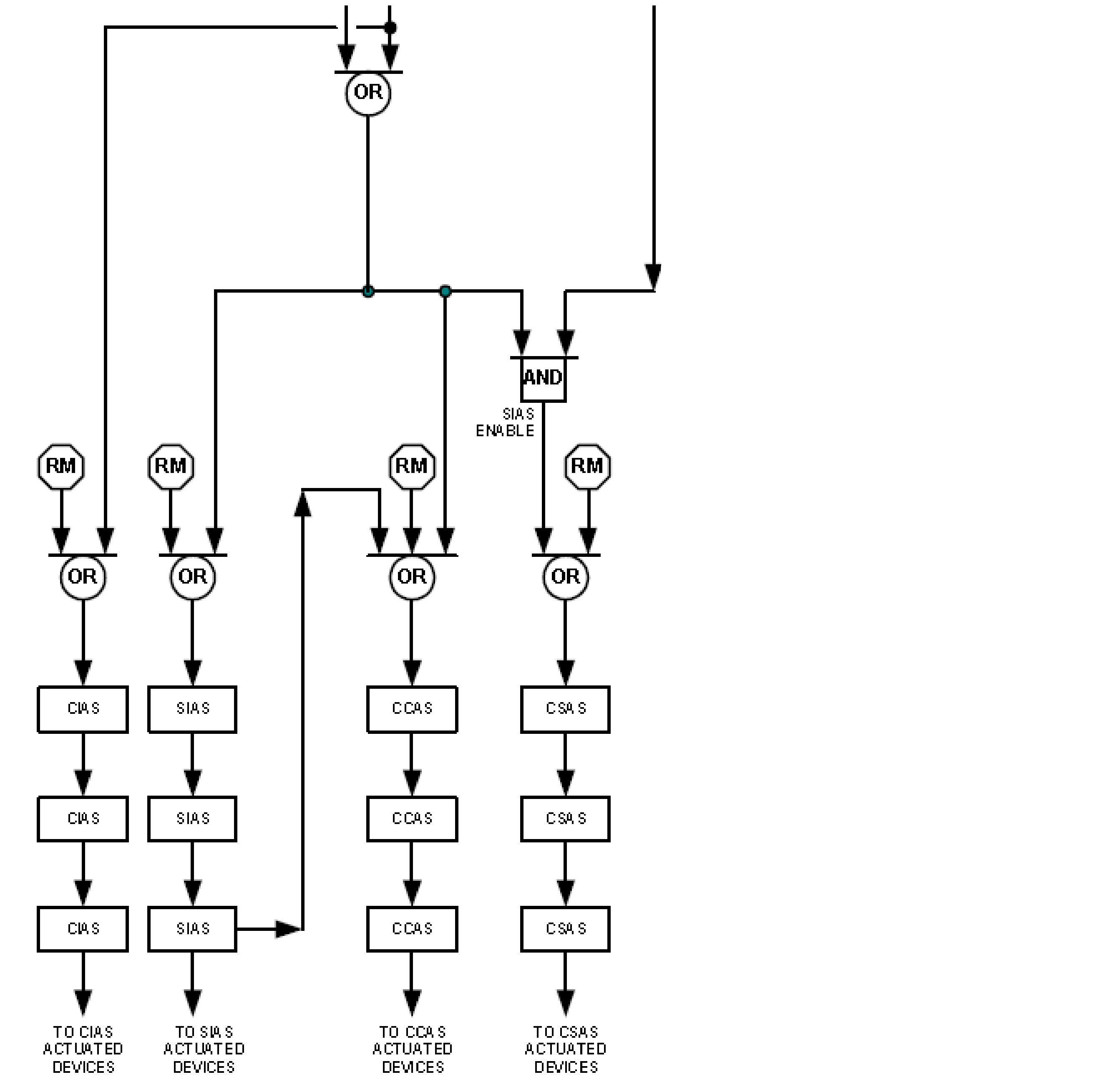
Proposed references to be provided during examination: None

Learning Objective: 56627	DESCRIBE the operation of the Plant Protection System components and instrumentation, including function, location, design basis, interlocks, setpoints, special features and power supplies, where applicable.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge ______
	Comprehension or Analysis X
10 CFR Part 55 Content:	55.41 7
	55.43 ______

Comments / Reference: From SD-SO23-720, Figure 1	Revision # 8



Comments / Reference: From SD-SO23-720, Figure 1	Revision # 8



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A1.04	
	Importance Rating	3.2	

Containment Cooling System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Cooling water flow

Proposed Question: Common 13

Given the following conditions:

- Unit 2 is in MODE 1.
- Train A Component Cooling Water System is in service with Train B Component Cooling Water System in standby.

Which ONE (1) of the following describes the normal flowpath alignment of Component Cooling Water (CCW) to the Containment Emergency Cooling Units (ECU)?

A. Train A ECU CCW Supply and Return Valves are open.
Train B ECU CCW Supply and Return Valves are open.

B. Train A ECU CCW Supply and Return Valves are open.
Train B ECU CCW Supply and Return Valves are closed.

C. Train A ECU CCW Supply Valves are closed and Return Valves are open.
Train B ECU CCW Supply Valves are open and Return Valves are closed.

D. Train A ECU CCW Supply Valves are open and Return Valves are closed.
Train B ECU CCW Supply Valves are closed and Return Valves are open.

Proposed Answer: A

Explanation:

A. Correct. This is the correct lineup for ECU cooling water flow when in MODE 1 with one Train of Component Cooling Water in operation. Supply and Return Valves on both Trains are open with CCW flow through the two (2) Train A ECUs.

B. Incorrect. Plausible because Train B CCW is in standby, however, at a minimum the Train B Inlet Valves would be open per L&S 5.7.

C. Incorrect. Plausible because the Train B alignment would be considered correct, however, with the Unit in MODE 1 the Train A ECU Supply Valves would also be open.

D. Incorrect. Plausible because the Train A Supply Valves are open and that is required, however, Train A Supply Valves must also be open.

Technical Reference(s) SO23-2-17, Section 6.1 Attached w/ Revision # See Comments / Reference
SO23-2-17, L&S 2.18 and 5.7

Proposed references to be provided during examination: None

Learning Objective: 81028 DESCRIBE the configuration and operational characteristics of CCW System components.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments / Reference: From SO23-2-17, Section 6.1	Revision # 27

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6.0 PROCEDURE

6.1 CCW ECU Return/SDCHX Outlet Valves Preferred Alignment

INFORMATION USE

6.1.1 Change alignments in a sequence that maintains CCW flows within limits (between 5,000 gpm and 16,000 gpm) and ends with the **preferred alignment** as listed below: (AR 040101606, 060300045)

.1 When Shutdown Cooling is in service, then changes in CCW flow have the potential to change reactivity.

SDC IN SERVICE	NUMBER OF TRAINS IN SERVICE	STATUS OF CCW TRAINS	2 ECU RETURN VALVES/ TRAIN	1 SDCHX OUTLET VALVE/ TRAIN	NCL SUPPLY VALVE	REFERENCE
NO	1	IN-SERVICE TRAIN	OPEN	CLOSED	OPEN	[1] LS-2.15
		STANDBY TRAIN	OPEN	CLOSED	CLOSED	
NO	2	IN-SERVICE TRAIN	OPEN	CLOSED	OPEN	[1] [2] LS-2.15
		IN-SERVICE TRAIN	OPEN	OPEN	CLOSED	
YES	1	IN-SERVICE TRAIN	CLOSED	OPEN	OPEN	LS-2.15
		STANDBY TRAIN	OPEN	OPEN	CLOSED	
YES	2	IN-SERVICE TRAIN	OPEN	OPEN	CLOSED	[2] LS-2.15
		IN-SERVICE TRAIN	CLOSED	OPEN	OPEN	

[1] **Alternate alignments** may be used for unavailable equipment. For example, 1 SDCHX Outlet Valve may be used in place of 2 ECU Return Valves on the Train supplying the NCL. Flow thru an ECU associated with an in service CCW Critical Loop may be terminated/initiated to support maintenance activities (MOVATS, etc.) provided CCW Flow Limits (between 5,000 and 16,000 gpm) are maintained for the operating requirements at the time. (LS-2.16, LS-2.18)

[2] If 2 trains are in service, then ensure **alternate alignment** provides proper flow balancing between the loops.

Comments / Reference: From SO23-2-17, L&S 2.18	Revision # 27

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ATTACHMENT 9

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CCW SYSTEM LIMITATIONS AND SPECIFICS (Continued)

2.0 **SYSTEM GUIDELINES** (Continued)

2.16 CCW Train preferred alignment is designed to provide the following:

- A balanced system flow and pressure to minimize cross-train leakage when both trains are in service
- Ensures a 5,000 gpm mini-flow protection flowpath for the pump on the standby train should it inadvertently start
- Maintains CCW flow through the SDCHX < 7,600 gpm
- Maintains CCW Pump flow < 16,000 gpm (pump runout)

Alternate alignments that ensure these conditions are satisfied, may be used. For example, substituting 1 SDCHX Outlet Valve in place of 2 ECU Return Valves on the Train supplying the NCL ensures that flow thru the SDCHX will not exceed 7,600 gpm.

2.17 Parallel pump operation of CCW Pumps (i.e., two Pumps in service on the same Train) is NOT allowed for IST's. The numerical valves obtained during an IST will be significantly altered, and invalid if two pumps are running in parallel. Running parallel pumps for short periods of time such as during pump transfers is allowed. (AR 050901401, AR 071101081)

2.18 A CCW Critical Loop can support 2 of the following 3 loads: a pair of ECUs, a SDC HX, or the NCL (includes SFP Cooling). Placing all three loads on the Critical Loop will cause the maximum flowrate to be exceeded (potential runout on a CCW Pump). (AR 060300045)

2.18.1 HV-6500 and HV-6501, SDCHX CCW Outlet Valves, fail open on loss of instrument air. When in the Preferred Alignment with SDC secured, then a loss of instrument air may cause CCW Pump runout to occur on the Train supplying the Non-Critical Loop. If this occurs, then SO23-13-5, Loss of Instrument Air, will direct closing one ECU CCW Return Valve on the Train supplying the Non-Critical Loop. (AR 070600872)

Comments / Reference: From SO23-2-17, L&S 5.7	Revision # 27

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CCW SYSTEM LIMITATIONS AND SPECIFICS (Continued)

5.0 **VALVE GUIDELINES** (Continued)

5.7 Containment ECU CCW Inlet Valves (HV-6366, HV-6368, HV-6370, HV-6372) must remain open in Modes 1, 2, 3, and 4 to minimize the differential pressure across the corresponding ECU Outlet Valve (HV-6367, HV-6369, HV-6371, HV-6373). Minimizing the differential pressure across the outlet valve ensures that it can be opened when required. If an ECU supply valve is closed (for surveillance testing, SDC flow balancing, etc.), then the associated ECU is inoperable. (AR 971201623, DBD 400, Sect. 4.4, Tech. Spec. 3.6.6.2 and Ref. 2.3.1.14)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026 A4.01	
Importance Rating	4.5	

Containment Spray System: Ability to manually operate and/or monitor in the control room: CSS controls

Proposed Question: Common 14

Given the following conditions:

- Unit 3 is operating at 100% power in MODE 1.
- A spurious Containment Spray Actuation Signal has just initiated.

Which ONE (1) of the following occurs as a result of the Containment Spray Actuation Signal?

The Containment Spray Pumps...

A. receive an Auto Start signal; SI Pumps and Containment Spray Pumps Mini-Flow Valves open.

B. do NOT receive an Auto Start signal; RWST Outlet Valves open.

C. receive an Auto Start signal; Containment Spray Header Isolation Valves open.

D. do NOT receive an Auto Start signal; Containment Spray Header Isolation Valves open.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that CSAS started the Containment Spray Pumps. The Safety Injection and Containment Spray Mini-Flow Valves open on an SIAS.

B. Incorrect. Plausible because the Containment Spray Pumps will not start, however, it is the Containment Spray Header Isolation Valves that open not the RWST Outlet Valves.

C. Incorrect. Plausible because the Containment Spray Header Isolation Valves will open, however, and SIAS is required to start the Containment Spray Pumps.

D. Correct. Without any SIAS present the Containment Spray Pumps will not start. The Containment Spray Header Isolation Valves will open with a CSAS signal.

Technical Reference(s)

SD-SO23-740, Section 2.3.2
SD-SO23-720, Section 2.1.2
SD-SO23-720, Appendix D
SD-SO23-720, Figure 2C

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 79745 / 79749 DESCRIBE the instrumentation used to monitor the operation of the CSS and SIS. Include the name, function, sensing points, normal values for the parameters being measured, and location of each instrument.
DESCRIBE the integrated operation of the CSS, and SIS.

Question Source:	Bank #	75141	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments / Reference: From SD-SO23-740, Section 2.3.2	Revision # 17

2.3.2 **Containment Spray System**

When a SIAS occurs the Containment Spray Pumps receive a start signal, with a ten (10) second time delay, and run on mini flow until needed. When Containment Pressure gets high enough, and a SIAS is present, a Containment Spray Actuation Signal (CSAS) is activated. Upon receipt of a CSAS the Containment Spray Header Isolation valves open to provide flow to the containment spray nozzles. The CSAS also opens cooling water valves for the Heat Exchangers.

Upon receipt of a RAS, the suction of the spray pumps shifts to the Emergency Sump, RAS isolates minimum HPSI flows if Emergency Sump HI-HI Level is coincident with RAS.

2.3.2 **Containment Spray System** (Continued)

The pumps and major valves can be controlled remotely from the Control Room. The Spray Pumps can also be controlled for their ESF switchgear rooms, Fire Isolation Switches determine controlling location. The Spray Pumps have SIAS override feature. System pressure, flow, temperature and level are provided in the Control Room for monitoring operation.

Comments / Reference: From SD-SO23-720, Section 2.1.2	Revision # 8

2.1.2 **General Control Scheme - NSSS ESFAS** (Continued)

.4 **CONTAINMENT SPRAY ACTUATION SIGNAL (CSAS)** (See Figures 1 & 2C)

PURPOSE:	To activate the Containment Spray System to remove heat and iodine from the Containment atmosphere in the event of a LOCA or MSLB.
INPUTS & SETPOINTS:	High-High Containment Pressure @14.0 psig, AND A Safety Injection Actuation Signal.
INITIATING DEVICES:	Containment Pressure Transmitters 2(3)PT-0352-1, -2, -3, -4.
LOGIC:	2/4 coincidence

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Comments / Reference: From SD-SO23-720, Section 2.1.2	Revision # 8

2.1.2 **General Control Scheme - NSSS ESFAS** (Continued)

.4 **CONTAINMENT SPRAY ACTUATION SIGNAL (CSAS)** (See Figures 1 & 2C) (Continued)

.4.4.2 Manual initiation of CSAS does not result in spray flow unless a SIAS is present. (This is a selective 2/4 logic.)

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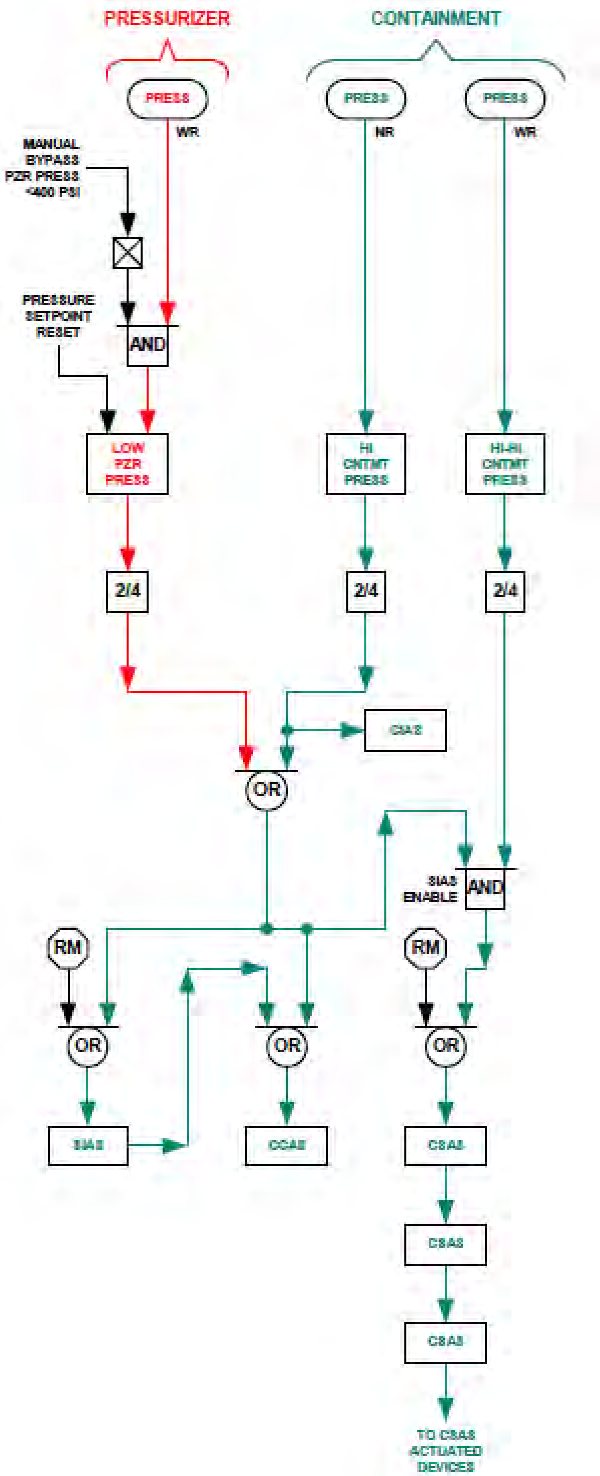
Comments / Reference: From SD-SO23-720, Appendix D	Revision # 8

APPENDIX D (per SO23-3-2.22, Att. 9)			
CONTAINMENT SPRAY ACTUATED (CSAS) EQUIPMENT LIST			
EQUIPMENT NUMBER	ESF DESCRIPTION	CSAS TRAIN	FUNCTION
2(3)HV-6501	Component Cooling Water from Shutdown Heat Exchanger, E-004	A	OPEN
2(3)HV-9367	Shutdown Heat Exchanger to Containment Spray Header No. 1	A	OPEN
2(3)P-012	Containment Spray Pump (a)	A	START
2(3)HV-6500	Component Cooling Water from Shutdown Heat Exchanger, E-003	B	OPEN
2(3)HV-9368	Shutdown Heat Exchanger to Containment Spray Header No. 2	B	OPEN
2(3)P-013	Containment Spray Pump (a)	B	START
a. Pumps START on a SIAS			

Comments / Reference: From SD-SO23-720, Figure 2C	Revision # 8



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Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 A2.01	
	Importance Rating	2.7	

Containment Spray System: Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reflux boiling pressure spike when first going on recirculation

Proposed Question: Common 15

Given the following conditions:

- Unit 3 has experienced a Loss of Coolant Accident.
- All safety systems actuated as designed.
- The Recirculation Actuation Signal (RAS) setpoint has been reached.
- The crew is preparing to perform SO23-12-11, EOI Supporting Attachments, Attachment 14, RAS Operation.

Which ONE (1) of the following describes the affect that RAS operation has on plant parameters and the actions taken to ensure Critical Safety Functions are maintained?

A. The suction pressure to the High Pressure Safety Injection Pumps will be greater and result in flow approaching runout.
Throttle High Pressure Safety Injection Cold Leg Injection Valves to prevent runout.

B. The suction pressure to the Containment Spray Pumps will be greater and result in flow approaching runout.
Throttle the Containment Spray Flow Control Valves to prevent runout.

C. The water being used is hotter and could result in rising Containment pressure.
Verify Containment pressure is less than 14 psig or ensure proper Containment Spray actuation.

D. The water being used is hotter and could result in rising core temperatures.
Raise Safety Injection flow rate by restarting Low Pressure Safety Injection Pumps as required to restore subcooling.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because suction pressure from the Containment Sump will be higher and result in higher flow. The action to throttle flow in this situation would be inappropriate.
B. Incorrect. Plausible because suction pressure from the Containment Sump will be higher and result in higher flow. The action to throttle flow in this situation would be inappropriate.
C. Correct. The higher temperature of the SI cooling flow will result in RCS temperatures and Containment pressure rising. Attachment 14 attempts to reduce this affect by maximizing Containment Cooling but Floating Step 12 instructs the operator to verify Containment pressure less than 14 psig or ensure CSAS actuation.
D. Incorrect. Plausible because core temperatures will rise initially but after RAS the actions to restart LPSI would be inappropriate and jeopardize the available NPSH to the HPSI and Containment Spray Pumps.

Technical Reference(s) SO23-12-11, Attachment 14 Attached w/ Revision # See Comments / Reference
SO23-12-11, Floating Step 12d

Proposed references to be provided during examination: None

Learning Objective: 79744 / 79749
DESCRIBE the operation of the Pumps, Tanks, and remotely operated valves of the CIS, CSS, and SIS. Include the controls, function, location, and specific features such as type, capacity, and power supplies where applicable.
DESCRIBE the integrated operation of the CSS, and SIS.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 10
55.43 ______

Comments / Reference: From SO23-12-11, Attachment 14	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 6
ATTACHMENT 14

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EOI SUPPORTING ATTACHMENTS

RAS OPERATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 VERIFY RAS Conditions Established:	
a. VERIFY RWST level - less than 19%.	a. CONTINUE FS-20, MONITOR RWST Level.
2 VERIFY RAS Actuation:	
a. VERIFY Containment Emergency Sump level – greater than 18 feet 4 inches.	a. 1) MONITOR Containment Emergency Sump level. 2) NOTIFY Shift Manager/Operations Leader. 3) MONITOR need for borated water flow from RWST.
b. ENSURE Containment Emergency Sump Outlet valves – open: Train A: HV-9303, HV-9305 Train B: HV-9302, HV-9304	
c. ENSURE LPSI Pumps – stopped.	
d. ENSURE SI Pumps and Containment Spray Pump Miniflow Isolation valves – closed: Train A: HV-9306, HV-9307 Train B: HV-9347, HV-9348.	

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EOI SUPPORTING ATTACHMENTS

RAS OPERATION

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

3 **MAXIMIZE Containment Cooling**

a. ENSURE CCAS – actuated.

b. ENSURE available Containment Emergency Cooling Units – operating:

Train A	Train B
E-399	E-400
E-401	E-402

c. ENSURE CCW valves to operating Emergency Cooling Units – open:

Train A	Train B
HV-6370	HV-6368
HV-6371	HV-6369
HV-6366	HV-6372
HV-6367	HV-6373

d. ENSURE available Containment Dome Air Circulating Fans – operating:

Train A	Train B
A-071	A-072
A-074	A-073

Comments / Reference: From SO23-12-11, Floating Step 12d	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 6
ATTACHMENT 2

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

FS-12 MONITOR Containment Pressure

Applicability: ❑ 12-3, ❑ 12-5, ❑ 12-8, ❑ 12-9

NOTE

ESF actuation provides for specific valve closure even though AC power may not be available for complete ESF actuation.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. VERIFY Containment pressure – less than 3.4 PSIG	a. ENSURE the following – actuated: SIAS CCAS CRIS CIAS
b. VERIFY Containment Area Radiation Monitors – NOT alarming or trending to alarm.	b. ENSURE available Containment Normal HVAC – operating
c. VERIFY Containment High Range Area Radiation Monitors reading – less than 40R/HR.	c. 1) ENSURE SIAS – actuated. 2) REQUEST Shift Manager/Operations Leader to evaluate: a) CIAS actuation b) CSAS actuation for iodine removal.
d. VERIFY Containment pressure – less than 14 PSIG.	d. 1) ENSURE CSAS – actuated. 2) CLOSE CCW to/from Letdown Heat Exchanger valves: Train A: HV-6293B/A Train B: HV-6522B/A

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	039 A4.01	
Importance Rating	2.9	

Main and Reheat Steam System: Ability to manually operate and / or monitor in the control room: Main steam supply valves

Proposed Question: Common 16

Given the following conditions:

- Unit is in MODE 3 following a Reactor trip from 100% power.
- Containment pressure is 1.3 psig.
- Steam Generator E088 pressure is 720 psia.
- Steam Generator E089 pressure is 770 psia.
- RCS Tcold is 535°F.
- No operator actions have been taken post-trip.

Which ONE (1) of the following is the correct position for the listed valves?

1. HV-8204, Steam Generator E089 Main Steam Isolation Valve
2. HV-8205, Steam Generator E088 Main Steam Isolation Valve
3. HV-8421, Steam Generator E089 Atmospheric Dump Valve
4. HV-8423, Steam Bypass Control System Valve

A. 1. Closed
2. Closed
3. Closed
4. Closed

B. 1. Closed
2. Closed
3. Closed
4. Open

C. 1. Open
2. Closed
3. Closed
4. Open

D. 1. Closed
2. Open
3. Open
4. Open

Proposed Answer: A

Explanation:

A. Correct. This is the correct configuration given current SG pressure.
B. Incorrect. Plausible because valves would be aligned as such if an MSIS had not occurred.
C. Incorrect. Plausible because given system pressure, it could be thought that the MSIV on E089 is still open along with the SBCS Valve.
D. Incorrect. Plausible because the SBCS Valve should be open and it could be thought that the Atmospheric Dump Valve could also be open given system temperature. Position of HV-8205 is incorrect.

Technical Reference(s): SD-SO23-720, Page 21 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 102427 / 102465
IDENTIFY Main Steam System flowpaths, components, and locations including being able to draw and label system diagrams.
DESCRIBE the configuration and operational characteristics of Main Steam System components.

Question Source: Bank # 128014
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: From SD-SO23-720, Page 21	Revision # 8

NUCLEAR ORGANIZATION
UNITS 2 AND 3

SYSTEM DESCRIPTION SD-SO23-720
REVISION 8 PAGE 21 OF 101

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

2.1.2 **General Control Scheme - NSSS ESFAS** (Continued)

.5 **MAIN STEAM ISOLATION SIGNAL (MSIS)** (See Figure 1 & 2D) (Continued)

.5.2 When Steam Generator Pressure indicates safety action is required the MSIS automatically CLOSES all valves to both Steam Generators:

.5.2.1 Main Steam Isolation Valves (MSIVs),

.5.2.2 Main Feedwater Isolation Valves (FWIVs),

.5.2.3 Atmospheric Dump Valves (ADVs),

.5.2.4 Auxiliary Feedwater (AFW) Isolation Valves,

.5.2.5 S/G Blowdown Valves, and

.5.2.6 S/G Sample Valves.

.5.3 When an Emergency Feedwater Actuation Signal (EFAS) is generated from the intact Steam Generator, output contacts from the EFAS actuation relays associated with the EFAS logic for the intact Steam Generator are used to block, at the equipment level, the signal from the MSIS actuation relay contacts.

.5.3.1 Additional EFAS contacts are then used to open appropriate valves and initiate Auxiliary Feedwater to the intact Steam Generator.

.5.3.2 The operator may manually initiate the MSIS if plant conditions require or for testing.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059 G 2.1.27	
Importance Rating	3.9	

Main Feedwater System: Conduct of Operations: Knowledge of system purpose and/or function

Proposed Question: Common 17

Given the following conditions:

- Unit 2 has tripped.
- Reactor Trip Override (RTO) actuated.

With NO operator action, which ONE (1) of the following identifies how the Main Feedwater Control System responds?

A. Each Main Feedwater Regulating Valve positions to 5% open.

B. Main Feedwater Pump speed lowers to 3600 rpm for 10 seconds.

C. All Feedwater Regulating Bypass Valves close.

D. Each Feedwater Regulating Bypass Valve positions to 25% open.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that Main Feedwater Regulating Valves did not fully close on an RTO. The 5% value refers to total Feedwater flow supply to the Steam Generators post-trip.
B. Correct. This is the response of the Main Feedwater Pump following a Reactor Trip Override.
C. Incorrect. Plausible because the Feedwater Regulating Valves close, however, the Feedwater Regulating Bypass Valves position to 50% open.
D. Incorrect. Plausible because the Feedwater Regulating Bypass Valves will reposition, however, they go to 50% open.

Technical Reference(s): LP 2XIR06, Section 6.3.3.2; SO23-9-6, Section 6.6 and 6.7 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55260 / 52424

DESCRIBE the operation of Feedwater Control System controls, including the name, function, interlocks, and location of each.

STATE the names of the systems interfacing with the Feedwater Control System and DESCRIBE the flowpath and purpose of each interconnection.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	4, 7	
	55.43		

Comments / Reference: From LP 2XIR06, Section 6.3.3.2	Revision # 2

2XIR06 FEEDWATER CONTROL SYSTEMS

6.0 Lesson Presentation | **Activities**

Lesson Presentation	Activities
.2 Reactor Trip Override (RTO)	
.2.1 Purpose of RTO Remove decay heat without overcooling.	
.2.2 Conditions that will cause a 'RTO'	
.2.2.1 Reactor tripped signal is received from CEDMCS UV coils - Coils de-energize Either UV1 and UV3 or UV2 and UV4	
.2.2.1.1 RTO is in effect for a minimum of 10 sec.	Typically it takes 10 to 15 minutes for RTO to clear.
.2.3 FWCS Response	Inputs 5% flow signal downstream of the Master Controller (T-4).
.2.3.1 Feed Water Reg Valve goes Shut	Grounds main valve program input ("0" input at T-6)
.2.3.2 Feed Water Reg Bypass Valve goes to 50%	5% of rated feedwater flow signal to bypass valve. The 5% flow signal is equivalent to 25% Master Controller Output (i.e. 25% flow Demand). This equates to 50% valve position.
.2.3.3 Feed Pumps goto Minimum speed ~3600 rpm	End result we get ~5% of rated Feedwater flow.

Comments / Reference: From SO23-9-6, Section 6.6	Revision # 21

6.0 PROCEDURE (Continued)

6.6 Feedwater Control System Operation During a VALID RTO

REFERENCE USE

6.6.1 **Validate the RTO, as follows:**

.1 VERIFY Main Feedwater Control Valves Close.

.2 VERIFY Bypass Feedwater Control Valves ramp to 50%.

.3 VERIFY controller BIAS is set to ZERO on HIC-1107 and HIC-1108, Feedwater Pump Speed Control Stations.

.4 VERIFY Feed Pump speeds begin ramping down to ≈3600 RPM for 10 seconds and then modulate to control Valve delta pressure at approximately 100 psid. (LS-2.3, LS-2.4)

.5 If the Steam Generator levels are not being properly controlled by DCS, then GO TO SO23-13-24, Feedwater Control System Malfunction.

6.6.2 **Reset the Valid RTO, as follows:**

CONTINUOUS USE

.1 ENSURE controller BIAS is set to ZERO on HIC-1107 and HIC-1108, Feedwater Pump Speed Control Stations. ☐

.2 LOWER Master Controller (FIC-1111/FIC-1121) Setpoint to within 4% of actual S/G Level. ☐

.3 After the RTO has reset, then set the Master Controller, (FIC-1111/FIC-1121), setpoint to 55% NR level, or as directed by the SRO Ops. Supv. ☐

.4 If the Steam Generator levels are not being properly controlled by DCS, then GO TO SO23-13-24, Feedwater Control System Malfunction.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 A1.02	
	Importance Rating	3.3	

Auxiliary/Emergency Feedwater System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: Steam generator pressure

Proposed Question: Common 18

Given the following conditions:

- Emergency Feedwater Actuation Signals 1 and 2 (EFAS-1 & EFAS-2) have actuated following a trip from full power.
- A manual Main Steam Isolation Signal was actuated following reports of steam in the Turbine Building.
- Steam Generator E088 is at 750 psia and 15% narrow range level.
- Steam Generator E089 is at 795 psia and 18% narrow range level.

Under these conditions, which of the following Steam Generator(s) are being automatically fed by Emergency Feedwater Actuation Signals/Auxiliary Feedwater System?

A. Only Steam Generator E088.

B. Only Steam Generator E089.

C. Both Steam Generators E088 and E089.

D. Neither Steam Generator E088 nor E089.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because E088 has the lower level, and is below the setpoint for Diverse Emergency Feed Actuation Signal, however, DEFAS is not applicable (needs ATWS, and no MSIS present) and both SG levels are below the 21% EFAS actuation setpoint.

B. Incorrect. Plausible because it may be thought that with MSIS present, only the SG with the higher pressure will be fed, however, SG pressures must be below 741 psia and the difference must be at least 125 psi.

C. Correct. With EFAS-1 and 2 present, both SG pressures above 741 psia, and both SG levels below 21%, both SGs will be fed.

D. Incorrect. Plausible because if both SG pressures were below 741 psia (setpoint for MSIS), and pressures within 125 psia as listed, neither SG would be fed. However, the MSIS was manual and SG pressures are above 741 psi.

Technical Reference(s) SD-SO23-780, Pages 77 & 83 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52728 / 55262
DESCRIBE the instrumentation used to monitor the operation of the Auxiliary Feedwater System, including the name, function, sensing points, normal values for the parameters being measured and location of each instrument.
DESCRIBE the cause/effect relationships associated with the following Auxiliary Feedwater System conditions/operations: The effect on EFAS components of an MSIS actuation.

Question Source: Bank # 129830
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: From SD-SO23-780, Page 77	Revision # 10

2.0 DESCRIPTION (Continued)

2.3.3 **Auxiliary Feedwater Pump Discharge Control Valves, 2(3)HV-4713, 4712, 4706 & 4705** (Continued)

.2.4 When placing the Isolation Switch in the LOCAL position, the Valve can only be controlled from its respective Motor Control Center; both Control Room control and EFAS and MSIS automatic operation are prevented.

.3 The LOCAL position is used following a Control Room fire or evacuation to protect the Valve from any spurious signal that could inadvertently actuate it or inhibit its operation.

.3.1 The Fire Isolation Switch is normally in the REMOTE position. Placing the Fire Isolation Switch in the LOCAL position will energize an Control Room annunciator, 57A14.

.4 Actuation of an MSIS automatically CLOSES the Pump Discharge Control Valves, provided an EFAS is not present.

.4.1 Should any Control Valve receive an EFAS, the MSIS to the Valve is blocked, at the equipment level, allowing the Valve to be controlled by the EFAS logic.

.4.2 The generation of an EFAS DE-ENERGIZES both "CYCLING" and "NON-CYCLING" Relays which OPEN or CLOSE contacts in the control power circuitry of the Valve.

.4.3 DE-ENERGIZING of both types of relays automatically OPENS the Valve.

.4.4 When the EFAS CLEARS, the Cycling Relays become RE-ENERGIZED to automatically CLOSE the Valve.

.4.5 The Non-cycling Relays remain DE-ENERGIZED.

.4.6 Depressing EFAS RESET Pushbuttons located on Auxiliary Relay Cabinets A and B, will RE-ENERGIZE the Non-cycling Relays and allows the Valve to be controlled normally.

.5 When an EFAS Signal is present (and both Cycling and Non-cycling Relays are DE-ENERGIZED), the affected valve OVERRIDE pushbutton control on Control Room Panel, 2(3)CR-52, is not functional.

.5.1 The OVERRIDE Pushbutton becomes functional only after the EFAS Signal has CLEARED.

Comments / Reference: From SD-SO23-780, Page 83	Revision # 10

2.0 DESCRIPTION (Continued)

2.3.6 **Emergency Feedwater Actuation Signal** (Continued)

.4 EFAS 1 Train A controls AC powered components and Train B controls DC powered components.

.5 EFAS 2 Train A controls DC powered components and Train B controls AC powered components.

.6 An EFAS 1 is initiated and Auxiliary Feedwater delivered to Steam Generator #1, 2(3)E-089, if:

.6.1 The water level in Steam Generator #1 falls below the low level setpoint of 21% Narrow Range and Steam Generator #1 pressure is above the MSIS setpoint (741 psia during normal operation), **or**

.6.2 The water level in Steam Generator #1 falls below the low level setpoint of 21% Narrow Range, Steam Generator #1 pressure is below the MSIS setpoint, and the pressure in Steam Generator #1 is at least 125 psi greater than Steam Generator 2.

.7 Initiation of an EFAS 2 for Steam Generator #2, 2(3)E-088, is similar. (See Figures 14 & 15)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	061 K1.07	
Importance Rating	3.6	

Auxiliary/Emergency Feedwater System: Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: Emergency water source

Proposed Question: Common 19

Given the following conditions:

- A seismic event has occurred and T-120, Condensate Storage Tank has ruptured.
- Unit 2 has tripped and a Loss of Offsite Power has occurred.
- Annunciator 53B58 - CONDENSATE TANK T-121 LEVEL HI / LO is in alarm.
- SO23-12-9, Functional Recovery Heat Removal Success Paths to provide makeup water to T-121, Condensate Storage Tank are being implemented.

With a REQUIRED cooldown rate of 110°F per hour to Shutdown Cooling System operation, which ONE (1) of the following is the preferred method for makeup to T-121, Condensate Storage Tank?

A. Makeup from the Demineralized Water Storage Tanks using the bypass around the Level Control Valve.

B. Directly from the hard-piped Fire Water Supply system.

C. Semi-automatic makeup from the T-120 vault using the Condensate Transfer Pump.

D. Cross-tie from Unit 3 Condensate Storage Tanks.

Proposed Answer: A

Explanation:

A. Correct. Normal automatic makeup valves fail closed and require actions to manually align the bypass valves.
B. Incorrect. Plausible because it is an available method, however, it would be the least preferred method of refill.
C. Incorrect. Plausible because under certain conditions the T-120 vault could be aligned to the Condensate Transfer Pump, however, it is not the preferred method in this condition.
D. Incorrect. Plausible because a cross-tie is available from the Unit 3 Condensate Storage Tanks, however, this piping will divert water to the Unit 2 Hotwell not to T-121.

Technical Reference(s)	SO23-12-9, Success Path HR-1, Step 10	Attached w/ Revision # See Comments / Reference
	SO23-15-53.B, 53B58	
	SD-SO23-320, Page 4	

Proposed references to be provided during examination: None

Learning Objective: 55217 — Per the Functional Recovery procedure SO23-12-9 DESCRIBE: The basis for each step, caution or note.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	5, 10
	55.43	

Comments / Reference: From SO23-12-9, HR-1, Step 10	Revision # 25

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 25
ATTACHMENT FR-5

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FUNCTIONAL RECOVERY

RECOVERY – HEAT REMOVAL

Success Path Actions: HR-1, S/G with no ECCS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10 VERIFY and MAINTAIN Condensate Inventory:	
a. INITIATE or CONTINUE FS-15, MONITOR Condensate Inventory.	
b. INITIATE or CONTINUE makeup to T-121, Condensate Storage Tank from at least one source: (listed in preferred order) 1) Demineralized Water Storage Tanks: T-266 T-267 T-268 OR 2) Gravity feed from T-120, Condensate Storage Tank: CLOSE – 1414MU092 OPEN – 1305MU476 OPEN – 1414MU052 OR 3) Crosstie to other Unit. (*SO23-9-5, CONDENSATE STORAGE AND TRANSFER*)	b. 1) IF required cooldown rate from SO23-12-11, Attachment 16, DETERMINE TIME UNTIL SHUTDOWN COOLING REQUIRED – less than 100°F/HR, THEN RAISE S/G steaming rate AND GO TO step 10c. 2) IF required cooldown rate from SO23-12-11, Attachment 16, DETERMINE TIME UNTIL SHUTDOWN COOLING REQUIRED – greater than 100°F/HR, THEN a) ENSURE Diesel Firewater Pump – operating. b) IF Firewater aligned to AFW System by step 7, THEN GO TO step 11. c) UNLOCK and CLOSE 1305MU082, Firewater to CST cross-connect drain valve. d) OBTAIN key number 71 from NOA. e) UNLOCK and OPEN 1305MU474, Firewater Supply to CST valve. f) GO TO step 11.

Comments / Reference: From SO23-15-53.B, 53B58	Revision # 16

53B58 CONDENSATE TANK T120 LEVEL HI/LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	WHITE	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)LSHL-4358	2(3)MT-120, Condensate Storage Tank, Level Switch HI/LO [RSMI]	HI 98.3% LO 86.2%	2(3)LI-4357A 2(3)LI-4357B 2(3)LI-4357C	NONE	1044/1055

1.0 **REQUIRED ACTIONS**:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
LOW LEVEL	
2.1 Normal usage	2.1 Verify 2(3)MT-120, Condensate Storage Tank, Auto makeup aligned per SO23-9-5, Section for Aligning Automatic Makeup to MT-120 using LV-4358. 2.1.1 If level continues to lower, then perform SO23-9-5, Attachment for Filling Condensate Storage Tank MT-120.

Comments / Reference: From SO23-15-53.B, 53B58	Revision # 16

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ALARM RESPONSE INSTRUCTION
REVISION 16
ATTACHMENT 2

SO23-15-53.B
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53B58 CONDENSATE TANK T120 LEVEL HI/LO (Continued)

2.0 CORRECTIVE ACTIONS: (Continued)

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
LOW LEVEL (Continued)	NOTE: Upon loss of power to 2(3)Q019 (at MCC BC), LV-4358 will fail Closed, isolating Automatic and Semi-automatic make-up to MT-120.
2.2 Auto makeup failure	2.2 Isolate Auto makeup and fill per SO23-9-5, Attachment for Filling Condensate Storage Tank MT-120.
2.3 System leak	2.3 Isolate the leak.
2.4 Inventory Diversion	2.4 Verify non-safety related makeup connections, e.g., Condenser makeup and BPS Sluice Water have not spuriously actuated as a result of a seismic event thereby diverting inventory from 2(3)MT-120.
HIGH LEVEL	
2.5 Auto makeup failure	2.5 CLOSE S2(3)1417MU225, 2(3)LV-4358 Inlet Isolation Valve.
2.6 Excessive Condenser Drawoff Operation	2.6 Consider disabling Drawoff and using Overboarding to maintain Hotwell levels, as follows:
	2.6.1 Deactivate 2(3)LV-3245, Condenser Draw Off Valve, by depressing DISABLE (HS-3245).
	2.6.2 Overboard the Hotwells per SO23-9-9.

2.0 DESCRIPTION

2.1 System Overview

2.1.1 **Main Flow Path** (See Figure 1)

.1 The Condensate Storage and Transfer Systems supplies the water through the use of two Condensate Storage Tank 2(3)T-120 and 2(3)T-121. 2(3)T-121 is a Seismic Category I Tank and is used solely for providing water to the Auxiliary Feedwater Systems for plant cooldown. 2(3)T-120 is a Seismic Category II Tank enclosed with a Seismic Category I retaining wall and is the primary source of makeup for 2(3)T-121. 2(3)T-121 can receive water by Transfer Pump, gravity feed or from the 2(3)T-120 Sump if the Tank is damaged. Also 2(3)T-120 is the normal source of makeup for the Secondary System.

Comments / Reference: From SD-SO23-320, Page 4	Revision # 13

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UNITS 2 AND 3 REVISION 13 PAGE 4 OF 30

2.0 DESCRIPTION

2.1 System Overview

2.1.1 **Main Flow Path** (See Figure 1)

.1 The Condensate Storage and Transfer Systems supplies the water through the use of two Condensate Storage Tank 2(3)T-120 and 2(3)T-121. 2(3)T-121 is a Seismic Category I Tank and is used solely for providing water to the Auxiliary Feedwater Systems for plant cooldown. 2(3)T-120 is a Seismic Category II Tank enclosed with a Seismic Category I retaining wall and is the primary source of makeup for 2(3)T-121. 2(3)T-121 can receive water by Transfer Pump, gravity feed or from the 2(3)T-120 Sump if the Tank is damaged. Also 2(3)T-120 is the normal source of makeup for the Secondary System.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	062 K3.02	
Importance Rating	4.1	

AC Electrical Distribution System: Knowledge of the effect that a loss or malfunction of the AC distribution system will have on the following: Emergency diesel generator

Proposed Question: Common 20

Given the following conditions on Units 2 and 3:

- Low voltage alarms on Buses 2A04, 2A06, 3A04 and 3A06 are annunciating.
- All 1E Bus voltages are approximately 3750 VAC.
- No SIAS actuation is present on either Unit.
- All other equipment is OPERABLE.

Which ONE (1) of the following identifies how the Voltage Protection Circuits respond on Unit 2?

Unit 2 Emergency Diesel Generators...

A. will start; Unit 3 energizes Buses 2A04 and 2A06 via Bus Tie Breakers.

B. remain off; Unit 3 energizes Buses 2A04 and 2A06 via Bus Tie Breakers.

C. will start; 2G002 energizes Bus 2A04 and 2G003 energizes Bus 2A06.

D. remain off; Unit 2 continues to supply Buses 2A04 and 2A06.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because plausible because the EDGs will start, however, with a LOVS signal present the Preferred Source of power to the Unit 2 Buses is the Emergency Diesel.
B. Incorrect. Plausible because under normal conditions the preferred source of power to the Unit 2 Buses is unit 3, however, a Loss of Voltage Signal (LOVS) will start the Emergency Diesel.
C. Correct. Given the conditions listed, the EDGs will start and power their respective 1E Buses.
D. Incorrect. Plausible because if voltage had remained above 3796 VAC <u>and</u> < two minutes had expired (SDVS) the EDGs would not start and 1E Buses would remained energized from Unit 2.

Technical Reference(s) SO23-15-63.B, 63B05 / SD-SO23-120, Page 109 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53492 / 52795	ANALYZE normal and abnormal operations of the Emergency Diesel Generators (EDGs) System. EXPLAIN the interfaces between the Emergency Diesel Generators (EDGs) System and other plant systems.

Question Source: Bank # 112931
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: From SO23-15-63.B, 63B04	Revision # 13

NUCLEAR ORGANIZATION
UNITS 2

ALARM RESPONSE INSTRUCTION
REVISION 13
ATTACHMENT 2

SO2-15-63.B
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63B05 2A04 VOLTAGE LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	RED [1]	N/A	63B15, 63B25, 63B28, 63B35, 63C11

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK #
127 Device	Undervoltage Device	N/A	2EI-1662	EY8180	1827

1.0 **REQUIRED ACTIONS**:

1.1 Notify the Unit 2 Control Room of the low voltage alarm.

1.2 Verify voltage indication and perform the following:

1.2.1 If 2A04 Control Room voltage indication is lost, and loads are still operating, then a C-A Phase PT failure may have occurred. (AR 031000894-2)

VOLTAGE IS > 4154

1.2.2 Locally inspect 2A0421 (PT Cubicle) for any Degraded Voltage Relays (127D-1, 2, 3 & 4) tripped (light ILLUMINATED).

.1 Notify the CRS/SM and the STA to review Tech. Spec. LCO 3.3.7, and initiate corrective actions, as required.

.2 Notify Test Techs and generate a notification to investigate/repair. |

VOLTAGE IS > 3796 and ≤ 4154

NOTE

When the Diesel Generator Output breaker is closed, then the SDVS (Sustained Degraded Voltage Signal) circuitry is defeated. If the Diesel Generator is in parallel with the preferred power source, and confirmed degraded voltage condition exists, then the Diesel Generator Output breaker must be opened to allow the SDVS timing relays to auto sequence (110 ± 22 sec.) to protect Class 1E equipment.

1.2.3 Degraded Grid Voltage condition exists. (110 ± 22 second timer starts when alarm annunciates.)

.1 Unload the Diesel Generator.

.2 Ensure Open Diesel Generator Output Breaker 2A0413.

Comments / Reference: From SO23-15-63.B, 63B04	Revision # 13

NUCLEAR ORGANIZATION
UNITS 2

ALARM RESPONSE INSTRUCTION
REVISION 13
ATTACHMENT 2

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63B05 2A04 VOLTAGE LO (Continued)

1.0 **REQUIRED ACTIONS**: (Continued)

VOLTAGE IS ≤ 3796

1.2.4 LOVS condition exists.

.1 If 2MG-002, Diesel Generator, is running and 2A04 can not be immediately energized, then STOP 2MG-002, Diesel Generator by selecting 2HS-1767-1, Maintenance Lockout Switch, to Lockout.

1.2.5 If a B-C Phase PT failure occurred, then 2A04 sync circuit will not be available. (AR 031000894-2)

.1 Declare 2G002 INOPERABLE.

1.3 Initiate SO23-13-4.

1.4 If 2A04 becomes de-energized, then Implement SO23-13-26, Attachment for Loss of a 1E 4 kV Bus.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Reserve Aux Transformer relays with failure of Bus Low Voltage Transfer	2.1 Refer to applicable Reserve Aux Transformer. Protection Trip ARP 63C01, 63C11, 63C21 or 63C31.
2.2 Trip of 2A0413, 2A0417, 2A0418 or 2A0419, 2A04 Supply Breakers	2.2 Refer to SO23-6-9, Section for 4 kV Bus Fault Relay.
2.3 2MG-002, Diesel Generator, Trip	2.3 Dispatch an Operator to Diesel Generator Building to investigate the cause of the Generator Trip.
2.4 Fault on 2XU1, Unit Auxiliary Transformer, while back feeding the 2XM1, Main Transformer, with failure of Bus Low Voltage Transfer	2.4 Refer to SO23-6-9, 4 kV Bus Fault Relay.

Comments / Reference: From SD-SO23-120, Page 109	Revision # 19

PART III 1E 4.16 kV AND 480 V ELECTRICAL DISTRIBUTION SYSTEM

2.0 DESCRIPTION (Continued)

2.1.4 **General Control Scheme**

1. A description of the automatic transfer capability that exists between the 4.16 kV 1E buses of each Unit is discussed below. For simplicity, this discussion covers only one load group (2A04). However, similar operations take place on the redundant load group and the load groups associated with the other unit.

2. Bus 2A04 is normally supplied from Reserve Auxiliary Transformer (2XR1). If power from the Reserve Auxiliary Transformer (2XR1) to bus 2A04 is lost, the following actions take place:

3. The LOVS or SDVS (Sustained Degraded Voltage System) also sends a signal to start Diesel Generator 2G002.

4. After the residual voltage at bus 2A04 has decayed to approximately 25%, as detected by bus 2A04 residual voltage relays, the LOVS or SDVS signals the Unit 3 bus tie circuit breaker 3A04-16 to close provided bus 3A04 has normal voltage and is being powered from its respective Reserve Auxiliary Transformer 3XR1 or Unit Auxiliary Transformer 3XU1.

5. After 3A04-16 closes, Unit 2 bus tie circuit breaker 2A04-17 closes and bus 2A04 will be powered from Reserve Auxiliary Transformer 3XR1, through bus 3A04.

6. If bus 3A04 is not being supplied by Reserve Auxiliary Transformer 3XR1 as detected by Reserve Auxiliary Transformer breaker (3A04-18) not being closed or if bus 3A04 has no voltage, the transfer will not be permitted and bus tie breaker 3A04-16 will not close. Additionally an interlock prevents crosstie circuit breaker 3A04-16 from closing if the Unit 3 Diesel Generator (3G002) circuit breaker 3A04-13 is closed and the Diesel Generator is supplying its designated load group. This prevents the Diesel Generator from supplying two load groups and overloading the Diesel Generator, since the Diesel Generator is only rated to carry the loads associated with one load group. However, if Diesel Generator (3G002) is paralleled with the Reserve Auxiliary Transformer (3XR1) during a periodic load test, a LOVS or SDVS at Bus 2A04 will initiate a transfer to Bus 3A04 and the Unit 3 Diesel Generator breaker 3A04-13 will be tripped.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	063 K4.02	
	Importance Rating	3.9	

DC Electrical Distribution System: Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties

Proposed Question: Common 21

With Unit 2 in MODE 1, which ONE (1) of the following describes the allowable lineup of B022, Swing Battery Charger?

A. Supply 1E DC Bus D1.

B. Supply Non-1E DC Bus D5.

C. Cross-tie 1E DC Buses D2 and D4.

D. Cross-tie 1E DC Buses D2 and Non-1E DC Bus D5.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that that the odd numbered DC buses were the ones supplied by Swing Charger B022, however, Swing Charger B022 supplies the even-numbered buses and DC Bus D5.
B. Correct. Swing Charger B022 supplies the even-numbered Buses and Non-1E DC Bus D5.
C. Incorrect. Plausible because these Buses can be cross-tied, however, there is a KIRK Key interlock between these two breakers preventing the Swing Battery Charger from supplying both Buses.
D. Incorrect. Plausible because both Buses are supplied from the Swing Battery Charger, however, there is a KIRK Key interlock between these breakers preventing the Swing Battery Charger from supplying both Buses.

Technical Reference(s) SD-SO23-140, Figure I-1; SO23-6-15, Attachment 17 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

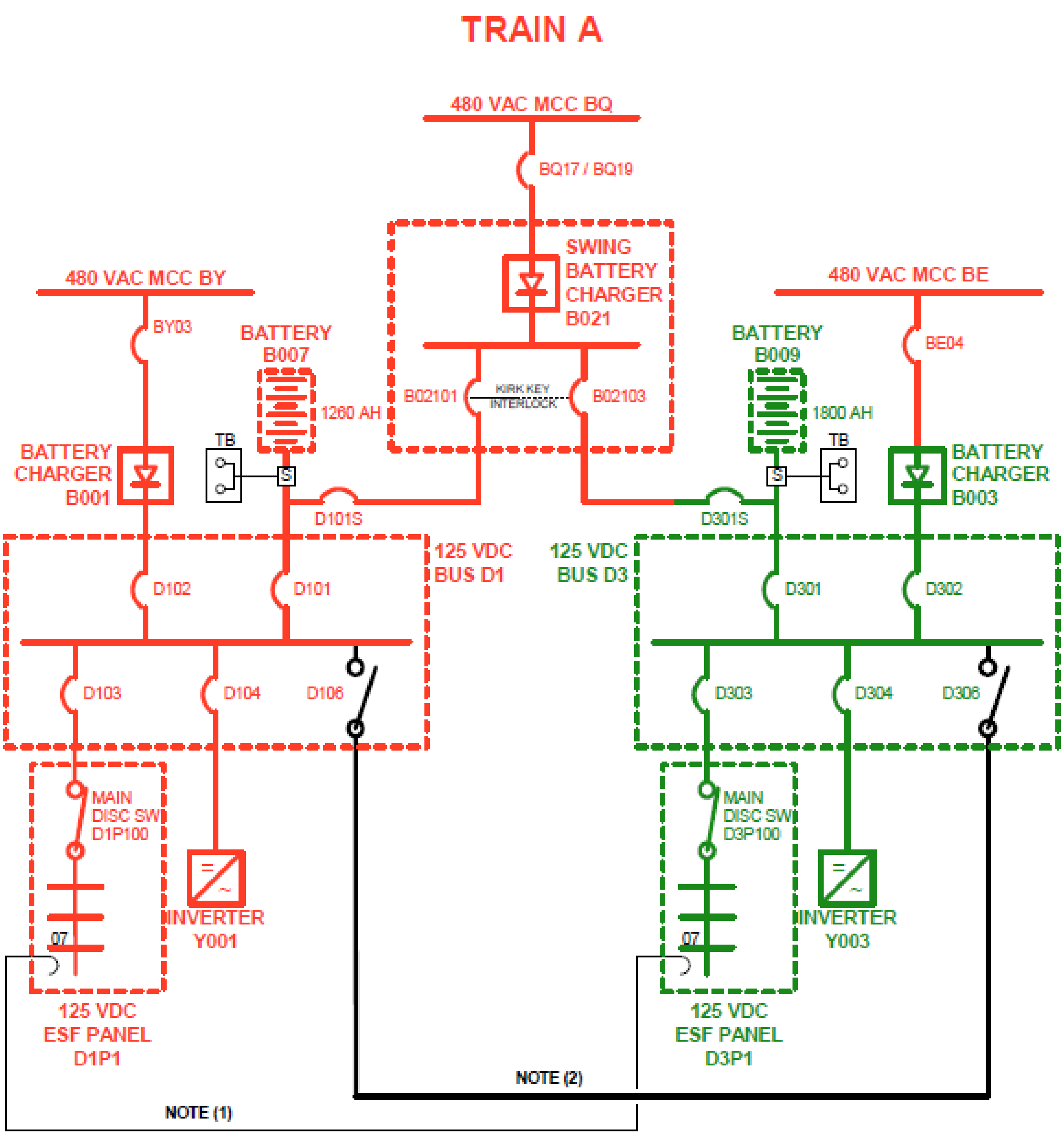
Learning Objective: 80702 / 80707

IDENTIFY Non-IE 120 VAC and 125 VDC Power Supply System flowpaths, components, and locations including being able to draw and label system diagrams.

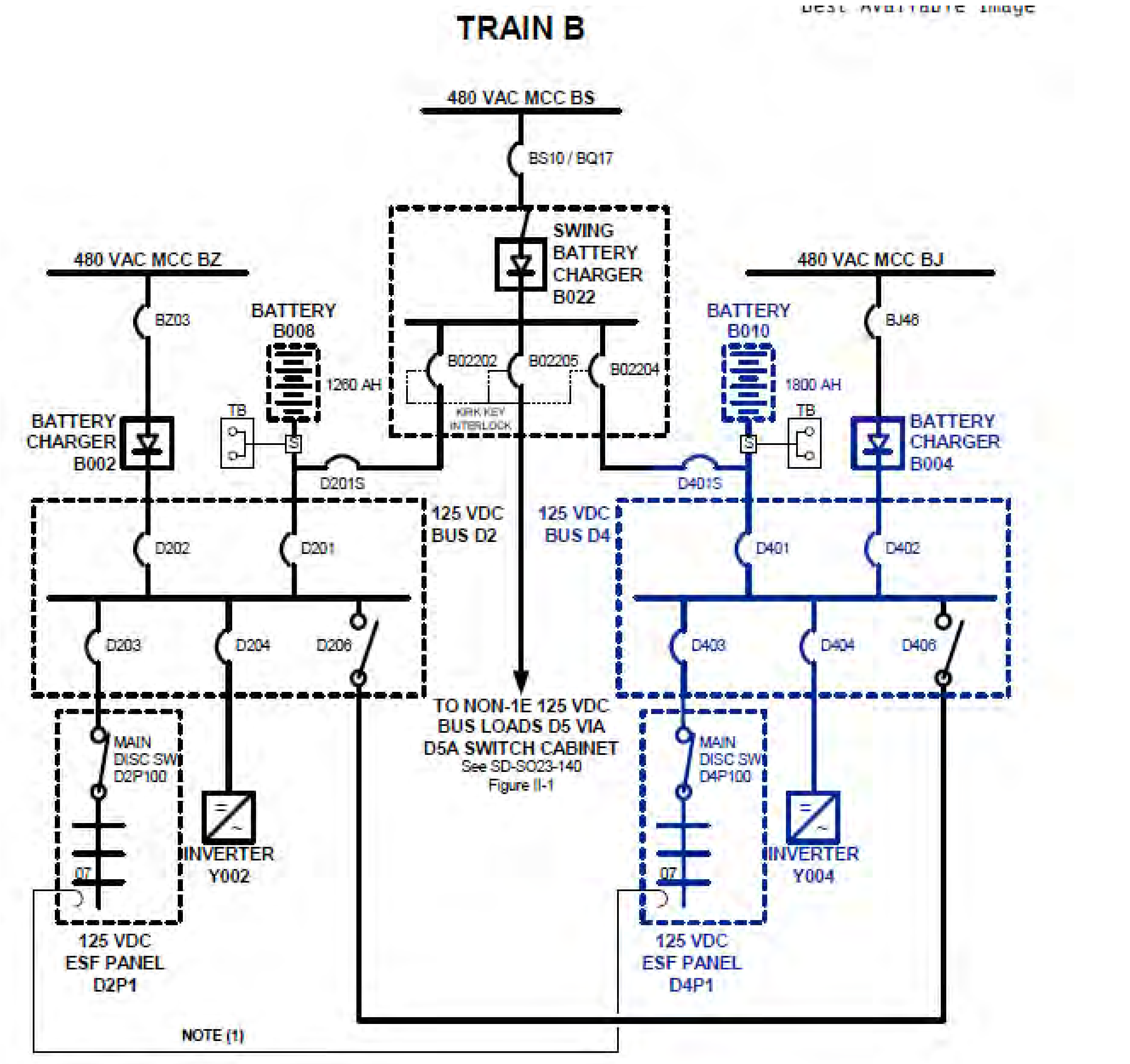
EXPLAIN the interfaces between the Non-IE 120 VAC and 125 VDC Power Supply System and other plant systems.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7, 10	
	55.43		

Comments / Reference: From SD-SO23-140, Figure I-1	Revision # 15



Comments / Reference: From SD-SO23-140, Figure I-1	Revision # 15



Comments / Reference: From SO23-6-15, Attachment 17	Revision # 29

NUCLEAR ORGANIZATION
UNITS 2 AND 3

OPERATING INSTRUCTION
REVISION 29
ATTACHMENT 17

SO23-6-15
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B022, SWING BATTERY CHARGER, OPERATIONS

CONTINUOUS USE

OBJECTIVE

Provide direction on placing B022, Swing Battery Charger, in service to DC Bus D2, D4, D5, or D2 or D4 Battery Bank (whether initially connected to the DC Bus or not). In order to prevent overloading the Diesel Generator, loads on MCC BS are restricted prior to energizing B022. B022 is connected to the Bus and/or Battery and the Dedicated Battery Charger is disconnected.

Provide direction on removing B022, Swing Battery Charger, from DC Bus D2, D4, D5, or D2 or D4 Battery Bank (whether connected to the DC Bus or not). The Dedicated Battery Charger will be placed in service, including Closing the Associated Battery Breaker if Open, and B022 will be removed from service.

UNIT ________ MODE _______ DATE ______________________ TIME ________

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 K2.02	
Importance Rating	2.8	

Emergency Diesel Generator System: Knowledge of bus power supplies to the following: Fuel oil pumps

Proposed Question: Common 22

Which ONE (1) of the following is the power supply to Emergency Diesel Generator 2G003 Fuel Oil Priming Pumps?

A. 125 VDC to Panel 2D1-P1

B. MCC 2BJ

C. 125 VDC to Panel 2L-161

D. MCC 2BQ

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because DC Panel 2D1P1 is an ESF panel, however, it is powered from the Train A side.
B. Incorrect. Plausible because MCC 2BJ is powered from Bus 2B06 which can be supplied from EDG G003.
C. Correct. The Fuel Oil Priming Pumps power supply for EDG 2G003 is DC Panel L-161. The Main Fuel Oil Pumps are engine driven.
D. Incorrect. Plausible because MCC 2BQ is powered from Bus 2B04 which can be supplied from EDG 2G002.

Technical Reference(s): SD-SO23-750, Page 88; SD-SO23-750, Figure III-1; SD-SO23-140, Figure I-2A — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 73305 DESCRIBE the configuration and operational characteristics of Emergency Diesel Generator Electrical Systems components.

Question Source:

Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments / Reference: From SD-SO23-750, Page 88	Revision # 16

PART III FUEL OIL SYSTEM

2.0 DESCRIPTION (Continued)

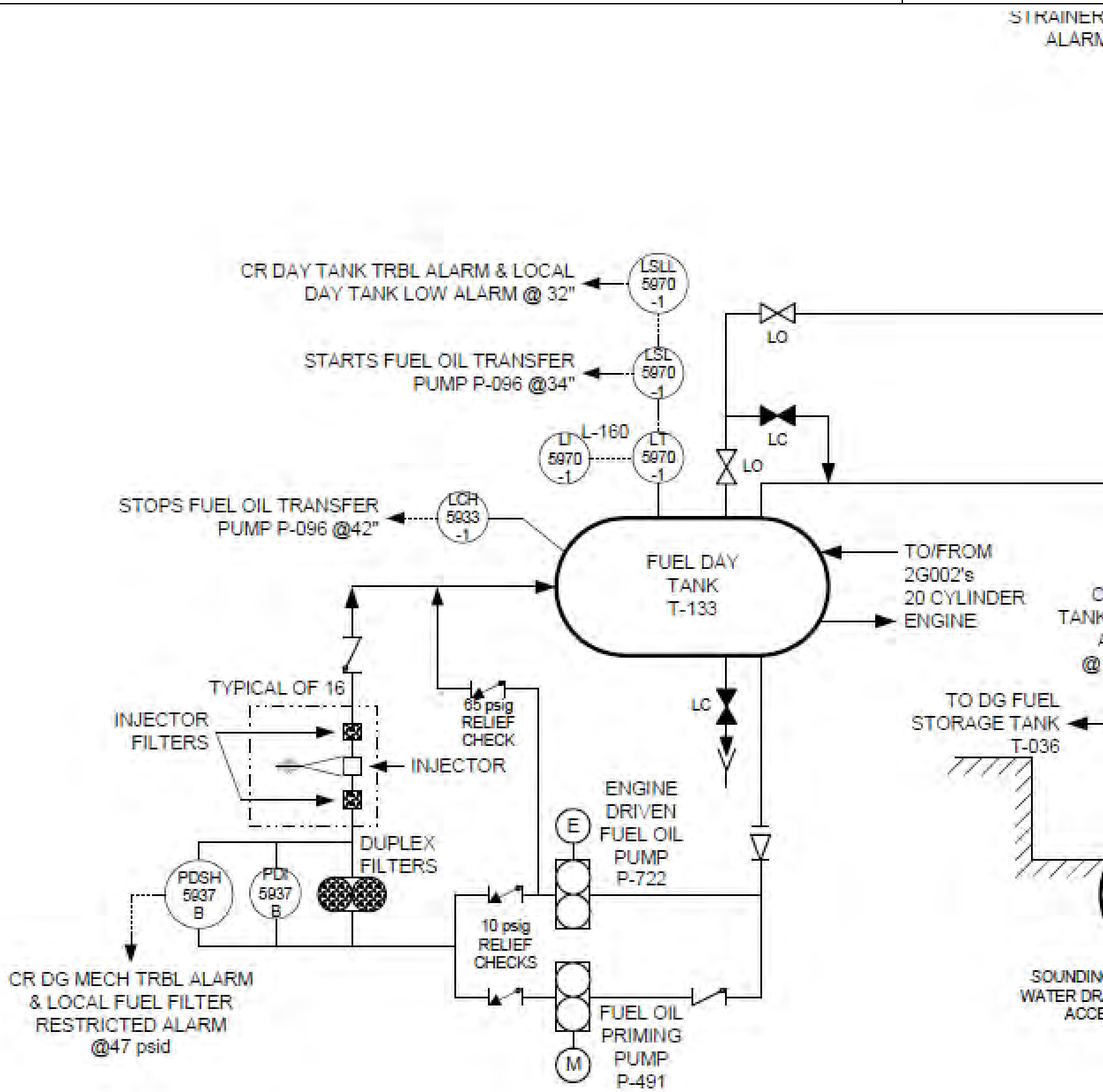
2.3 Detailed Control Scheme

2.3.1 See General Control Scheme (Part III, Section 2.1.2) and Component Description (Part III, Section 2.2.1 to Section 2.2.6).

2.4 Power Supplies

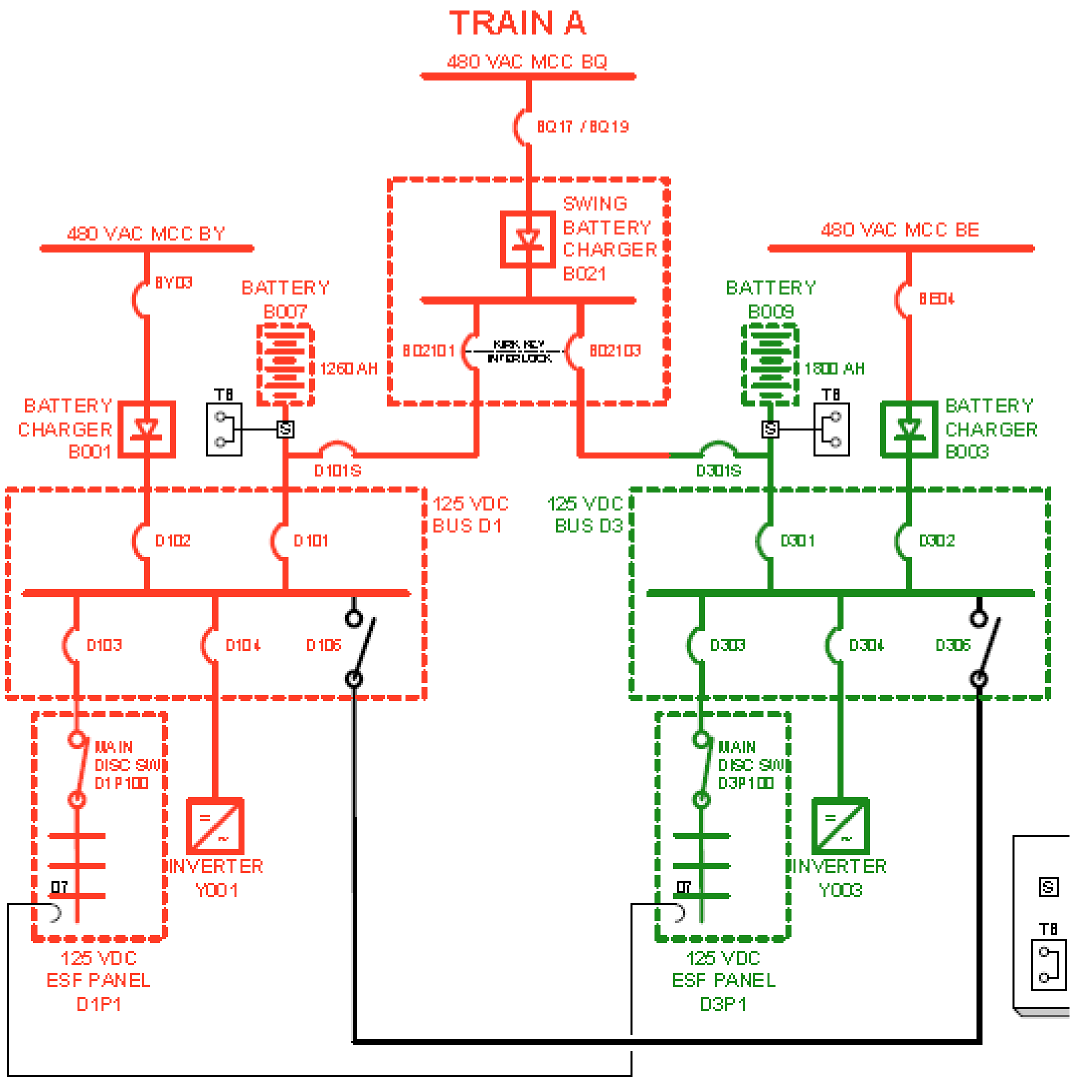
COMPONENT	MCC BREAKER	LOCATION
DG Fuel Oil Transfer Pump:		
P096	BD24	G002 Room; 30' Elev
P093	BD23	G002 Room; 30' Elev
P094	BH09	G003 Room; 30' Elev
P095	BH08	G003 Room; 30' Elev
Motor Enclosure Heaters for Fuel Oil Transfer Pump:		
P096	BD18	G002 Room; 30' Elev
P093	BD18	G002 Room; 30' Elev
P094	BH03	G003 Room; 30' Elev
P095	BH03	G003 Room; 30' Elev
COMPONENT & UNIT 2(3)	**LOCAL DIESEL CONTROL PANEL**	**LOCATION**
Fuel Priming Pump supplied with 125 VDC Power		
P491	L-160	G002 Room; 30' Elev
P490	L-160	G002 Room; 30' Elev
P493	L-161	G003 Room; 30' Elev
P492	L-161	G003 Room; 30' Elev

Comments / Reference: From SD-SO23-750, Figure III-1	Revision # 16



NOTE: FUEL OIL SYSTEM FOR DIESEL GENERATOR 2GOO3 ARE IDENTICAL EXCEPT FOR EQUIPMENT NUMBERS

Comments / Reference: From SD-SO23-140, Figure I-2A	Revision # 15



NOTE: FUEL OIL SYSTEM FOR DIESEL GENERATOR 2G003 ARE IDENTICAL EXCEPT FOR EQUIPMENT NUMBERS

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 A2.08	
Importance Rating	2.7	

<u>Emergency Diesel Generator System</u>: Ability to (a) predict the impacts of the following malfunctions or operations on the EDG system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of opening / closing breaker between buses (VARS, out of phase, voltage)

Proposed Question: Common 23

Given the following conditions:

- Emergency Diesel Generator (EDG) 3G002 is being paralleled to 1E Bus 3A04 which is aligned to the Reserve Auxiliary Transformer.
- 3G002 Output Breaker is closed with EDG voltage <u>greater than</u> 1E Bus 3A04 voltage.

Which ONE (1) of the following:

1.) Identifies the impact on the Emergency Diesel Generator?

2.) What action must be taken?

A. 1.) EDG VAR meter will move in the negative (-) VAR (BUCK) direction.
2.) ADJUST Voltage Regulator to establish a positive VAR load (+0.1 to +0.5 MVARS).

B. 1.) EDG VAR meter will move in the positive (+) VAR (BOOST) direction.
2.) ADJUST Voltage Regulator to establish a positive VAR load (+0.1 to +0.5 MVARS).

C. 1.) EDG VAR meter will move in the negative (-) VAR (BUCK) direction.
2.) ADJUST Voltage Regulator to establish a negative VAR load (-0.1 to -0.5 MVARS).

D. 1.) EDG VAR meter will move in the positive (+) VAR (BOOST) direction.
2.) ADJUST Voltage Regulator to establish a negative VAR load (-0.1 to -0.5 MVARS).

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this would be the correct action if generator voltage were lower than Bus 3A04 voltage when the breaker was closed and it was desired to establish a positive VAR load.

B. Correct. With EDG voltage greater than bus voltage when the breaker is closed, a positive VAR load will be "supplied out" the Emergency Diesel Generator (known at SONGS as BOOST). Because Diesel voltage is significant kindling greater than bus voltage, the Voltage Control Switch is adjusted to decrease generator terminal voltage and establish the VAR load specified.

C. Incorrect. Plausible if thought that this would be the correct action if generator voltage were lower than Bus 3A04 voltage when the breaker was closed and that it was desirable to establish a negative VAR load.

D. Incorrect. Plausible because the VAR meter will move in the positive direction, however, adjusting the Voltage Control Switch in this fashion will cause more VARs to be "absorbed into" the Generator.

Technical Reference(s) SO23-2-13, Attachment 2, Step 2.6 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56392 / 56520
MONITOR the operation of the Diesel Generators per SO23-2-13.
SYNCHRONIZE a Diesel Generator to the 4 kV Bus per SO23-2-13 or SO23-3-3.23.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.41 5
55.43

Comments / Reference: From SO23-2-13, Attachment 2, Step 2.6	Revision # 37

NUCLEAR ORGANIZATION UNITS 2 AND 3 | OPERATING INSTRUCTION REVISION 37 ATTACHMENT 2 | SO23-2-13 PAGE 36 OF 175

2.0 PROCEDURE (Continued)

2.6 Paralleling a Diesel Supplied Isochronous Bus to the RAT: (LS-6.6)

f 2.6.1 Ensure the affected Switchgear Room is clear of all unnecessary personnel and maintain it clear until after the Diesel is paralleled to the 4kV bus.

2.6.2 Verify that the associated Reserve Auxiliary Transformer is energized and available to pick up the load.

2.6.3 PLACE Synchronization Master Control switch to ON.

2.6.4 DEPRESS the Reserve Auxiliary Transformer XR1(XR2) FDR BKR A0418 (A0618) SYNC Pushbutton.

2.6.5 Using HS-1669-1(HS-1648-2), VOLTAGE REGULATOR, MATCH incoming and running voltages at the synchroscope.

2.6.6 Using HS-1671-1(HS-1650-2), GOVERNOR CONTROL, ADJUST D/G SPEED so that the synchroscope is ***moving slowly in the clockwise direction.***

NOTE

To prevent a reverse power condition, the Diesel should have a minimum load applied immediately after being paralleled to the 4kV bus.

2.6.7 When the Synchroscope is within "3 minutes" of the straight up position, then CLOSE the Reserve Auxiliary Transformer Breaker. (LS-6.8)

2.6.8 RAISE LOAD on the Diesel to approximately 1.2 MW by depressing HS-1671-1(HS-1650-2), GOVERNOR CONTROL.

2.6.9 VERIFY ILLUMINATED HS-1671-1(HS-1650-2), GOVERNOR CONTROL DROOP IN light.

2.6.10 MAINTAIN VARS between 0.1 to 0.5 MVARS positive by adjusting the D/G Voltage Regulator using HS-1669-1 (HS-1648-2), VOLTAGE REGULATOR.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	073 A1.01	
Importance Rating	3.2	

Process Radiation Monitoring System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels

Proposed Question: Common 24

Given the following conditions:

- Unit 3 is in MODE 6.
- Irradiated fuel movement is in progress.
- A spent fuel assembly is damaged while being transported to the spent fuel racks.
- RE-7822 and RE-7823, Fuel Handling Building (FHB) Air Exhaust Process Radiation Monitors high alarms have actuated.

Which ONE (1) of the following describes the resulting ventilation alignment and effect on radiation levels?

Fuel Handling Building normal...

A. supply fan trips, normal exhaust fan remains running, and PACUs align to lower radiation levels.

B. supply and exhaust fans trip and PACUs align to lower radiation levels.

C. supply fan remains on, normal exhaust fan trips and radiation levels lower.

D. supply and exhaust fans trip, and radiation levels remain the same.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this condition would create a negative pressure inside the FHB and radiation levels do lower because the PACUs are aligned, however, this alignment would allow a radioactive release.
B. Correct. With a Process Radiation Monitor high alarm the normal supply and exhaust fans will trip and PACUs will take suction from the FHB atmosphere and discharge back into the FHB to lower radiation level.
C. Incorrect. Plausible because this flowpath would suspend the release to atmosphere and adding air to the Fuel Handling Building could dilute the atmosphere, however, both of these fans trip.
D. Incorrect. Plausible because the normal supply and exhaust fans will trip, however, the PACUs are aligned to reduce radiation level.

Technical Reference(s): SD-SO23-435, Page 22; SD-SO23-435, Figure 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 81409 / 81410
DESCRIBE the configuration and operational characteristics of Fuel Handling Building HVAC System components.
INTERPRET instrumentation and controls utilized in the Fuel Handling Building HVAC System.

Question Source:
Bank # 127549
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam SONGS 2006

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

10 CFR Part 55 Content:
55.41 7, 8
55.43 ______

Comments / Reference: From SD-SO23-435, Page 22	Revision # 3

NUCLEAR ORGANIZATION
UNITS 2 AND 3

SYSTEM DESCRIPTION SD-SO23-435
REVISION 3 PAGE 22 OF 35

3.0 OPERATIONS (Continued)

3.3 Emergency Operations

3.3.1 Fuel Handling Building Isolation

.1 An automatic Fuel Handling Isolation Signal is initiated if train "A" radiation monitor 2(3)RE-7822G1, or train "B" radiation monitor 2(3)RE-7823G2 reaches an alarm state. High gaseous activity level, or a loss of power to the Radiation Monitoring Panel, 2(3)L103, or Vital Bus 2(3)Y01 or 2(3)Y02 will produce the initiating signal. A manual Fuel Handling Isolation Signal can be initiated from 2(3)CR60 by depressing 2(3)HS-7822A1 for Train A or 2(3)HS-7823A2 for Train B. If the Radiation Monitor is in "Bypass" it does not prevent a manual FHIS actuation or an auto actuation due to loss of power to the Vital Bus, Aux Relay Cabinets, or the Remote Display Unit (RDU).

.2 Upon receipt of a Fuel Handling Isolation Signal (FHIS), the associated Train's Isolation Dampers go closed to isolate the FHB:

.2.1 High airborne radiation sensed by RE7822 CLOSES the TRAIN A Dampers ONLY.

.2.2 High airborne radiation sensed by RE7822 CLOSES the TRAIN B Dampers ONLY.

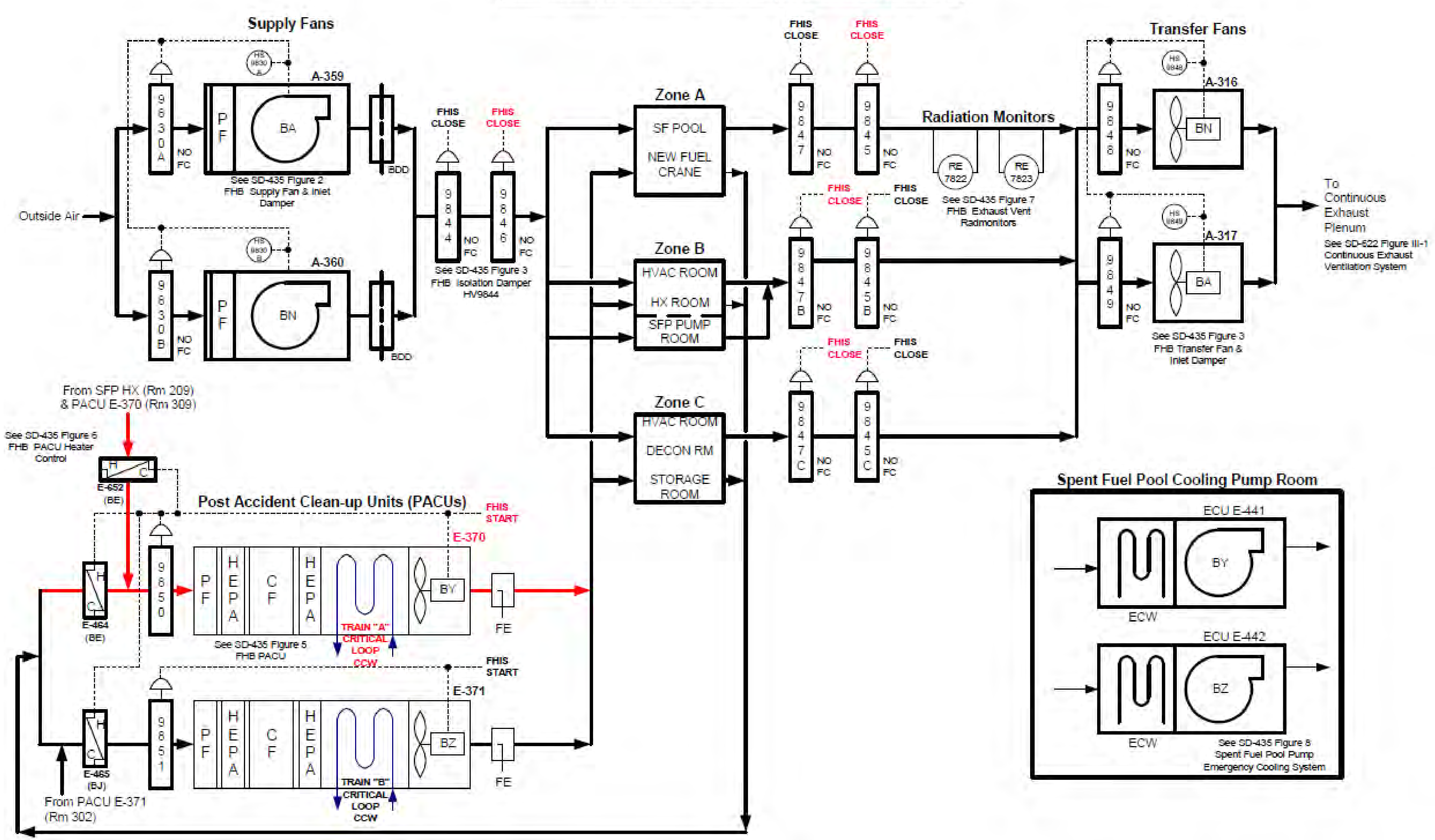
.2.3 The TRAIN separation ensures that a single failure does not prevent the FHB from being isolated upon receipt of a FHIS.

.3 FHB Ventilation must be in service (Supply and Exhaust Fan) for the rad monitors to obtain a representative sample of the FHB atmosphere, otherwise, they are not OPERABLE.

.4 Upon receipt of a Fuel Handling Isolation Signal, the normal Fuel Handling Building supply and exhaust isolation dampers, 2(3)HV-9846, 2(3)HV-9847, 2(3)HV-9847B and 2(3)HV-9847C for Train A and 2(3)HV-9844, 2(3)HV-9845B, 2(3)HV-9545C and 2(3)HV-9845 for Train B will close, which, in turn will stop the Fuel Handling Building Normal Ventilation Supply and Exhaust Fans. The Fuel Handling Isolation Signal will also start the Post-Accident Cleanup Units 2(3)E-370 and 2(3)E-371, and open their corresponding inlet dampers 2(3)HV-9850 and 2(3)HV-9851.

Comments / Reference: From SD-SO23-435, Figure 1	Revision # 3

FIGURE 1: FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	076 A4.02	
Importance Rating	2.6	

Service Water System: Ability to manually operate and/or monitor in the control room: SWS valves

Proposed Question: Common 25

Which ONE (1) of the following describes the operation of the Train A Saltwater Cooling Heat Exchanger Outlet Valve (HV-6497)?

The Train A Saltwater Cooling Heat Exchanger Outlet Valve (HV-6497) ______________ when Saltwater Cooling Pump P-307 is auto started and ______________ when SWC Pump P-307 is stopped.

A. must be manually opened
will automatically close

B. will automatically open
will automatically close

C. will automatically open
must be manually closed

D. must be manually opened
must be manually closed

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the valve could be manually opened upon SWC Pump auto start, however, there is no automatic closure associated with this valve.
B. Incorrect. Plausible because the valve will automatically open upon SWC Pump auto start, however, there is no automatic closure associated with this valve.
C. Correct. The CCW/SWC Heat Exchanger Outlet Valve will automatically open when the SWC Pump is started but must be manually closed to avoid an unacceptable failure mode.
D. Incorrect. Plausible because the valve must be manually closed upon SWC Pump start, however, there is no requirement for the valve to be manually opened.

Technical Reference(s) SO23-2-8, L&S 4.8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 60306 / 60307	INTERPRET instrumentation and controls utilized in the Salt Water Cooling System. ANALYZE normal and abnormal operations of the Salt Water Cooling System.		
Question Source:	Bank #	127289	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: From SO23-2-8, L&S 4.8	Revision # 30

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3.0 BUMPING SWC PUMPS

3.1 When a SWC Pump is uncoupled, then it may be bumped for rotation, or run uncoupled, as required, to support maintenance activities. If the Kirk Key interlock has been defeated to allow the start, then the start button will need to be held for five seconds for the pump to start. The SWC Train remains operable during this evolution. (AR 021000323)

4.0 SWC SYSTEM OPERATION

4.1 If the CCW outlet temperature approaches 95°F, or Saltwater outlet approaches 90°F, then the SRO Operations Supervisor should be notified for an operability assessment.

4.2 When the SWC discharge path to the Circulating Water System is required to be operable, then power to Gate No. 6 shall be removed, or an Operator should be available to manually position Gate No. 6. Otherwise, a single failure in the gate circuitry can cause gate closure and inoperability of both SWC Trains. (Ref. 2.3.9 and UFSAR Section 9.2.1.3.C)

4.3 Indication of blockage to the SWC Discharge to Outfall:

- Saltwater Cooling discharge pressure > 22.5 psig
- CCW HX E-001(E-002) Differential Pressure < 3 psid
- Saltwater Cooling Flow drops below 12,000 gpm

4.4 If SWC Pump flow will be < 3500 gpm (i.e., in support of testing), then Maintenance Engineering should be notified of the low flow operating condition. (Ref. 2.3.8)

4.5 If a CCW Heat Exchanger Auto Vent valve is isolated while the HX is in Service or in Standby (e.g., CCW HX Auto Vent Valve has failed Open), then the saltwater side of the CCW Heat Exchanger should be manually vented on a daily basis until the Auto Vent Valve is returned to Service.

4.6 If a start of the Standby CCW/SWC Loop occurs, and the Auto Vent Valve on the saltwater side of the CCW Heat Exchanger is isolated, then a manual vent of the CCW HX should be performed.

4.7 If CCW Heat Exchanger heat transfer efficiency is questionable (i.e., CCW outlet temperature is trending up), then this may indicate air fouling of the CCW HX Saltwater side due to a failed closed CCW HX Auto Vent Valve.

4.8 The SWC Heat Exchanger Outlet Valves (HV6495/HV6497) will automatically open on start of SWC Pump. However, they do not automatically close as this is an unacceptable failure mode.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 G 2.1.25	
	Importance Rating	3.9	

Service Water System: Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: Common 26

Given the following conditions:

- Unit 2 has been in MODE 1 for 7 days following a 40 day Refueling Outage.
- Spent Fuel Pool temperature is 92°F.
- Spent Fuel Pool level is 26 feet 4 inches.
- Component Cooling Water/Salt Water Cooling (CCW/SWC) Heat Exchanger E001 differential pressure is 10 psid.
- Salt Water Cooling injection temperature is 70°F.

Referring to SO23-2-8, Salt Water Cooling System Operation, Attachment 4, which ONE (1) of the following is the minimum required flow of CCW/SWC Heat Exchanger E001?

Minimum required flow is approximately...

A. 15,900 gpm.

B. 16,800 gpm.

C. 17,200 gpm.

D. 17,900 gpm.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because with the Normal Curve and 9 psid is referenced, the required resultant flow would be 16,600 gpm.

B. Incorrect. Plausible because if the wrong (reverse flow) Normal Curve is used instead of the proper Normal Curve, the required resultant flow would be 16,800 gpm.

C. Correct. Due to the time after the beginning of the Refueling Outage being greater than 45 days, the proper curve to use is the Normal Curve for normal system alignment (not reverse flowing). With 10 psid, the curve arrives at 17,200 gpm.

D. Incorrect. Plausible because if the Alternate Curve is used the required resultant flow would be 17,900 gpm.

Technical Reference(s)	SO23-2-8, Attachment 4	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: SO23-2-8, Attachment 4

Learning Objective: 60305 / 60307	DESCRIBE the configuration and operational characteristics of Salt Water Cooling System components. ANALYZE normal and abnormal operations of the Salt Water Cooling System.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	5, 10	
	55.43		

Comments / Reference: From SO23-2-8, Attachment 4	Revision # 30

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2.0 PROCEDURE (continued)

2.2 Determine which curve to use in Modes 1-4

2.2.1 If *all* of the following are true, then use the applicable Normal Curve:

- Spent Fuel Pool level is ≥ 26'
- Spent Fuel Pool Temperature is ≤ 95°F
- Time elapsed since the start of the last refueling outage is ≥ 45 days

2.2.2 If *any* of the following are true, then use the applicable Alternate Curve:

- Spent Fuel Pool level is < 26'
- Spent Fuel Pool Temperature is > 95°F
- Time elapsed since the start of the last refueling outage is < 45 days

2.3 Determine which curve to use in Modes 5-6

2.3.1 Use the applicable Normal Curve. (There are no restrictions in Modes 5 and 6.)

Comments / Reference: From SO23-2-8, Attachment 4	Revision # 30

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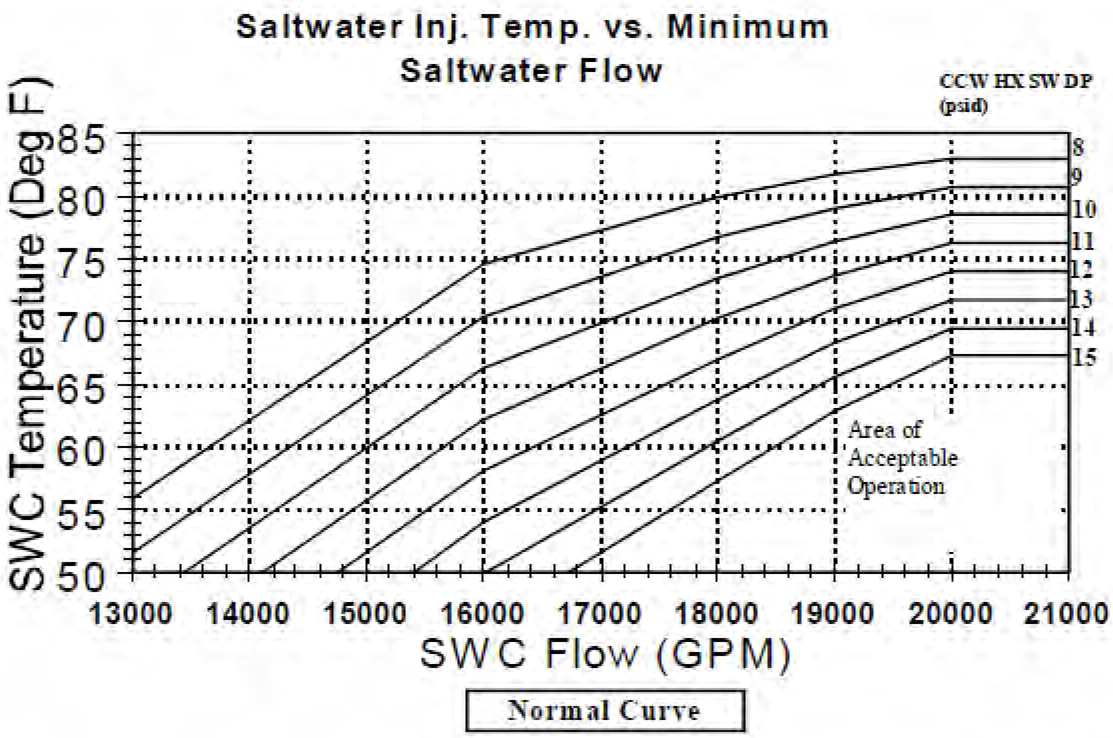
NORMAL FLOW OPERATIONS

(REFER TO STEPS 2.1.2.2 AND 2.1.2.3 FOR REQUIRED TEMPERATURE ADJUSTMENT)



Comments / Reference: From SO23-2-8, Attachment 4	Revision # 30

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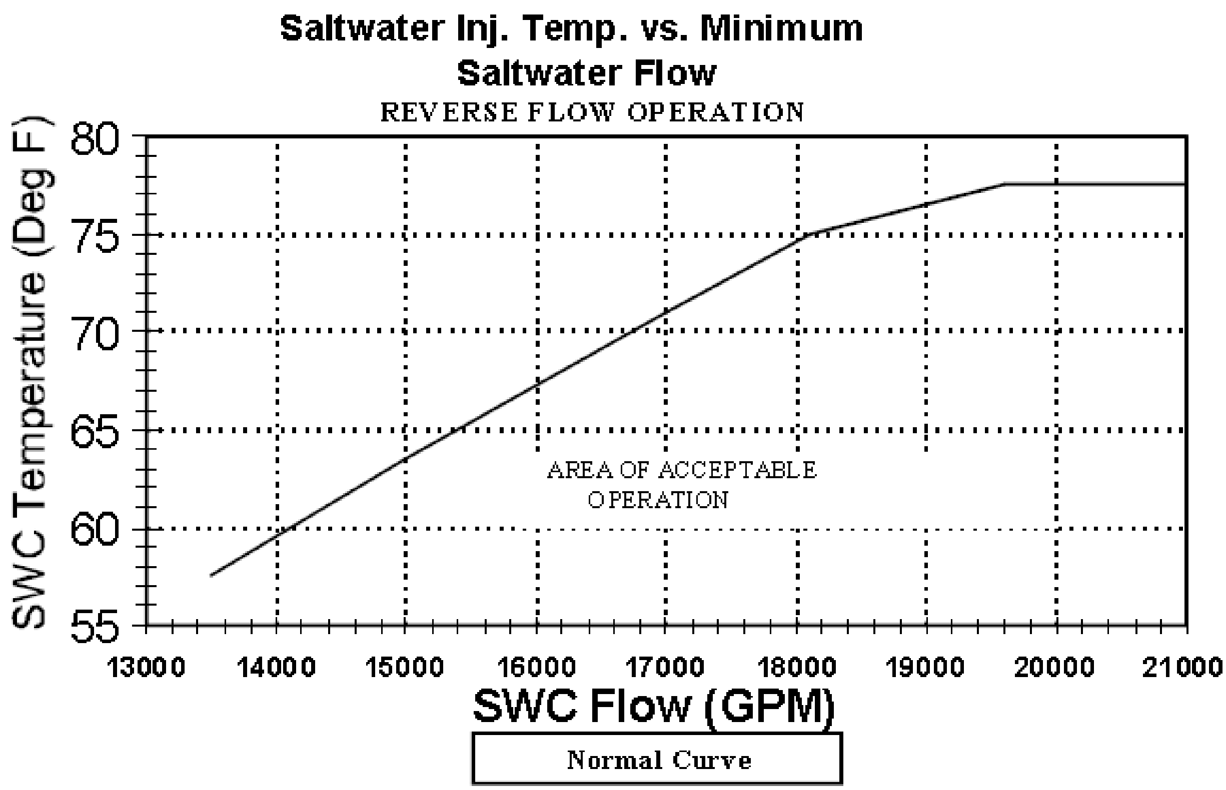
NORMAL REVERSE FLOW OPERATIONS

(REFER TO STEPS 2.1.2.2 AND 2.1.2.3 FOR REQUIRED TEMPERATURE ADJUSTMEN



Comments / Reference: From SO23-2-8, Attachment 4	Revision # 30

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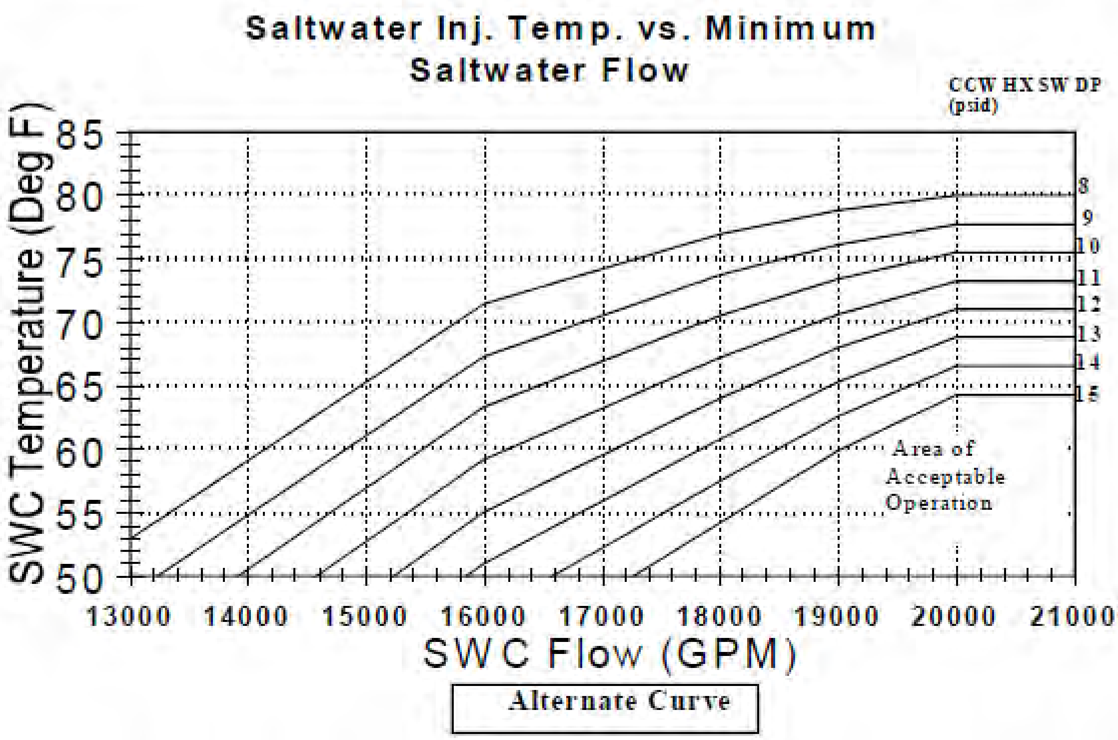
ALTERNATE FLOW OPERATIONS

(REFER TO STEPS 2.1.2.2 AND 2.1.2.3 FOR REQUIRED TEMPERATURE ADJUSTMENT)



Comments / Reference: From SO23-2-8, Attachment 4	Revision # 30

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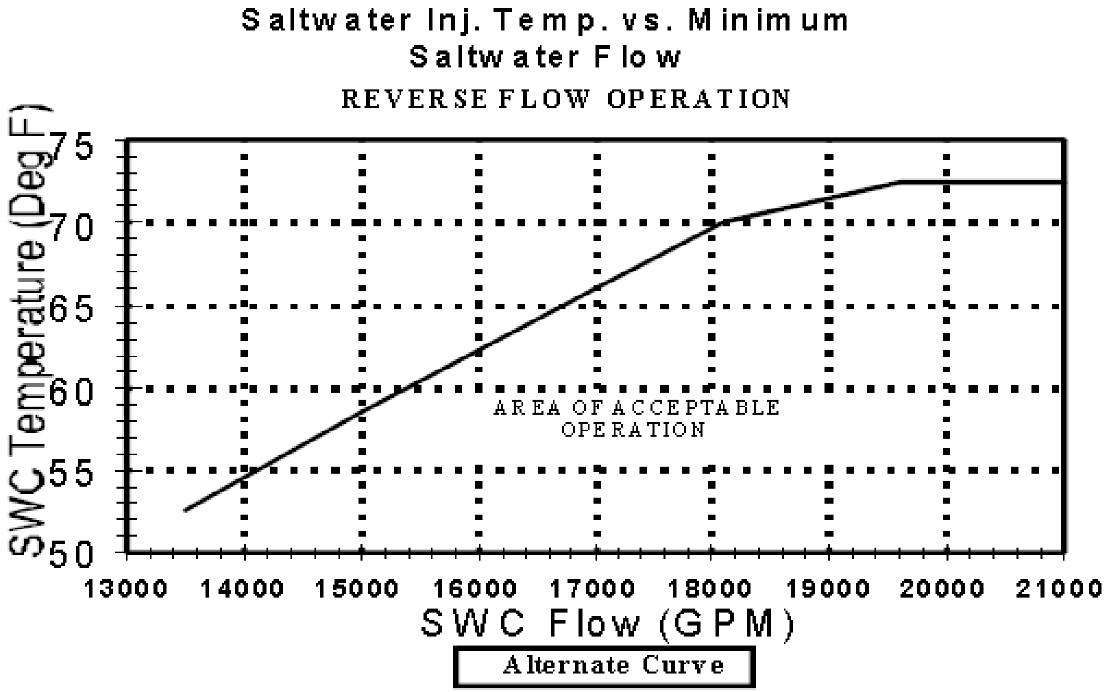
ALTERNATE REVERSE FLOW OPERATIONS

(REFER TO STEPS 2.1.2.2 AND 2.1.2.3 FOR REQUIRED TEMPERATURE ADJUSTMENT



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 K1.02	
Importance Rating	2.7	

Instrument Air System: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service air

Proposed Question: Common 27

Given the following conditions:

- Unit 2 is operating at 100% power with the Instrument Air system aligned for normal operations.
- Subsequently, an air leak upstream of the Instrument Air Dryers occurs.
- Instrument Air header pressure lowered to 85 psig and is currently 87 psig and steady.

Which ONE (1) of the following describes the current status of the Instrument Air System?

A. Three (3) Instrument Air Compressors are running fully loaded.
Service Air is maintaining Instrument Air via a pressure control valve.

B. Two (2) Instrument Air Compressors are running fully loaded.
Nitrogen Backup is maintaining Instrument Air via a pressure control valve.

C. One (1) Instrument Air Compressor is running fully loaded.
Two (2) Instrument Air Compressors are running half loaded.
Nitrogen Backup and Service Air is maintaining Instrument Air via pressure control valves.

D. Two (2) Instrument Air Compressors are running fully loaded.
One (1) Instrument Air Compressor is running half loaded.
Nitrogen Backup and Service Air is maintaining Instrument Air via pressure control valves and Instrument Air to Containment has closed.

Proposed Answer: A

Explanation:

A. Correct. Lowering of Instrument Air header pressure to 85 psig caused all three Instrument Air Compressors to run fully loaded. The Service Air Pressure Control Valve opens at 88 psig.

B. Incorrect. Plausible because at least two Instrument Air Compressors are running fully loaded, however, the Instrument Air header would have to drop to 83 psig for Nitrogen Backup to actuate.

C. Incorrect. Plausible because three Instrument Air Compressors are running, however, they would all be at full load. Nitrogen Backup would not be in service because Instrument Air Header pressure did not drop low enough.

D. Incorrect. Plausible because three Instrument Air Compressors are running, however, they would all be at full load. Nitrogen Backup would not be in service because Instrument Air Header pressure did not drop low enough. Instrument Air to Containment is isolated for a different reason.

Technical Reference(s): SO23-13-5, Attachment 6, L&S 1.2; SO23-1-1, Attachment 22, L&S 2.1; SO2-15-61.C, 61C19 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 72865 / 72866

DESCRIBE the configuration and operational characteristics of Instrument and Respiratory & Service Air Systems components.

INTERPRET instrumentation and controls utilized in the Instrument and Respiratory & Service Air Systems.

Question Source:
- Bank #
- Modified Bank # 127082 (Note changes or attach parent)
- New

Question History: Last NRC Exam SONGS 2005A

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

10 CFR Part 55 Content:
- 55.41 7, 10
- 55.43

Comments / Reference: From SO23-13-5, Attachment 6, L&S 2.1	Revision # 7

NUCLEAR ORGANIZATION
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ABNORMAL OPERATING INSTRUCTION
REVISION 7
ATTACHMENT 6

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LOSS OF INSTRUMENT AIR LIMITATIONS AND SPECIFICS

OBJECTIVE

To increase understanding of actions when responding to and mitigating an Instrument Air event caused by a large leak. Since an Instrument Air leak requires prompt action, targeted L&Ss are not used, to prevent unnecessary distractions. (ACE 070900675-1)

1.0 General Information

1.1 A loss of Instrument Air due to a leak involves both Units, but in very different ways. The Unit that has the active leak is called the ***affected Unit***. The Unit without the active leak is called the ***unaffected Unit***. This convention is used to distinguish between the Unit which may have serious problems due to the leak from the Unit that should not see any serious effects. While in reality both Units are affected, this distinction accounts for the fact that the ***unaffected Unit*** will be relying on backup Nitrogen.

1.2 The Instrument Air Compressors cannot maintain header pressure when there is ≥ 1.5 inch hole in the system (e.g., line break, joint separation, aggregate of smaller leaks, etc.). That size leak will lower header pressure enough to start the backup RSAS Air Compressors at 88 psig and supply the Instrument Air header. However, pressure will continue to fall to 83 psig and backup Low Pressure Nitrogen will begin supplying the header. The Nitrogen header splits into two headers, one supplying each Unit (see Attachment 5). Near this split, each of the two headers has an excess flow check valve. As pressure drops below 83 psig, the excess flow check valve to the ***affected Unit*** will close, isolating that Unit from Nitrogen. Backup Nitrogen header pressure will then recover to 83 psig, and the ***unaffected Unit*** will be supplied by backup Nitrogen, terminating the effects of the event for that Unit.

Comments / Reference: From SO23-1-1, Attachment 22, L&S 2.1	Revision # 19

NUCLEAR ORGANIZATION
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INSTRUMENT AIR LIMITATIONS AND SPECIFICS (Continued)

2.0 NORMAL OPERATION OF THE AIR COMPRESSORS

2.1 The Instrument Air Compressors are started and loaded in the sequence selected by the Operator on local control panel 2/3L-102:

PANEL 2/3L-102 CONTROL SWITCHES		PSL			SET POINT
		C-001	C-002	C-003	
LEAD	50% loaded 100% loaded	PSL-5348A	PSL-5350A	PSL-5352A	106 to 110 psig 102 to 106 psig
LAG 1	50% loaded 100% loaded	PSL-5348B	PSL-5350B	PSL-5352B	98 to 102 psig 94 to 98 psig
LAG 2	50% loaded 100% loaded	PSL-5348C	PSL-5350C	PSL-5352C	90 to 94 psig 86 to 90 psig
MANUAL	50% loaded 100% loaded	PSL-5348B	PSL-5350B	PSL-5352B	98 to 102 psig 94 to 98 psig
Each Instrument Air Compressor is rated for 400 scfm Half-Load and 800 scfm Full Load.					

Comments / Reference: From SO2-15-61.C, 61C19	Revision # 6

NUCLEAR ORGANIZATION
UNIT 2

ALARM RESPONSE INSTRUCTION
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61C19 INST AIR HEADER PRESS LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	NO	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2
2PSL-5342	Instrument Air Header Pressure Switch Low	75 PSIG	2PI-5344A	NONE	1638
2PSL-5378	Instrument Air Header Pressure Switch Low	75 PSIG			

1.0 **REQUIRED ACTIONS**:

1.1 GO TO SO23-13-5, Loss of Instrument Air.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
NONE	NONE

3.0 ASSOCIATED RESPONSES:

NONE

4.0 COMPENSATORY ACTIONS:

DEVICE NUMBER	SPECIFIC COMPENSATORY ACTIONS
4.1 2PI-5344A, Instrument Air Header Pressure Indicator	4.1 Monitor Instrument Air Header pressure at least once per shift.

Comments / Reference: From SONGS Exam Bank #127082	Revision 10/20/06

Given the following conditions:

- Unit 2 is operating at 100% power with the Instrument Air system aligned for normal operations.
- Subsequently, a valid Instrument Air Header Pressure Low alarm is received, due to an air leak.
- Instrument Air header pressure is currently 87 psig and steady.

Which ONE (1) of the following correctly describes the current status of the Instrument Air System?

A. Three Instrument Air Compressors are running fully loaded.
Nitrogen backup is maintaining Instrument Air via a pressure control valve.

B. Three Instrument Air Compressors are running fully loaded.
Service Air is maintaining Instrument Air via a pressure control valve.

C. Two Instrument Air Compressors are running fully loaded.
One Instrument Air Compressor is running half loaded.
Nitrogen backup is maintaining Instrument Air via a pressure control valve.

**D. Two Instrument Air Compressors are running fully loaded.
One Instrument Air Compressor is running half loaded.
Service Air is maintaining Instrument Air via a pressure control valve.**

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	103 A3.01	
Importance Rating	3.9	

Containment System: Ability to monitor automatic operation of the containment system, including: Containment isolation

Proposed Question: Common 28

Which ONE (1) of the following valves receives a CLOSE signal on a Containment Isolation Actuation Signal?

A. Steam Generator Blowdown Isolation Valves.

B. Containment Sump Pump Discharge Isolation Valves.

C. Main Steam Valves to P140, Auxiliary Feedwater Pump.

D. Blended Makeup to Volume Control Tank Isolation Valves.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because these valves could allow contamination to escape Containment during a tube rupture, however, these valves are closed on an MSIS.
B. Correct. These valves are isolated on a CIAS.
C. Incorrect. Plausible because the Main Steam Isolation Valves are closed on a CIAS, however, these valves are not closed until an MSIS is received.
D. Incorrect. Plausible because this interfaces with Letdown, however, it is closed on an SIAS.

Technical Reference(s) SO23-3-2.22, Attachments 4, 11, and 17 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 79743 DESCRIBE the flowpaths and major components of the CIS, CSS, and SIS utilized for normal and emergency operations.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7, 9
55.43

Comments / Reference: From SO23-3-2.22, Attachment 17	Revision # 16

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2.0 PROCEDURE (Continued)

2.3 **VERIFY CIAS Train B component actuation at CR-57**: (Continued)

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.3.17	HV-7806	Containment Rad Mon Tr B Inlet Isolation	[1][2]	CLOSED	______
2.3.18	HV-9900	Containment Chill Water Inlet Isolation	[1][2]	CLOSED	______
2.3.19	HV-5686	Containment Fire Water Isolation	[1][2]	CLOSED	______
2.3.20	HV-7513	RCDT T-012 Drain Isolation	[1][2]	CLOSED	______
2.3.21	HV-7258	Containment Waste Gas Vent Header Isolation Valve	[1][2]	CLOSED	______
2.3.22	HV-5804	Containment Sump Pump Discharge Isolation	[1][2]	CLOSED	______
2.3.23	HV-7800	Containment Rad Mon Tr A Inlet Isolation	[1][2]	CLOSED	______
2.3.24	HV-7805	Containment Rad Mon Tr B Outlet Isolation	[1][2]	CLOSED	______
2.3.25	HV-9971	Containment Normal Chilled Water Isolation	[1][2]	CLOSED	______
2.3.26	HV-9821	Containment Mini Purge Supply Isolation		CLOSED	______
2.3.27	HV-9824	Containment Mini Purge Exhaust Isolation		CLOSED	______
2.3.28	HV-5434	SIT Nitrogen Supply Isolation	[1][2]	CLOSED	______

Comments / Reference: From SO23-3-2.22, Attachment 11	Revision # 16

NUCLEAR ORGANIZATION
UNITS 2 AND 3

OPERATING INSTRUCTION
REVISION 16
ATTACHMENT 11

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2.0 PROCEDURE (Continued)

2.2 ENSURE MSIS Train A/B component actuation on CR-52:

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.2.1	HV-4048	S/G E-088 Main Feedwater Isolation Valve		CLOSED	________
2.2.2	HV-8205	S/G E-088 Main Steam Isolation Valve		CLOSED	________
2.2.3	HV-4052	S/G E-089 Main Feedwater Isolation Valve		CLOSED	________
2.2.4	HV-8204	S/G E-089 Main Steam Isolation Valve		CLOSED	________

2.3 ENSURE MSIS Train A component actuation on CR-52:

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.3.1	HV-8419	S/G E-088 Atmospheric Dump Valve	[1]	CLOSED	________
2.3.2	HV-8203	MSIV HV-8205 Bypass Valve	[1]	CLOSED	________
2.3.3	HV-4054	S/G E-088 Blowdown Isolation Valve	[1]	CLOSED	________
2.3.4	HV-4058	S/G E-088 Water Sample Isolation Valve	[1]	CLOSED	________
2.3.5	HV-8201	Main Steam to P-140 Isolation	[1][2]	CLOSED	________

Comments / Reference: From SO23-3-2.22, Attachment 4	Revision # 16

NUCLEAR ORGANIZATION
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2.0 PROCEDURE (Continued)

2.4 **VERIFY SIAS/CCAS Train A component actuation at CR-57**: (Continued)

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.5.37	HV-6202	P-307 Discharge Valve (N/A if P-112 running)		OPEN	________
2.5.38	HV-6378	P-307 Bearing Water Supply Valve (N/A if P-112 running)		OPEN	________
2.5.39	TV-0221	L/D to Regen HX E-063 Isolation	[2]	CLOSED	________
2.5.40	HV-9205	Regen to L/D HX Isolation	[2][3]	CLOSED	________
2.5.41	HV-9236	P-174 to T-071 Recirc.		CLOSED	________
2.5.42	HV-9231	P-175 to T-072 Recirc.		CLOSED	________
2.5.43	FV-9253	Blended Makeup to VCT Isolation		CLOSED	________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002 K5.01	
	Importance Rating	3.2	

Reactor Coolant System: Knowledge of the operational implications of the following concepts as they apply to the RCS: Basic heat transfer concepts

Proposed Question: Common 29

Given the following condition:

- During an outage, chemical cleaning is performed on Steam Generator tubes, and no tubes in either Steam Generator are plugged.

Given the same 100% Reactor power level before and after the outage, which ONE (1) of the following will be observed?

A. Turbine Governor Valves will be more OPEN than before the outage.

B. Turbine Governor Valves will be more CLOSED than before the outage.

C. Steam Generator ΔT will be LARGER than before the outage.

D. Steam Generator ΔT will be the SAME AS before the outage.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that cleaning of the Steam Generator tubes caused the U in $Q = UA\Delta T$ to increase. In this case, the ΔT will be smaller and the Turbine Governor Valves would have to be opened further for the same power level.

B. Correct. Given the conditions listed, the Turbine Governor Valves will be more closed than before the outage at 100% power.

C. Incorrect. Plausible given the equation $Q = UA\Delta T$. If U decreases then ΔT will be larger, however, cleaning the tubes causes U to increase and the ΔT will decrease.

D. Incorrect. Plausible given the equation $Q = UA\Delta T$. If U increases then A would have to decrease to maintain ΔT constant. Since cleaning the tubes causes U to increase without any associated change in A the ΔT needs to decrease.

Technical Reference(s): LP 0HT122, Page 33 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53321	Given formulas applicable to heat transfer, and selected heat exchanger parameters, CALCULATE and/or PREDICT changes in heat transfer rate and/or heat exchanger inlet/outlet temperatures.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge ______
	Comprehension or Analysis X
10 CFR Part 55 Content:	55.41 5
	55.43 ______

Comments / Reference: From LP 0HT122, Page 33	Revision # 1-2

The value of U decreases as heat exchanger tube surfaces become fouled. This occurs because the effective thermal conductivity of the tubes is decreased and because the effective thickness of the tubes increases due to the foreign material which attaches itself to the tubes. An accepted method to monitor heat exchanger operation is to monitor changes in UA:

$UA = Q / \Delta T$

When the value of UA has decreased below its minimum acceptable (design) value, the heat exchanger is removed from service and cleaned.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	015 A2.05	
Importance Rating	3.3	

Nuclear Instrumentation System: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Core void formation

Proposed Question: Common 30

Given the following conditions:

- A Loss of Coolant Accident is in progress.
- Core Exit Saturation Margin is 10°F superheat.

Which ONE (1) of the following:

1.) Explains the reason for the indicated response(s) of the Source Range Nuclear Instruments if excessive voids are formed in the core due to inadequate core cooling?

2.) What action must be taken to mitigate the situation?

A. 1.) Count rates would LOWER due to more moderation within the core causing LESS fast leakage.
   2.) Raise Steam Generator feeding rate.

B. 1.) Count rates would LOWER due to less moderation within the core causing MORE fast leakage.
   2.) Open Pressurizer Vent Valves to restore Pressurizer level so that heaters can be energized.

C. 1.) Count rates would RISE due to less moderation within the core causing MORE fast leakage.
   2.) Raise Steam Generator steaming rate.

D. 1.) Count rates would RISE due to more moderation within the core inserting positive reactivity to increase the shutdown power level.
   2.) Lower Reactor Coolant System pressure to inject the Safety Injection Tanks.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the action to take is correct, however, voiding causes a reduction in moderation and an increase in fast neutron leakage.
B. Incorrect. Plausible because most of the reason for voiding is correct, however, this action would only be taken if an excessive Core Exit Saturation Margin existed.
C. Correct. This is the correct reason for voiding and the correct action to perform per Attachment 5.
D. Incorrect. Plausible because this action is performed during an extended Station Blackout and count rate does rise, however, it is due to less moderation and more fast neutron leakage.

Technical Reference(s)	SO23-12-11, FS-10	Attached w/ Revision # See Comments / Reference
	SO23-12-11, Attachment 5	
	LP 2LC750, Page 44	

Proposed references to be provided during examination: None

Learning Objective: 53354	ASSESS the response of the following instrumentation to changes in reactor vessel water level during degraded core conditions: Excore Nuclear Instrumentation.

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

Question History:	Last NRC Exam	

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

10 CFR Part 55 Content:	55.41	5, 10
	55.43	

Comments / Reference: From SO23-12-11, FS-10	Revision # 6

NUCLEAR ORGANIZATION
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ATTACHMENT 2

EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
FS-10 ELIMINATE Voids (Continued)	
g. RE-EVALUATE RCS voiding per FS-9, VERIFY RCS Free of Voids.	g. 1) REQUEST Shift Manager/Operations Leader to evaluate opening Reactor Vessel Head Vent or PZR Vent to remove non-condensable gases per *SO23-3-2.33, REACTOR COOLANT GAS VENT SYSTEM.* 2) GO TO step a.
h. RESTORE Core Exit Saturation Margin to optimum range per Attachment 5, CORE EXIT SATURATION MARGIN CONTROL.	
i. REQUEST Shift Manager/Operations Leader to evaluate restoring normal Letdown.	

Comments / Reference: From SO23-12-11, Attachment 5	Revision # 6

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ATTACHMENT 5

EOI SUPPORTING ATTACHMENTS

CORE EXIT SATURATION MARGIN CONTROL

NOTE

During ESDE the value of PTS Subcooling (CFMS page 311) should be used in place of CESM.

CAUTION

Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS is the controlling attachment when: 1) the Natural Circulation cooldown strategy of minimizing Reactor Vessel Upper Head void formation is used, or 2) the EOIs are entered from a lower mode and Shutdown Cooling was NOT initially in service.

<table>
<tr><th colspan="4">CORE EXIT SATURATION MARGIN (CESM)</th></tr>
<tr><td>CONTROL METHOD</td><td>LOCA,SGTR,SBO,FR: less than 20°F
ESDE,LOFW, LOOP/LOFC: less than 80°F
LOWER MODE ENTRY: less than 20°F</td><td><u>OPTIMUM</u>
SBO: between 20°F and 50°F
LOCA,SGTR,FR: between 20°F and 160°F
ESDE,LOFW, LOOP/LOFC: between 80°F and 160°F
LOWER MODE ENTRY: greater than 20°F (No Upper Limit)</td><td>SBO: greater than 50°F
OTHER: greater than 160°F
LOWER MODE ENTRY: No Upper Limit</td></tr>
<tr><td>Feedwater Flowrate</td><td>RAISE,
MAINTAIN S/G Levels
– less than 80% NR.</td><td>STABILIZE S/G Level
– between 40% and 80% NR.</td><td>LOWER,
MAINTAIN S/G Levels
– greater than 40% NR</td></tr>
<tr><td>S/G Steaming Rate</td><td>RAISE</td><td>MAINTAIN</td><td>LOWER</td></tr>
<tr><td>SI Flowrate</td><td rowspan="2">RAISE,
ATTEMPT to maintain PZR level
– less than 60%.</td><td rowspan="2">IF SI throttle/stop criteria (FS-7) – satisfied,
THEN throttle flowrate to maintain PZR level
– between 30% and 60%</td><td rowspan="2">IF SI throttle/stop criteria (FS-7) – satisfied,
THEN lower flowrate and maintain PZR level
– greater than 30%.</td></tr>
<tr><td>Charging Flowrate</td></tr>
<tr><td>Letdown Flowrate</td><td>LOWER</td><td>IF SIAS – reset, THEN ATTEMPT to place PLCS in AUTO.</td><td>RAISE</td></tr>
<tr><td>Normal Spray</td><td rowspan="2">LOWER,
ATTEMPT to maintain PZR level
– less than 60%.</td><td rowspan="3">MAINTAIN Saturation Margin as RCS temperatures are reduced.</td><td rowspan="2">RAISE,
REQUEST SM/OL evaluate opening PZR Vents per SO23-3-2.33, REACTOR COOLANT GAS VENT SYSTEM.</td></tr>
<tr><td>Auxiliary Spray</td></tr>
<tr><td>PZR Htrs.</td><td>If PZR level greater than 30%, ENSURE ON.</td><td>ENSURE OFF</td></tr>
</table>

Comments / Reference: From LP 2LC750, Page 44	Revision # 3-3

B. Neutron Detector Response

1. Excore NIS
   a. General System Description
      1) Safety Channels (4)
         a) Use 3 vertically stacked fission chambers.
         b) Are not post-accident qualified.
      2) Startup/Wide Range Channels (2)
         a) Use 2 side by side fission chambers.
         b) Channels are post-accident qualified.
   b. Expected Response of Excore NIs
      1) Similar to response at TMI-2.
      2) Can be used to detect:
         a) Core water level
         b) RCS voiding (bulk boiling) with RCPs running
         c) Core re-arrangement due to severe damage.
      3) Excore response changes due to 3 fundamental causes:
         a) Change in source multiplication
         b) Reduced shielding due to voiding (bulk boiling)
         c) Change in nature or location of neutron sources.
      4) As core and downcomer water level decreases:
         a) Neutron leakage increases due to loss of moderator/reflector.
         b) More neutrons reach detector.
         c) Indicated power level will increase.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	016 A4.01	
	Importance Rating	2.9	

Non-Nuclear Instrumentation System: Ability to manually operate and/or monitor in the control room: NNI channel select controls

Proposed Question: Common 31

Given the following conditions:

- Unit 2 is operating at normal pressure, temperature and level with the Pressurizer Level Controller, LIC-0110 in LOCAL (Operator) Setpoint Control.
- All other Pressurizer level and pressure controls are in Automatic.
- A power descension is in progress.

While in LOCAL (Operator) Setpoint Control, which ONE (1) of the following describes how Pressurizer level is controlled?

Pressurizer level setpoint is...

A. automatically maintained as T_{AVE} changes.

B. automatically maintained as Letdown flow changes.

C. adjusted based on level deviation from setpoint.

D. manually controlled by the Operator.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because T_{AVE} changes will generate a deviation from setpoint, however, only when the controller is in REMOTE.
B. Incorrect. Plausible because as letdown flow changes pressurizer level will change, however, not in the reverse order.
C. Incorrect. Plausible because it could be thought that the deviation signal will prompt the operator to make level setpoint changes, however, SO23-3-10 directs the operator to follow Attachment 5 when in LOCAL (Operator) Setpoint Control.
D. Correct. The operator adjusts the LOCAL setpoint based on SO23-3-10, Attachment 5.

Technical Reference(s) SO23-3-1.10, Attachment 1, Step 1.2.1 / SO23-3-1.10, Attachments 5 and 8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56421 — DESCRIBE the process for performing Pressurizer Pressure and Level Control evolutions, including the consequences of misalignment or misoperations.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments / Reference: From SO23-3-1.10, Attachment 1, Step 1.2.1 — Revision # 21

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1.0 PROCEDURE (Continued)

1.2 Transfer Pressurizer Level Auto Control (Remote - Local)

NOTES

1. During REMOTE level control, the level setpoint is determined by a computer driven TAVG program. During LOCAL level control, the level setpoint is determined by the Operator.
2. On LIC-0110 Page 1, Remote setpoint is indicated as "ESP". Local setpoint is indicated as "OSP".
3. During normal operations, the Local setpoint (OSP) will automatically track with the Remote Setpoint (ESP).

1.2.1 **Transfer from REMOTE to LOCAL setpoint control**

.1 Ensure a Reactivity Brief has been conducted for this activity per SO123-0-A1, Section for Reactivity. ☐

.2 Ensure LIC-0110, PZR Level Controller, is in MANUAL with stable letdown flow. ☐

.3 Transfer LIC-0110 setpoint control to LOCAL by depressing the R/L pushbutton. ☐

.4 Adjust LIC-0110 Local setpoint, OSP, (left column) to match the actual PZR level (middle column). ☐

.5 Ensure letdown and charging flows are approximately equal. ☐

.6 Transfer LIC-0110, PZR Level Controller, back to AUTO. ☐

.7 Initiate maintaining PZR level at the required setpoint as determined by RCS temperature per Attachment 5. ☐

Comments / Reference: From SO23-3-1.10, Attachment 5	Revision # 21

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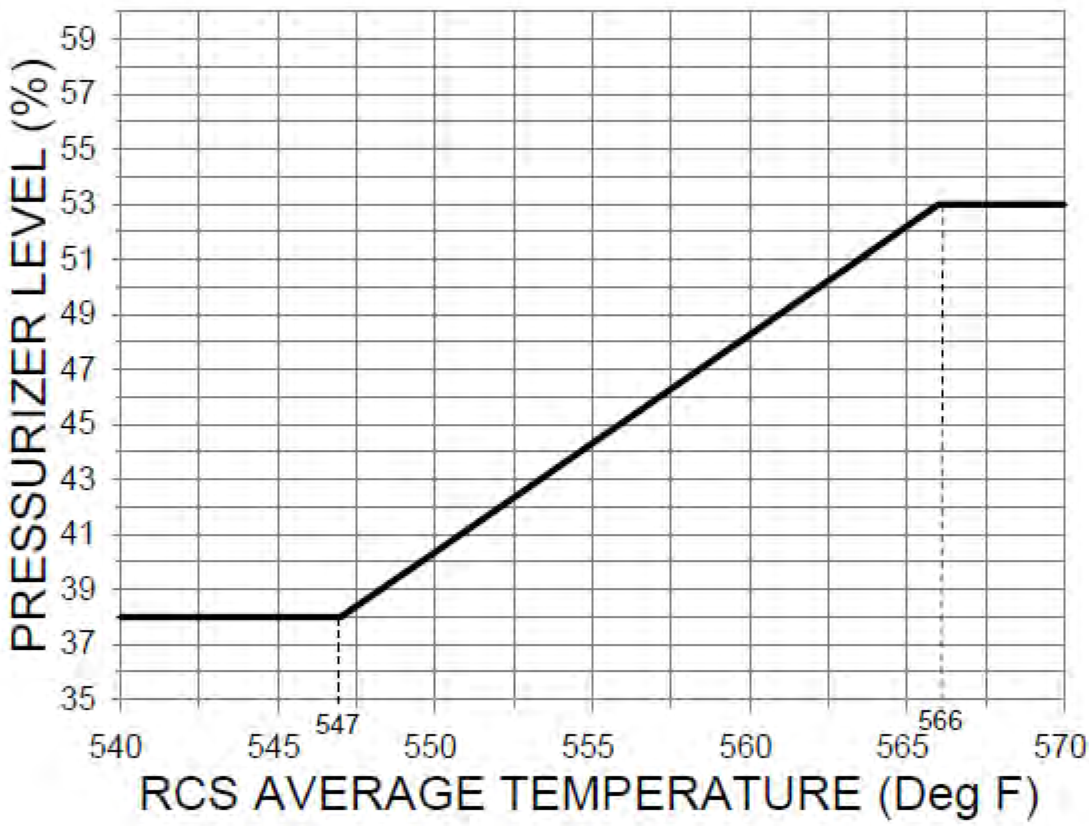
PRESSURIZER LEVEL CONTROL PROGRAM



Comments / Reference: From SO23-3-1.10, Attachment 8	Revision # 21

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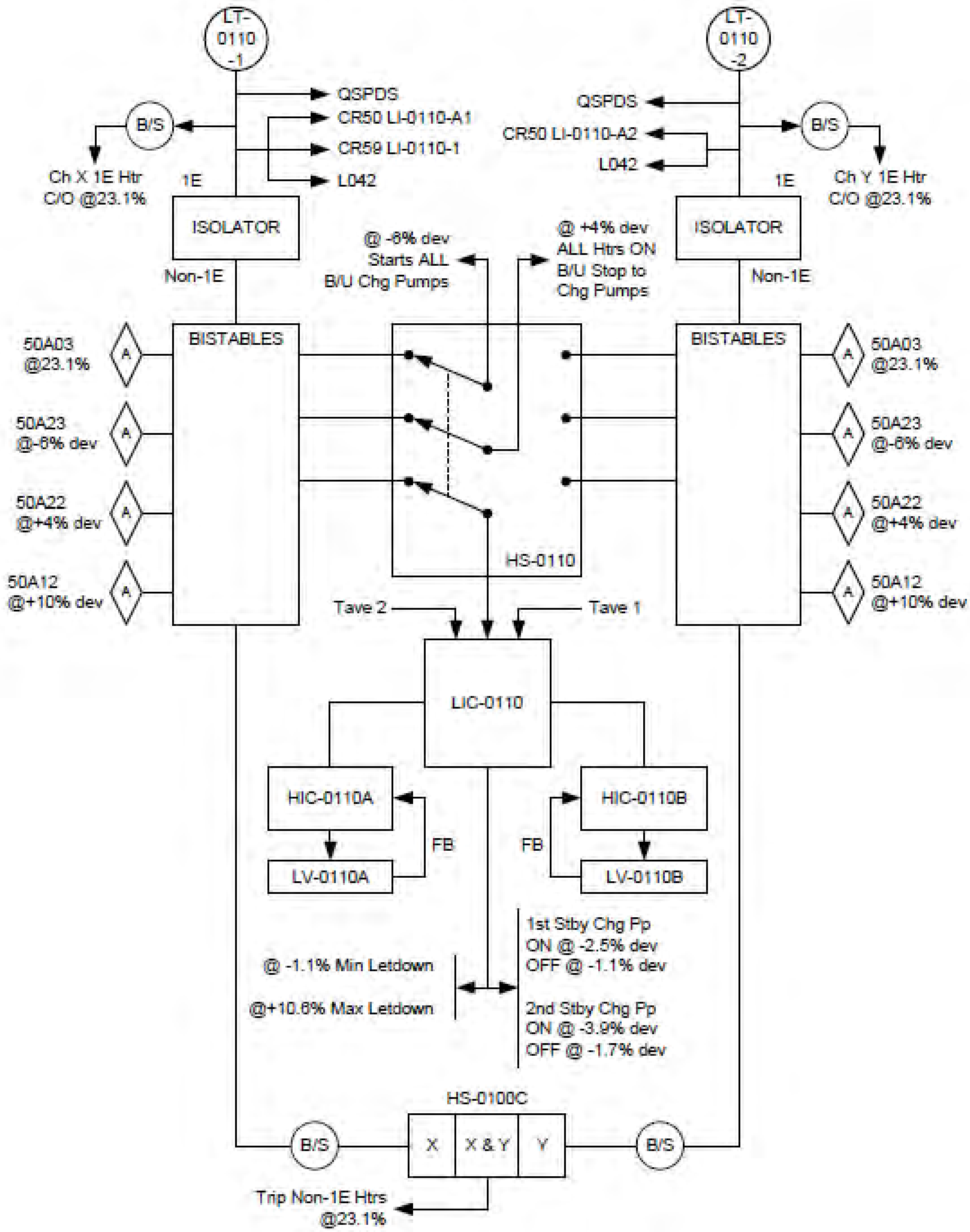
PRESSURIZER CONTROL BLOCK DIAGRAMS (Continued)

PRESSURIZER LEVEL CONTROL SYSTEM BLOCK DIAGRAM



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	029 K3.01	
Importance Rating	2.9	

Containment Purge System: Knowledge of the effect that a loss or malfunction of the Containment Purge System will have on the following: Containment parameters

Proposed Question: Common 32

Given the following conditions:

- The Unit is in MODE 4 with a cooldown to MODE 5 in progress.
- A Containment Mini-Purge is in progress.
- A Containment Mini-Purge Exhaust Isolation Valve fails closed.
- NO other Containment Mini-Purge components have repositioned.
- Current weather conditions are 75°F.

Which ONE (1) of the following Containment parameter changes can be observed within five (5) minutes if no operator action is taken?

A. Humidity level rises.

B. Temperature rises.

C. Radiation level rises.

D. Pressure rises.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because with exhaust isolated, humid air is still being added to Containment, but at a continuously reducing rate as pressure in Containment builds. Normal Containment Cooling would be reducing humidity.

B. Incorrect. Plausible because the flow of air being added to Containment via the Mini-Purge Supply is reduced, however, there is no forced cooling of this air and Containment Normal Cooling is still in service. Also, there is an RCS cooldown in progress.

C. Incorrect. Plausible because Mini-Purge Exhaust flow is reducing airborne radionuclide concentration and that flow has just been reduced. Over time normal RCS leakage would add radioactivity to the Containment atmosphere but this would not be observable in such a short time span.

D. Correct. With Mini-Purge Exhaust isolated and Mini-Purge Supply still in service, air volume is being added to Containment and Containment pressure would slowly but steadily rise.

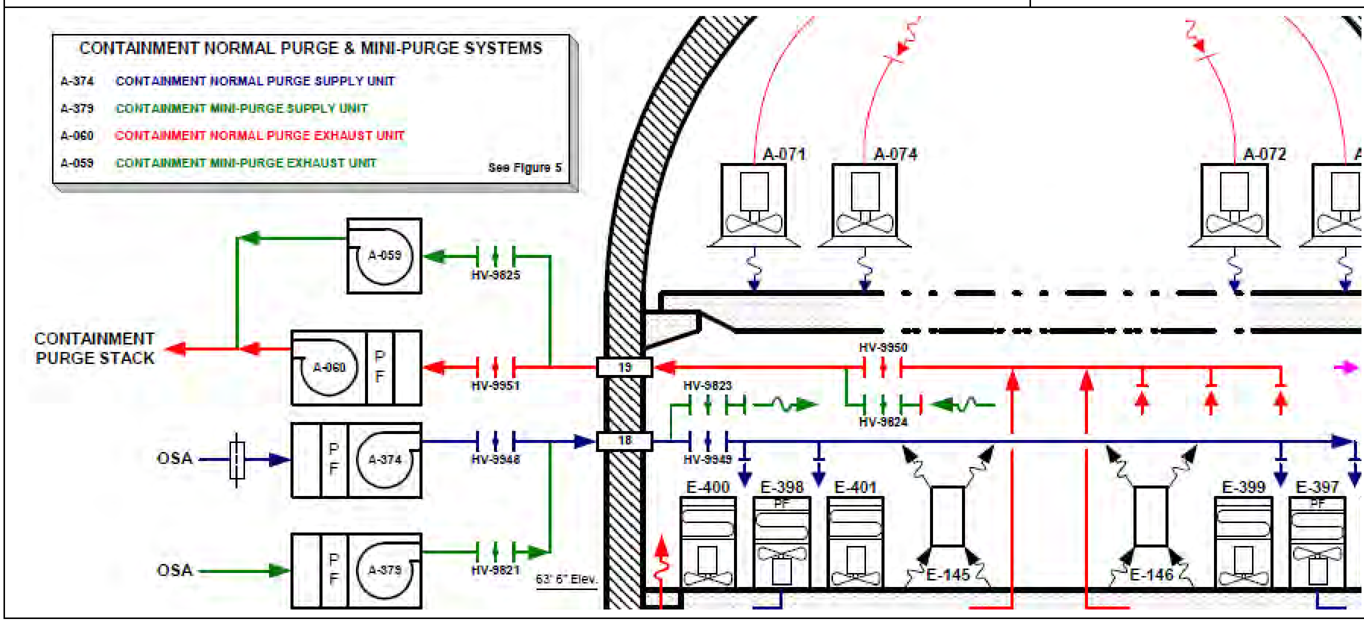
Technical Reference(s)	SD-SO23-770, Figure 1 and 5	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

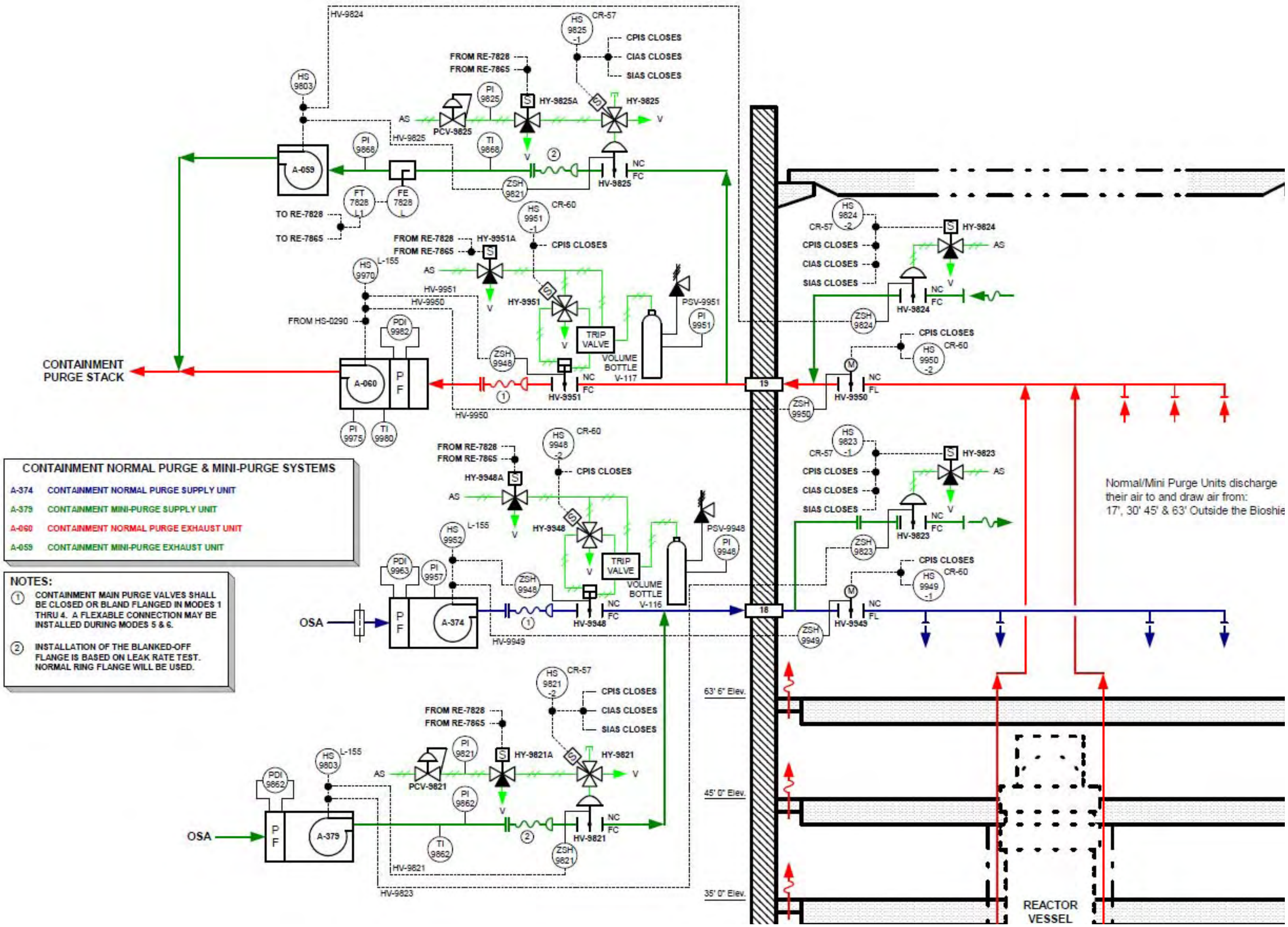
Learning Objective: 79747	STATE the names of the systems interfacing with the CIS, CSS, and SIS and DESCRIBE the flowpath and purpose of each interconnection.

Question Source:	Bank #	127145	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2005B	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	9	
	55.43		

Comments / Reference: From SD-SO23-770, Figure 1	Revision # 8



Comments / Reference: From SD-SO23-770, Figure 5	Revision # 8



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	035 K6.03	
Importance Rating	2.6	

Steam Generator System: Knowledge of the effect of a loss or malfunction on the following will have on the SGs: Steam generator level detector

Proposed Question: Common 33

Given the following condition:

- Unit 2 is at 100% power with Main Feedwater controls in Automatic.

Which ONE (1) of the following describes the impact on Steam Generator E088 level control if Narrow Range Level Transmitter LT-1123-1 fails off-scale high?

Steam Generator E088 level control...

A. raises mean level signal value to the Feedwater Control System causing level to lower until operator manual actions are taken.

B. automatically excludes the failed channel from the mean average on poor quality and controls on the remaining good channels average.

C. automatically transfers to MANUAL control if deviation is greater than 7% narrow range.

D. generates a High Level Override signal to close the Feedwater Control Valves to E088.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that the FWCS would continue to average the signals and only provide alarms to alert the operator.
B. Correct. Three channels are compared and a poor quality signal would be discarded automatically from the average.
C. Incorrect. Plausible because it could be thought that a single channel failure would shift the FWCS to MANUAL, however, for this to occur there must be more than one failed channel.
D. Incorrect. Plausible because it could be thought that a single channel failure would result in generation of a HLO signal, however, it must be actual level to generate the HLO signal.

Technical Reference(s) SD-SO23-250, Pages 59 & 73 Attached w/ Revision # See Comments / Reference
SO23-9-6, L&S 4.5

Proposed references to be provided during examination: None

Learning Objective: 52830 DESCRIBE the instrumentation used to monitor the operation of the Feedwater Control System, including the name, function, sensing points, normal values for the parameters being measured, and the location of each instrument.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: From SD-SO23-250, Page 59	Revision # 14

PART III FEEDWATER CONTROL SYSTEM

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

2.1.1 **General Control Scheme** (Continued) (Figure III-1)

.5 The DFWCS Input Algorithms use 4 field inputs (e.g., 1E S/G N/R Level) or 2 field inputs (e.g., FW Temp).

.5.1 The Signal Validation Circuit can only accommodate

.5.1.1 3 inputs for validation for any 4-field input validations. This requires 1 input to be in BYPASS and the other 3 go through a median sheet.

.5.1.2 2 inputs for validation use a "mean average".

.5.1.3 If there is only one input (for any reason), shifts the validation to either the "last know good" or to MANUAL.

Comments / Reference: From SD-SO23-250, Page 65	Revision # 14

PART III FEEDWATER CONTROL SYSTEM

2.0 DESCRIPTION (Continued)

2.3 Detailed Control Scheme

2.3.1 The DFWCS has an input for each S/G from: (Figure III-1)

.1 **Narrow Range Steam Generator Level**

.1.1 Main level input for control – used in Low Power and High Power modes

.1.1.1 E089 LT1113-1, -2, -3, -4

.1.1.2 E088 LT1123-1, -2, -3, -4

.1.2 Signal quality is a range check of 4 to 20 ma. It looks for only a gross failure.

.1.2.1 See Section III-2.1.1.5 for detailed explanation of signal validation and fault.

.2 **Wide Range Steam Generator Level**

.2.1 Used in Low Power mode to bias Proportional Integral Derivative (PID) Output

.2.1.1 E089 LT1115-1, -2

.2.1.2 E088 LT1125-1, -2

.2.1.3 Average value is used for each S/G (if 2 good signals)

.2.2 Rate of change & deviation **or** Bad Quality will cause a Channel Failure.

.2.2.1 On a channel failure, the system will use the remaining good signal.

.2.3 Deviation between the signals will cause the "last known good" value of signal be used, with different gains and reset times used in PI controller for low power control.

Comments / Reference: From SO23-9-6, L&S 4.5	Revision # 12

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ATTACHMENT 12

FEEDWATER CONTROL SYSTEM LIMITATIONS AND SPECIFICS (Continued)

4.5 The Digital FWCS has multiple inputs with selection logic, bypass capability, and signal quality determinations. Use the below chart for determination of attributes.

INPUT SIGNAL	SELECTION LOGIC	SIGNAL QUALITY	USE	FAILURE MODE
Narrow Range Lvl LT1113-1 thru 4 LT1123-1 thru 4	1 signal bypassed, Median Value is selected	Signal deviations and bad quality.	Low Power and High Power Modes	3 failures switch FWCS to manual

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	041 A2.03	
	Importance Rating	2.8	

Steam Dump/Turbine Bypass Control System: Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of IAS

Proposed Question: Common 34

Given the following conditions:

- Unit 3 experienced a complete Loss of Instrument Air and tripped from 100% power.

Which ONE (1) of the following:

1.) Describes the post-trip plant response?

2.) What action must be taken to mitigate the situation?

A. 1.) Steam Bypass Control Valves fail open and will NOT modulate closed as Steam Generator pressure lowers.
2.) Place the Master Controller in MANUAL to close valves.

B. 1.) Steam Bypass Control Valves fail closed until manually opened.
2.) Place the Master Controller in MANUAL and open as necessary to control Steam Generator pressure.

C. 1.) Steam Bypass Control Valves fail closed.
2.) Open the Atmospheric Dump Valves to restore Steam Generator pressure to normal.

D. 1.) Steam Bypass Control Valves fail as is.
2.) Place the Master Controller in MANUAL and close as necessary to control Steam Generator pressure.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because placing the Master Controller in MANUAL would close the valves for most conditions, however, during a loss of Instrument Air the SBCS Valves fail closed.

B. Incorrect. Plausible because it could be thought that a backup nitrogen supply was available to operate the valves, however, only the Atmospheric Dump Valves have this feature.

C. Correct. All air is lost to the SBCS Valves and they fail closed. The Atmospheric Dump Valves are in MANUAL and CLOSED until the operator takes control per SO23-12-1, SPTAs.

D. Incorrect. Plausible because it could be thought that SBCS Valves would fail as is (which under normal conditions would be closed) to prevent a power excursion. If this were the case the action described would be correct if backup air were available.

Technical Reference(s): SO23-13-5, Attachment 2; SO23-12-1, Step 8b RNO — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54348 — DESCRIBE the cause/effect relationships associated with the following Steam Bypass Control System conditions/operations:
EFFECT on plant operation of a failure of the Steam Bypass Control System.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 5, 10
55.43 ______

Comments / Reference: From SO23-13-5, Attachment 2	Revision # 7

NUCLEAR ORGANIZATION
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AFFECTED UNIT EQUIPMENT RESPONSE (Continued)

SECONDARY PLANT: CLOSING S2(3)2417MU028 WILL RESULT IN THE RESPONSE OF ALL EQUIPMENT IN THE AFFECTED AREAS

AFFECTED AREA	IMPACTED EQUIPMENT TO ISOLATED AREA
ADV/MFW Area Isolation: S2(3)2417MU076 (Above MP-053 Suction)	• ADVs lose air, will rely on Backup Nitrogen • HV-1105 and HV-1106, Feedwater bypass Valves, FAIL-CLOSED. • FV-1111 and FV-1121, Feedwater Regulating Valves, FAIL-AS-IS.
MSIV/Tank Farm Area Isolation: S2(3)2417MU043 (7' TB at Seal Oil Skid)	• HV-8200 and HV-8201, P-140 Main Steam ISO VLVs, FAIL-OPEN • HV-4053 and HV-4054, Blowdown ISO VLVs, FAIL-CLOSED • HV-4762 and HV-4763, AFW Low Flow Bypass Valves, FAIL-CLOSED
SBCS Area Isolation: S2(3)2417MU117	• Steam Bypass Valves FAIL-CLOSED

Comments / Reference: From SO23-12-1, Step 8b RNO	Revision # 21

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STANDARD POST TRIP ACTIONS

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7 VERIFY Core Heat Removal criteria satisfied:	
a. VERIFY at least one RCP – operating.	a. GO TO step c.
b. VERIFY core loop ΔT ($T_H - T_C$) – less than 10°F.	
c. VERIFY Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311.	
8 VERIFY RCS Heat Removal criteria satisfied:	
a. VERIFY at least one S/G level – between 21% NR and 80% NR. AND Feedwater – available.	a. ENSURE EFAS – actuated.
b. VERIFY heat removal adequate: 1) RCS T_C – trending to between 545°F and 555°F.	b. 1) IF RCS T_C – greater than 555°F, THEN a) OPERATE SBCS to maintain RCS T_C – between 545°F and 555°F. OR b) OPERATE ADVs to maintain RCS T_C – between 545°F and 555°F.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	055 A3.03	
	Importance Rating	2.5	

Condenser Air Removal System: Ability to monitor automatic operation of the CARS, including: Automatic diversion of CARS exhaust

Proposed Question: Common 35

Given the following conditions:

C is a typo

- Unit 2 is operating at 100%.
- Annunciator C60A46 - SECONDARY RADIATION HI is in alarm.
- RE-7818, Low Range Air Ejector Radiation Monitor has alarmed and RE-7870, Wide Range Air Ejector Radiation Monitor has a rising trend but is not in alarm.
- A Steam Generator Blowdown Radiation Monitor is in alarm.

Which ONE (1) of the following correctly describes the current condition of A361, Steam Air Ejector Exhaust Unit and the proper operator response?

A. Flow has been automatically aligned through A361, Steam Air Ejector Exhaust Unit. Place A361 Control Switch in DIRECT to ensure flow through the unit.

B. Flow is normally aligned through A361, Steam Air Ejector Exhaust Unit. Place the Exhaust Unit Heater to ON.

C. Flow has been automatically aligned through A361 due to RE-7818 ALERT alarm. Place A361 Control Switch to ON and ensure heater is in AUTO.

D. Flow is bypassing A361, Steam Air Ejector Exhaust Unit. Place A361 Control Switch in DIRECT to manually align flow through the unit.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that this automatic feature exists, however, this action must be performed by the operator.
B. Incorrect. Plausible because it could be thought that this could be the normal alignment, however, this is not the normal alignment in order to preserve A361.
C. Incorrect. Plausible because it could be thought that this feature exists and is from the wide range monitor versus the narrow range, however, alignment must be performed by an operator.
D. Correct. The unit has no AUTO alignment features and is not normally aligned. The operator places the control switch in DIRECT based on Annunciator Response Procedure guidance.

Technical Reference(s)	SO23-15-60.A2, 60A46 SD-SO23-190, Page16	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52660	Given plant conditions, PREDICT and EXPLAIN the response of major plant systems, equipment and parameters to a steam generator tube rupture.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7, 10, 13	
	55.43		

Comments / Reference: From SO23-15-60.A2, 60A46	Revision # 14

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60A46 SECONDARY RADIATION HI (Continued)

2.0 CORRECTIVE ACTIONS:

NOTE

Due to the high air flow with 2(3)MP-054, Condenser Vacuum Pump, running, the sensitivity of 2(3)RE-7870, Condenser Air Ejector Radiation Monitor, to monitor for steam generator tube leakage is greatly reduced.

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 HI radiation in Steam Generator Blowdown Processing	2.1 Check radiation levels of 2(3)RE-7818 and 2(3)RE-7870. 2.1.1 If high radiation levels are present, then perform the following: .1 START Vacuum Pump 2(3)MP-054 per SO23-10-7, Section for Starting 2(3)MP-054, Vacuum Pump (Vacuum Established). .2 PLACE A-361, Air Ejector Exhaust Unit, in DIRECT. .3 GO TO SO23-13-14, Reactor Coolant Leak. 2.1.2 Notify 70' HP Control Point to monitor increasing radiation levels on BPS filters, demineralizers and effluent. NOTE: Main Steam Line Monitors will not respond to Tech. Spec. primary to secondary leakage limits of 1 gpm total, with typical RCS activities (i.e., < 1% failed fuel).

Comments / Reference: From SD-SO23-190, Page 16	Revision # 11

2.3.2 **Steam Air Ejector Exhaust Unit**

.1 The Air Ejector Exhaust Unit is controlled from a Main Control Room 2(3)CR-60. The control switch has "DIRECT" and "BYPASS" positions. When switch is in BYPASS, valve 2(3)HV-9792A is OPEN and valves 2(3)HV-9792B & C are CLOSED. When the DIRECT position is selected, valves 2(3)HV-9792B & C OPEN and valve 2(3)HV-9792A CLOSES placing the Unit in service.

.2 The Air Ejector Exhaust Unit is normally in the "BYPASS" position. When high radiation is detected, by Radiation monitors 2(3)RE7818A and/or 2(3)RE7870A1, B1, C1, the Unit is normally shifted to the "DIRECT" position. For details of operation of radiation monitors refer to Radiation Monitoring System Description (SD-SO23-690).

.3 The Exhaust Unit Heater is controlled by handswitch 2(3)HS-9797 on Ventilation Control Panel 2(3)L-155. The switch has three positions: OFF, ON and AUTO. In AUTO, humidity of 50% in the exhaust flow energizes the Heater.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	056 K1.03	
Importance Rating	2.6	

Condensate System: Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: MFW

Proposed Question: Common 36

Given the following conditions:

- Unit 2 is operating at 100% power.
- All Condensate Pumps are running.
- Condensate Pump P053 trips on overcurrent.
- Main Feedwater Pumps are in their normal alignment.

Which ONE (1) of the following describes the effect on the Main Feedwater Pumps?

A. Reheat Steam Supply Valves open and Main Feedwater Pump speed rises.

B. Reheat Steam Supply Valves close and Main Feedwater Pump speed lowers.

C. Main Steam Supply Valves open and Main Feedwater Pump speed rises.

D. Main Steam Supply Valves close and Main Feedwater Pump speed lowers.

Proposed Answer: A

Explanation:

A. Correct. With all four Condensate Pumps running at 100% power the trip of one Condensate Pump will lower suction pressure to the MFW Pump which will lower discharge pressure. With the lower discharge pressure less Feedwater will enter the Steam Generator, Steam Generator level will lower, and the Feedwater Control Valve will open to raise Steam Generator level. As the control valve opens valve differential pressure will lower and the Master Controller will send a signal to the MFW Pump to increase its speed. Given the conditions listed, the Reheat Steam Supply Valves control steam flow to the MFW Pump and open to raise MFW Pump speed.

B. Incorrect. Plausible because at this power level the Reheat Steam Supply Valves are in control, however, the valves would open and speed would rise.

C. Incorrect. Plausible because Main Feedwater Pump rises, however, at this power level the Reheat Steam Supply Valves are in control.

D. Incorrect. Plausible if thought that Main Feedwater Pump speed lower, however, at this power level the Reheat Steam Supply Valves are in control.

Technical Reference(s) SO23-9-6, Section 6.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 64706 DESCRIBE the configuration and operational characteristics of Main Feedwater Pump and Turbine system components.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 7, 10
55.43

Comments / Reference: From SO23-9-6, Section 6.2	Revision # 21

NUCLEAR ORGANIZATION UNITS 2 AND 3 — OPERATING INSTRUCTION REVISION 21 — SO23-9-6 PAGE 5 OF 55

6.0 PROCEDURE (Continued)

6.2 Feedwater Control System Operation - On Line

INFORMATION USE

NOTE

S/G level fluctuations have a direct relationship to the ΔP across the Main Feedwater Control Valves and the number of Condensate Pumps running. When the fourth Condensate Pump is put on line, then the ΔP across the Main Feedwater Control Valves is increased. The Digital Control System minimizes Steam Generator level oscillations when the fourth Condensate Pump starts. (LS-2.3, LS-2.4)

6.2.1 The recommended ΔP across the Feedwater Control Valves is 90 to 120 psid, regardless of the number of Condensate Pumps in service. (LS-4.4)

.1 Change ΔP using the bias adjustment on HIC-1107 and HIC-1108, Main Feedwater Pump Turbine (MFWPT) Speed Controllers, while matching MFW Pump speeds (RPM) as closely as possible. (AR 070100098) [LS-2.3, LS-2.4]

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072 A1.01	
	Importance Rating	3.4	

Area Radiation Monitoring System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: Radiation levels

Proposed Question: Common 37

Which ONE (1) of the following Area Radiation Monitors (ARM) initiates an automatic actuation of equipment on increasing radiation levels?

A. RE-7851, Control Room General Area Radiation Monitor.

B. RE-7820, Containment High Range Radiation Monitor.

C. RE-7874, Main Steam Line Radiation Monitor.

D. RE-7839, PASS Lab Radiation Monitor.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that this monitor what automatically realigned Control Room ventilation, however, it is a different monitor located inside the air plenum.
B. Incorrect. Plausible because it could be thought that this monitor would initiate a Containment Purge Isolation Signal, however, it is the Containment Airborne Radiation Monitors that perform this function.
C. Incorrect. Plausible because it could be thought that this monitor would close the Main Steam Isolation Valves, however, this monitor is used for indication only.
D. Correct. This monitor will isolate sampling to the Chemistry Lab and bypass the Sample Coolers on high radiation.

Technical Reference(s) SD-SO23-690, Page 60 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 103329 DESCRIBE the configuration and operational characteristics of Radiation Monitoring System components.

Question Source: Bank # 75147
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 11
55.43

Comments / Reference: From SD-SO23-690, Page 60	Revision # 16

NUCLEAR ORGANIZATION SYSTEM DESCRIPTION SD-SO23-690
UNITS 2 AND 3 REVISION 16 PAGE 60 OF 157

2.0 DESCRIPTION (Continued)

2.3.3 **Liquid Process Radiation Monitoring System** (Continued)

.3 Normal Sample Lab Isolation Area Radiation Monitors, 2/3RE-7838 & 2/3RE-7839 (See Section 2.2.5.2 and Figure 7)

.3.1 The Normal Sample Lab Isolation monitors are not traditional process monitors. They are physically area radiation monitors located in the vicinity of pipes carrying liquid sample from:

.3.1.1 the RCS Hot Leg 1

.3.1.2 the RCS Hot Leg 2, and

.3.1.3 the Pressurizer Surge Line

.3.1.4 to the Normal Sample Lab (Radio-Chem Lab).

.3.2 They consist of a GM detector and ionization chamber. The monitors are located in the Post Accident Sample System (PASS) lab valve gallery, at elevation 24 foot of the Radwaste Building.

.3.3 The monitor functions to alarm, isolate sampling to the Radio-Chem Lab and bypass the Sample Coolers upon high radiation in the sample lines.

.3.4 The communication/control modules for these monitors are on 2/3L13B in the PASS lab.

.3.5 2/3RE-7838 and 2/3RE-7839 have indication of radiation levels in the PASS room on 2/3L-13B (2/3RI-7838 and 2/3RI-7839), on the wall of the stairway down to the PASS Lab (2/3RI-7838A and 2/3RI-7839A) and in the Radio-Chem Lab (2/3RI-7838B and 2/3RI-7839B). High radiation alarms at all of these places.

Comments / Reference: From SO23-12-11, FS-28, Step b1 Caution	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 6
ATTACHMENT 2

SO23-12-11 ISS 2
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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
FS-28 MONITOR Isolated S/G (Continued)	
b. **MONITOR Isolated S/G Level and Pressure:**	b.

CAUTION

When isolated S/G level is excessively high (90% NR or greater), the possibility of valve damage and uncontrolled radioactive releases from direct water relief through the ADVs should be considered before steaming the isolated S/G.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1) VERIFY isolated S/G level – less than 80% NR. OR S/G cooldown in progress per Attachment 18, BACKFLOWING THEN FEEDING A S/G.	1) REQUEST Shift Manager/Operations Leader to evaluate a method of lowering S/G level: a) INITIATE backflow to RCS: 1) INITIATE Shutdown Margin monitoring per step c.2) RNO. 2) IF no RCPs are operating, THEN ENSURE RCPs in affected loop disabled per step 6). RNO. 3) LOWER RCS pressure below isolated S/G pressure per Attachment 3, COOLDOWN / DEPRESSURIZATION. b) INITIATE draining S/G to Radwaste per Attachment 25, S/G DRAIN ALIGNMENT TO RADWASTE. c) INITIATE steaming the affected S/G by SBCS operation through MSIV bypass. d) INITIATE blowdown of the affected S/G e) INITIATE steaming the affected S/G by ADV operation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	086 K6.04	
	Importance Rating	2.6	

Fire Protection System: Knowledge of the effect of a loss or malfunction of the following will have on the Fire Protection System: Fire, smoke, and heat detectors

Proposed Question: Common 38

Which ONE (1) of the following conditions will result following an inadvertent actuation due to a Fire Detection System failure?

A. Emergency Diesel Generator Building Pre-Action Sprinkler System pressurized dry pipe will depressurize and cause the deluge valve to open initiating flow.

B. Hydrogen Seal Oil System Water Spray System will actuate to pressurize the header and when a fusible link melts spray flow is initiated.

C. Main Turbine Bearing Boat HALON will actuate and flow is immediately initiated through open nozzles.

D. Computer Room Fire Dampers close and HALON is discharged.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this type of pre-action system exists but not in the Emergency Diesel Generator Building.
B. Incorrect. Plausible because it could be thought that to preclude an inadvertent spray down of the Seal Oil Unit a second, confirmatory actuation should occur. This would be a pre-action type of system instead of the water spray system installed at SONGS.
C. Incorrect. Plausible because the statement is correct with the exception that the Main Turbine system uses carbon dioxide.
D. Correct. This is the correct sequence for an inadvertent actuation due to detector failure.

Technical Reference(s) SD-SO23-590, Pages 12, 14, 20, 69 & 70 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 107169 / 107170
DESCRIBE the configuration and operational characteristics of Fire Protection System components.
INTERPRET instrumentation and controls utilized in the Fire Protection System.

Question Source: Bank # ____________

	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7, 8	
	55.43		

Comments / Reference: From SD-SO23-590, Page 12	Revision # 14

.5 **Automatic Water Suppression Systems** (see Figures 1-A, B) (Continued)

.5.2 Wet Pipe Sprinklers

.5.2.1 Wet Pipe Sprinklers are supplied water from the Firemain through an isolation valve and an alarm check valve.

.5.2.2 The Sprinklers are Fusible Link Type, which will initiate spray when the fusible link melts.

.5.2.3 Control Room Panel CR61 indicates position of each wet pipe sprinkler inlet isolation valve.

.5.2.3.1 Upon initiation of flow through a sprinkler, the alarm check valve repositions to annunciate actuation on Control Room Panel CR61 and the Fire Detection CRT.

.5.2.3.2 Flow also actuates a local alarm bell.

.5.3 Automatic Pre-action Sprinklers

.5.3.1 Automatic Pre-action Sprinkler Systems are supplied by water from the Firemain via an isolation valve and a pre-action deluge valve.

.5.3.1.1 The pipe downstream of the deluge valve is normally dry.

.5.3.1.2 The sprinklers are fusible link type.

.5.3.1.3 A typical deluge valve and its associated Release Enclosure Box are shown in Figure 5.

.5.3.1.4 The valve may be automatically actuated by the associated local panel, or manually by using the manual pull handle.

.5.3.1.5 The clapper to the deluge valve is normally held closed by a latch.

Comments / Reference: From SD-SO23-590, Page 14	Revision # 14

2.1.2 **The Fire Protection System consists of eight separate subsystems:** (Continued)

.5 **Automatic Water Suppression Systems** (see Figures 1-A, B) (Continued)

.5.3.7 Control Room indication of status of the inlet isolation valve is provided at Panel CR61.

.5.3.7.1 Initiation of the deluge valve is monitored on the Fire Computer Display CRT.

.5.4 Water Spray Systems

.5.4.1 Water Spray Systems are supplied water from the Firemain through an isolation valve and a deluge valve.

.5.4.1.1 The sprinklers are open nozzle type.

.5.4.2 Deluge valves similar to those used in the pre-action sprinkler systems are used, except that water spray systems are initiated by thermal fire detectors, and spray commences immediately upon valve actuation.

.5.4.3 Control Room indication of inlet isolation valve status is provided at Panel CR61.

.5.4.3.1 Initiation of the deluge valve is monitored by the Fire Computer Display CRT.

.5.4.4 Fire water to the water spray systems for the Reactor Coolant pumps and Charcoal Filter Unit 2(3)A-353, enter Containment via a Containment Isolation valve prior to reaching the deluge valve and sprinklers.

Comments / Reference: From SD-SO23-590, Page 20	Revision # 14

.7 **Halon 1301 Suppression System** (see Figures 3A - 3C) (Continued)

.7.3 Thermal detectors are used in the Computer Rooms.

.7.3.1 Ionization smoke detectors are used in the Telecommunications Room and Radio Chemical Counting Room #1 for fire detection.

.7.3.2 The Radio Chemical Counting Room #1 also has a photoelectric smoke detector.

.7.3.3 Upon detection, a pre-discharge alarm is initiated in the area.

.7.3.4 Halon system actuation is alarmed on a CRT display on Panel CR61.

.7.3.5 A common alarm signal is annunciated on a window on CR61.

.7.4 The first detector alarm received in the Telecommunications Room initiates an alarm and automatically shuts down the Room's A/C Unit.

.7.4.1 The second alarming detector signal initiates release of Halon after a 30 second time delay.

.7.5 Discharge of Halon is delayed for 30 seconds to allow the personnel time to leave the area.

.7.5.1 The Smoke Duct Dampers are automatically CLOSED in the Computer Rooms and the Radio Chemical Counting Room #1.

.7.5.2 Computer Room Fire Dampers, 2(3)HV-9715A and B and 2(3)HV-9734A and B, CLOSE to isolate the area.

.7.5.3 The Telecommunications Room Air Handling Units are automatically shut down upon Halon actuation.

Comments / Reference: From SD-SO23-590, Page 69	Revision # 14

TABLE 1
FIRE PROTECTION SYSTEM DETECTION, ALARMS, AND SUPPRESSION

AREA	DETECTION DEVICE	ALARM POINT	SUPPRESSION SYSTEMS
Primary and Containment			
Reactor Coolant Pumps	fixed temp	CR and Local	Water Spray System, portable CO2, portable dry chemical extinguishers, rate of rise fire hose station. A lube oil collection system is provided for the RCPs.
Charcoal Filters	fixed temp	CR and Local	Water Spray System, portable CO2, portable dry chemical extinguishers, rate of rise fire hose station.
General areas	smoke detector	CR	Portable CO^2, portable dry chem. extinguishers, fire hose station.
Elevator Machinery Room	smoke detector	CR and Local	Portable CO_2 extinguishers
Control Building			
Control Room	smoke detector	CR and Local	Portable CO_2 extinguishers, portable dry chem. extinguishers, manual fire hose station, fire hose from hydrant, portable water exting.
Cable Spreading Room	smoke detector fixed temp. rate of rise	CR and Local	Water Spray System, portable CO_2 extinguishers, manual fire hose station, fire hose from hydrant.
Plant Computer Rooms	smoke detector fixed temp. rate of rise	CR and Local	Automatic Halon 1301 System, portable CO_2 extinguishers, fire hose station, fire hose from hydrant.

Comments / Reference: From SD-SO23-590, Page 70	Revision # 14

TABLE 1
FIRE PROTECTION SYSTEM DETECTION, ALARMS, AND SUPPRESSION

AREA	DETECTION DEVICE	ALARM POINT	SUPPRESSION SYSTEMS
Switchgear Rooms	smoke detector fixed temp. rate of rise	CR and Local	Fire hose station, fire hose from hydrant, portable CO_2 extinguishers.
Remote Shutdown Room	smoke detector	CR and Local	Fire hose station, fire hose from hydrant, portable CO_2 extinguishers
Station Battery Rooms Safety Related Nonsafety Related	 smoke detector smoke detector	 CR and Local CR and Local	Portable CO_2 extinguishers, portable dry chem. Fire hose station, fire hose from hydrant, portable CO_2 extinguishers. Fire hose station, fire hose from hydrant, portable CO_2 Fixed Automatic Water Spray System, portable CO2 extinguishers.
Cable riser areas	smoke detector fixed temp.	CR and Local	Water Spray System, portable CO2 extinguisher, fire hose station, fire hose from hydrant rate of rise.
Emergency chillers	smoke detector	CR and Local	Manual Water Spray System, fire hose stations, fire hose from hydrant.
Fan Room Charcoal Filters	smoke detector fixed temp. rate of rise	CR and Local	Water Spray System, fire hose stations, fire hose from hydrant.
El 9', 30', 50', and 70' corridors	smoke detector fixed temp. rate of rise	CR and Local	Wet Pipe Sprinkler system.
Technical Support Center	smoke detector fixed temp. rate of rise	CR and Local	Portable CO_2 extinguisher, portable dry chem. extinguisher, fire hose from hydrant, portable water extinguishers.

Comments / Reference: From SD-SO23-590, Page 71	Revision # 14

Turbine Building and Deck			
Turbine - Generator	smoke detector fixed temp. rate of rise	CR and Local	Water Spray System, portable CO_2 extinguishers, portable dry chemical extinguishers, fire hose station, fire hose from hydrant.
Turbine - Generator Bearings	Fixed temp. rate of rise	CR and Local	Automatic CO_2 System, portable CO_2 extinguishers, portable dry chemical extinguishers, fire hose station, fire hose from hydrant.
Lube Oil Reservoir Room 103	Fixed temp. rate of rise	CR and Local	Water Spray System, Wet Pipe Sprinkler system, portable CO_2 extinguishers, portable dry chemical extinguishers, fire hose stations, fire hose from hydrant.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	007 EK2.03	
Importance Rating	3.5	

Reactor Trip - Stabilization - Recovery: Knowledge of the interrelations between a reactor trip and the following: Reactor trip status panel

Proposed Question: Common 39

Given the following conditions:

- An Anticipated Transient Without Scram (ATWS) has occurred on Unit 2.
- The Reactor Operator has opened the supply breakers to 2B15 and 2B16.

Which ONE (1) of the following describes the Reactor trip indication available on the Reactor Trip Status Panel?

1. MG Set Output Contactor RED lights _____.
2. UV WHITE lights _____.
3. Reactor Trip Circuit Breaker RED lights _____.

A. 1.) ON
2.) ON
3.) OFF

B. 1.) OFF
2.) OFF
3.) ON

C. 1.) ON
2.) OFF
3.) OFF

D. 1.) OFF
2.) ON
3.) ON

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this is the indication available following a Reactor trip.
B. Correct. The MG set breakers open (green lights are illuminated), the UV lights extinguish, however, the RTCBs remain closed.
C. Incorrect. Plausible because the UV lights are extinguished, however, MG set red lights are off and the Reactor Trip Circuit Breakers are still closed with their lights illuminated.
D. Incorrect. Plausible because the MG set lights are extinguished and the RTCBs remain closed, however, when power is lost to the UV coils their respective lights extinguish.

Technical Reference(s): SD-SO23-710, Figure 2; SD-SO23-520, Page 4 & Figure 2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56628 / 56622

DESCRIBE the inputs to the Plant Protection System, the purpose of each, their trip setpoints and actuation logic.

DESCRIBE the response of the Plant Protection System to failures and alarms, including possible causes, effects on the system or overall plant, and operator actions to mitigate the effects.

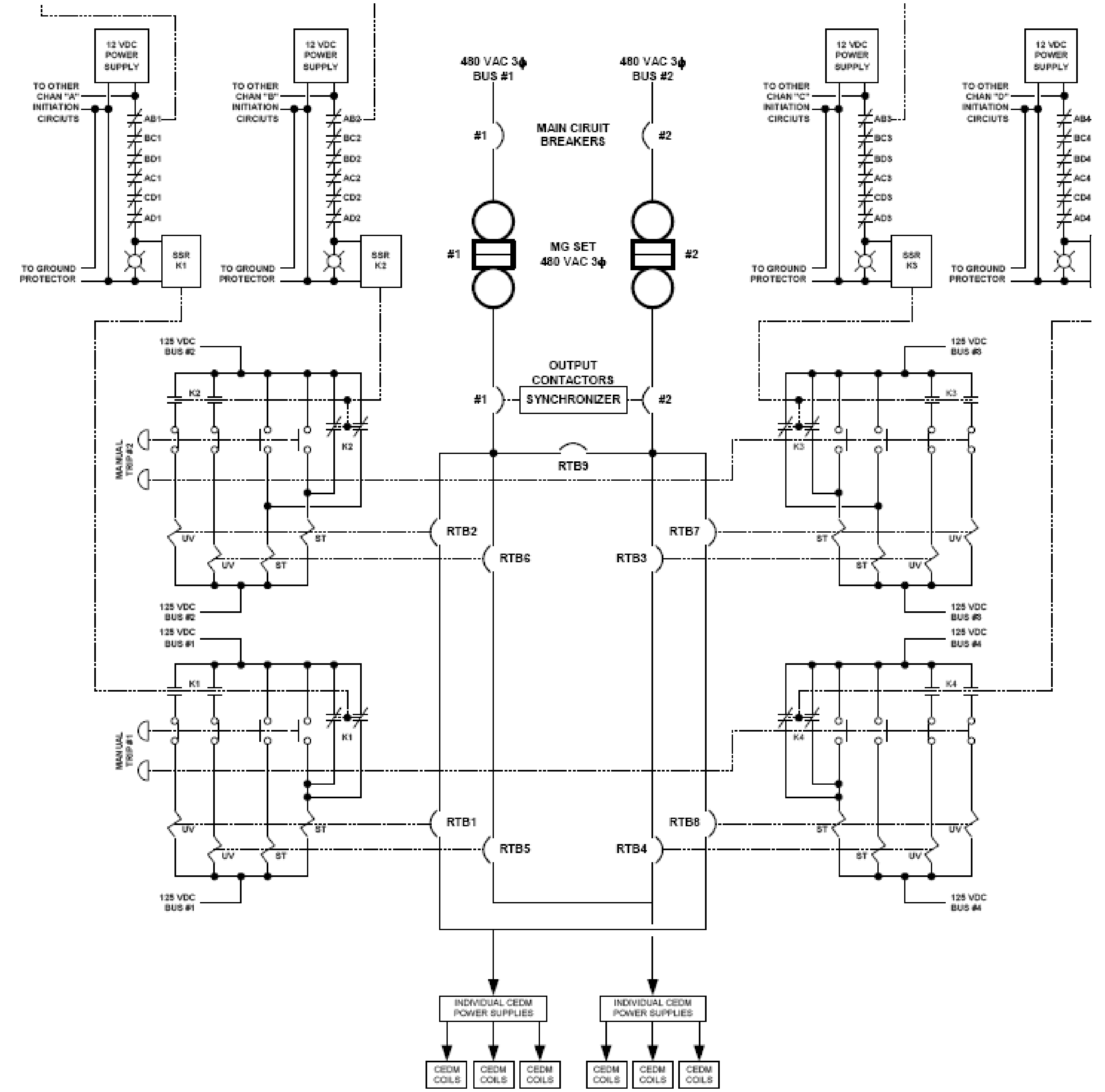
Question Source:
Bank # 128139
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 6
55.43 ____

Comments / Reference: From SD-SO23-710, Figure 2	Revision # 7



Comments / Reference: From SD-SO23-520, Page 4	Revision # 6

2.1.1 Main Signal Paths (see Figures 2 & 3)

.1 The ATWS/DSS Main Signal Path, which consists of four measurement channels, four 2-out-of-4 Logics and two Trip Paths, initiates a Reactor Trip in the event of Hi-Hi PZR pressure (indicative of an ATWS).

.2 Each measurement channel consists of a Sensor (Pressure Transmitter), a Signal Conditioner (Current to Voltage Converter) and an Alarm Block and Timer Blocks which are part of the configured function blocks of the Spec. 200 Micro Control Module. (See Figures 2 & 3)

.3 The Measurement Channel measures the Pressurizer Pressure and generates a Trip Signal Output to all four 2-out-of-4 Logics when the Pressurizer Pressure reaches or exceeds the 2428 psia setpoint indicative of an ATWS event.

.4 Each of the four 2-out-of-4 Logics, which is also a configured function block of the Foxboro Spec. 200 Micro-module, activates one of the two Trip Paths to OPEN an M-G Set Output Contactor.

.4.1 This occurs when any two of the four inputs from the four Measurement Channels reach the High-High Pressurizer Pressure Setpoint.

.4.2 This produces an output from all four 2 out of 4 Logic Channels and activates trip path 1 and Trip Path 2 Relays.

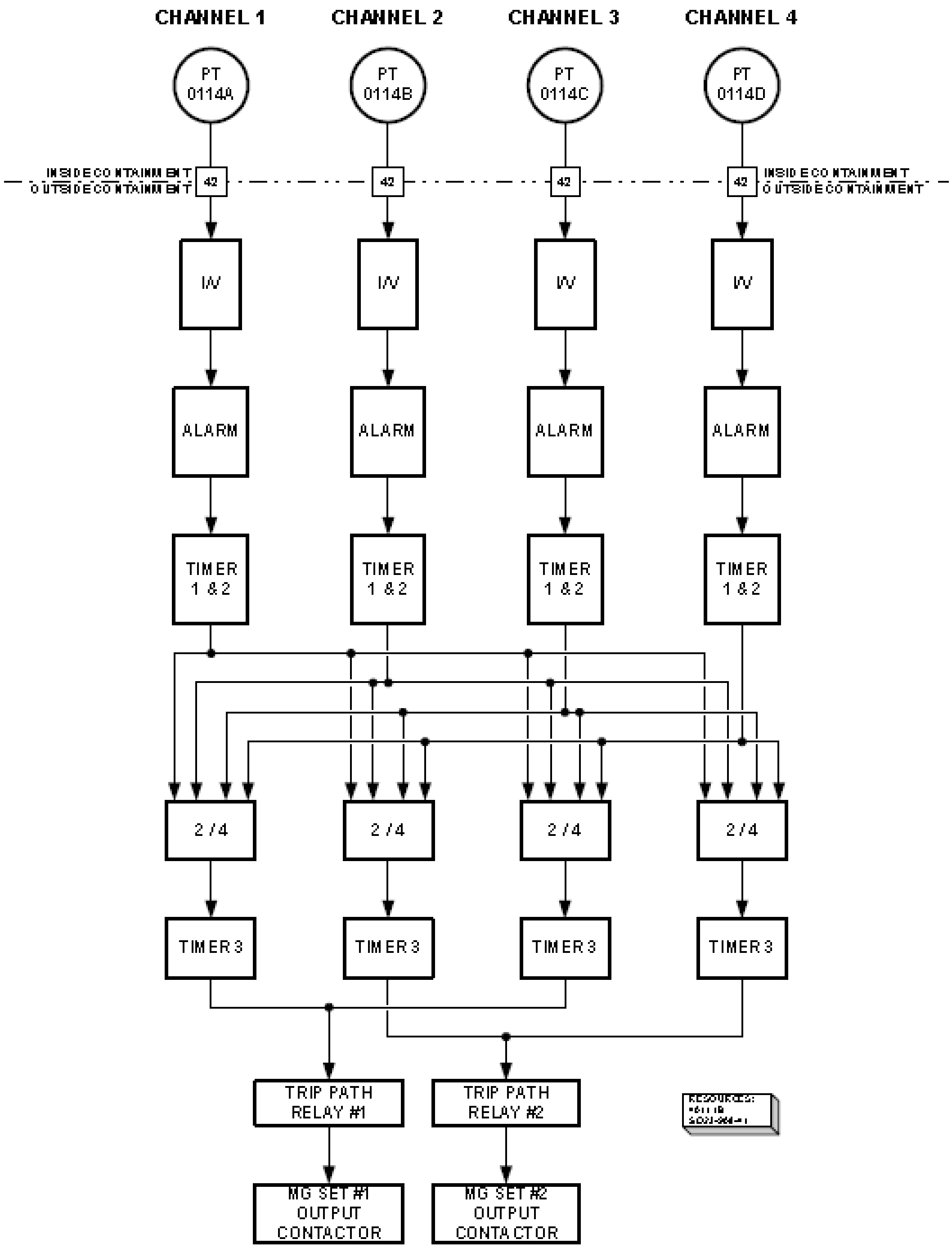
.4.3 These Trip Relays in turn OPEN M-G Sets 1 & 2 Output Contactors.

.4.3.1 The 2 out of 4 Logic Channels 1 and 3 activate Trip Path 1 Trip Relays which OPEN M-G Set #1 Contactor.

.4.3.2 The 2 out of 4 Logic Channels 2 and 4 activate Trip Path #2 Relays which OPEN M-G Set #2 Contactor.

.4.3.3 Outputs from Logic Channel 1 **or** 3 **and** 2 **or** 4 are required to OPEN both M-G Set Contactors to Trip the Reactor.

Comments / Reference: From SD-SO23-520, Figure 2	Revision # 6



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008 AK1.02	
Importance Rating	3.1	

Pressurizer Vapor Space Accident: Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: Change in leak rate with change in pressure

Proposed Question: Common 40

Given the following initial conditions:

- A Reactor Coolant System leak has occurred.
- Reactor Coolant System pressure is 2200 psia.
- Pressurizer level is 20% and lowering.
- The leak size is estimated at 500 gpm.

Current conditions:

- The Reactor Coolant System leak is determined to be a Pressurizer vapor space break.
- Reactor Coolant System pressure is 1600 psia.
- Pressurizer level is 50% and rising.

With the change in Reactor Coolant System pressure noted above, which ONE (1) of the following is the resultant Reactor Coolant System leak rate?

A. ~300 gpm.

B. ~375 gpm.

C. ~425 gpm.

D. ~500 gpm.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this answer could be arrived at if an error is made calculating the square root of the differential pressure to determine flow rate.
B. Incorrect. Plausible if thought that leakrate was based on a simple ratio/proportion calculation. In this instance 3/4^{ths} the pressure would yield 1/4th the leak rate.
C. Correct. This question meets the KA because it tests the operator's understanding of the change in leakrate based on a change in pressure. Leak rate is proportional to the square root of the ΔP. 1/4th the original pressure will correspond to a leak rate approximately 85% of the original leak rate. The operational implications of this are important when evaluating ability to keep the core covered.
D. Incorrect. Plausible if thought that the leak rate would not change because it is a vapor space leak and Pressurizer level is rising.

Technical Reference(s): NRC Exam Formula Sheet — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: NRC Exam Formula Sheet

Learning Objective: 54932 — Per the Reactor Coolant Leak procedure, SO23-13-14, DESCRIBE : The expected plant response for each step.

Question Source:
Bank # ______
Modified Bank # 127170 (Note changes or attach parent)
New ______

Question History: Last NRC Exam SONGS 2005B

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

10 CFR Part 55 Content:
55.41 14
55.43 ______

Comments / Reference: Square Root of Differential Pressure Calculation	Revision # N/A
Square root of 2200 psia = 46.9 psid Square root of 1600 psia = 40 psid therefore: 40 / 46.9 = .852 x 500 gpm = ~426 gpm	
Comments / Reference: Ratio & Proportion Calculation	Revision # N/A
2200 psia / 1600 psia = 500 gpm / x gpm → 800000 = 2200 x → x = ~363 gpm	

Comments / Reference: Exam Bank #127170	Revision #10/23/06

Given the following initial conditions:

- A Reactor Coolant System leak has occurred.
- Reactor Coolant System pressure is 2200 psia.
- Pressurizer level is 20% and lowering.
- The leak size is estimated at 1000 gpm.

Current conditions:

- The Reactor Coolant System leak is determined to be a Pressurizer vapor space break.
- Reactor Coolant System pressure is 1100 psia and stable.
- Pressurizer level is 70% and rising.

Which ONE (1) of the following is the current approximate Reactor Coolant System leak rate?

A. ~300 gpm.

B. ~500 gpm.

C. <u>~700 gpm.</u>

D. ~1000 gpm.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	009 EK2.03	
Importance Rating	3.0	

Small Break LOCA: Knowledge of the interrelations between the small break LOCA and the following: SGs

Proposed Question: Common 41

Given the following conditions:

- A Small Break Loss of Coolant Accident is in progress.
- The Safety Injection Actuation Signal has actuated.
- All systems are operating as expected.

Per the stated conditions, which ONE (1) of the following is the basis for maintaining a secondary heat sink?

A. To minimize boron stratification of the RCS.

B. Cooling from the injection flow alone is inadequate to remove decay heat.

C. Reflux boiling is the primary means of heat removal prior to voiding in the hot legs.

D. Minimize potential for PTS during cooldown and depressurization phase.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because boron stratification is a concern, however, more so during a Large Break LOCA.
B. Correct. Any Reactor Coolant System pressure that remains above the shutoff head of the High Pressure Safety Injection Pumps will result in a lowering of inventory without the attendant makeup. The Steam Generators provide decay heat removal until the system is cooled down and depressurized.
C. Incorrect. Plausible because this statement is true if talking about a large break LOCA.
D. Incorrect. Plausible because it could be thought that PTS was a concern during a small break LOCA, however, the concern is the ability to remove decay heat.

Technical Reference(s) SO23-14-3, Section 2.0 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53006	PREDICT and EXPLAIN the response of major plant systems, equipment and parameters to a loss of forced circulation		
Question Source:	Bank #	75386	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2008	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	5	
	55.43		

Comments / Reference: From SO23-14-3, Section 2.0	Revision # 8

2.0 EVENT DESCRIPTION

LOCA is an accident that is caused by a break in the Reactor Coolant System (RCS) pressure boundary. The break can be as large as a double-ended guillotines break in the hot leg or as small as a break that results in a failure of one of the Safety Functions, (i.e., Containment pressure or Containment radiation levels). A LOCA is characterized by an initial decrease in RCS pressure and inventory. Subsequent RCS inventory and pressure response depends upon the size of the break.

Small and large break LOCAs differ in their effect on the post-LOCA RCS heat removal process. For a large break, the heat removal path is flow out the break, with the Shutdown Cooling System (SDC) Heat Exchangers providing cooling, (after a Recirculation Actuation Signal has occurred). For small breaks, heat removal via the flow out the break is not sufficient to provide cooling; therefore Steam Generator (S/G) heat removal is required. The action steps to be used during the actual emergency do not require the operator to distinguish between break sizes.

For large breaks inside Containment, an increase in Containment temperature and pressure occurs relatively soon after the LOCA. However, in the short term, a small break LOCA may not be detectable on Containment temperature and pressure instruments, but may be seen on radiation monitoring instruments in Containment.

The LOCA primarily affects RCS Inventory and Pressure Control, RCS and Core Heat Removal Safety Functions; and, to a lesser degree, Reactivity Control, Containment Isolation, and Containment Pressure and Temperature Control Safety Functions. However, all Safety Functions should be monitored to assure public safety and to detect failures that may indicate other events in progress.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	011 EA1.05	
Importance Rating	4.3	

Large Break LOCA: Ability to operate and/or monitor the following as they apply to a large break LOCA: Manual and/or automatic transfer of suction of charging pump to borated source

Proposed Question: Common 42

Given the following conditions:

- Unit 3 has experienced a large break Loss of Coolant Accident with initiation of SIAS and CIAS.
- Train A 4160 V Bus 3A04 was lost following a plant trip.
- 3LV-0227B, Volume Control Tank Outlet Valve failed to operate on SIAS.
- NO operator action has been taken.

Which ONE (1) of the following describes the status of boration flow to the Charging Pump suction?

A. Boric Acid Makeup Pumps are supplying adequate suction flow through 3HV-9247, Emergency Boration Block Valve.

B. Refueling Water Storage Tank is supplying adequate suction flow through 3LV-0227C, RWST to Charging Pump Suction Isolation.

C. Boric Acid Makeup Tanks are supplying adequate suction flow through 3HV-9240 and 3HV-9235, Gravity Feed Valves.

D. There is NO boric acid flow to the Charging Pump suction; the VCT continues to supply adequate suction head.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the power loss did not affect the BAMU Pumps and Emergency Boration Valve. With this path aligned the failure of LV-0227B would have no affect.

B. Incorrect. Plausible because it could be thought that LV-0227C still had power and the RWST height of water would overcome the VCT.

C. Incorrect. Plausible because it could be thought that the BAMU Tank height of water would overcome the VCT or that there were two trains of isolation from the VCT outlet.

D. Correct. The Gravity Feed path is aligned but can not overcome the VCT NPSH.

Technical Reference(s) SD-SO23-390, Appendix F, Various Pages Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52868 / 53388

DESCRIBE the operation of the following Chemical and Volume Control System controls, including the name, function, interlocks, and location of each: Boric Acid Makeup Pumps (P-174 & P-175) control.

DESCRIBE the cause/effect relationships associated with the following Chemical and Volume Control System conditions and/or operations: The effect on overall plant operation of transferring charging pump suction from the VCT to the RWST.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
- 55.41 7, 8
- 55.43

Comments / Reference: From SD-SO23-390, Appendix F, Page 184	Revision # 17

CNTMT - Containment PEN - Penetration Area RWB - Radwaste Building MCR - Main Control Room

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	CONTROLS/ INTERLOCKS
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS	
Volume Control Tank (VCT), 2(3)T-077 and VCT Vent Valve, 2(3)HV-9209	50' RWB	MCR CR-58 L-042	MCR CR-58 L-042	Volume Control Tank (VCT), 2(3)T-077, purposes: - RCS Degas; - accumulate RCP Controlled; - Bleed-Off (CBO); - control RCS Hydrogen Concentration; - provide a surge volume of Reactor Coolant for RCS makeup - provide Charging Pump suction head VCT Vent is provided to relieve excess VCT pressure to the Waste Gas System.	Vertical cylindrical tank with a capacity of 4,780 gallons Design data: 75 psig; 250 °F; Blanket Gas: Normal: Hydrogen Shutdown: Nitrogen VCT Vent Valve: 3/4" air operated globe valve VCT Relief Valve: set at 75 psig with a capacity of 305 gpm **NORMALLY - 37 to 60%; 25 - 35 psig; 120 °F; Flow: Letdown 38 gpm; RCP Seal Flow 6 gpm**	HV-9209 has OPEN/CLOSED Pushbuttons AUTO level control will divert Letdown Flow to the Coolant Radwaste System on a high VCT Level @78%. AUTO makeup to the VCT will start on a low VCT Level @32%. Makeup to the VCT is provided by a preset blend of Primary Plant Make-up (demineralized) Water and Boric Acid.

Comments / Reference: From SD-SO23-390, Appendix F, Page 185	Revision # 17

VCT Outlet Valve, 2(3)LV-0227B	43' RWB	CR-57	CR-57 CR-58	To provide isolation of the VCT Outlet to the Charging Pumps on a Low-low Level or a SIAS.	4", motor operated gate valve **NORMALLY - OPEN**	HV-0227B: fails "AS-IS" on a loss of power OPEN/AUTO/MANUAL/ CLOSED Pushbuttons Provides isolation of VCT Outlet upon receipt of: Safety Injection Actuation Signal (SIAS) VCT Low-low Level @ 6%

Comments / Reference: From SD-SO23-390, Appendix F, Page 186	Revision # 17

CNTMT - Containment PEN - Penetration Area RWB - Radwaste Building MCR - Main Control Room

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	CONTROLS/ INTERLOCKS
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS	
RWST to Charging Pumps 2(3)LV-0227C	9' RWB	CR-57	CR-57 CR-58	LV-0227C prevents a complete loss of Charging Pump suction fluid during normal and emergency operations.	3", motor operated gate valve **NORMALLY - CLOSED**	HV-0227C: fails "AS-IS" on a loss of power OPEN / AUTO-MANUAL / CLOSED / OVERRIDE Pushbuttons Provides AUTO OPEN of RWST to Charging Pump Suction upon receipt of: Safety Injection Actuation Signal (SIAS) VCT Low-low Level @ 6% A SIAS Override Switch is provided to allow opening LV-0227C after an automatic OPEN.

Comments / Reference: From SD-SO23-390, Appendix F, Page 192	Revision # 17

PEN - Penetration Area FHB - Fuel Handling Building RWB - Radwaste Building SEB - Safety Equipment Building MCR - Main Control Room CB - Control Building

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	CONTROLS/ INTERLOCKS
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS	
					NORMALLY - 145-155 F	
Boric Acid Makeup Tanks 2(3)T-071 & 072	24' RWB	MCR CR-58	MCR CR-58 L-42	BAMU tanks provide a source of concentrated boric acid solution, with a minimum Technical Specification Limit of 2.3 wt%, for RCS injection during normal and emergency conditions.	TANK: Vertical, Cylindrical, 11,800 gal, 15 psig, 200°F HEATER: Electrical Strap-On, 2.25 kW each (2 Banks of 3 Each) FLUID: 2.25 wt% to 3.5 wt% Boric Acid **NORMALLY - ATMOSPHERIC PRESSURE, 80°F**	None

Comments / Reference: From SD-SO23-390, Appendix F, Page 193	Revision # 17

PEN – Penetration Area FHB – Fuel Handling Building RWB – Radwaste Building SEB – Safety Equipment Building MCR – Main Control Room CB – Control Building

COMPONENT	LOCATION			FUNCTIONS	DESIGN DATA	CONTROLS/ INTERLOCKS
	COMPONENT	CONTROL	INSTRUMENT		CAPACITY/TEMP/PRESS	
Boric Acid Makeup Pumps 2(3)P-174 & P-175	9' RWB	MCR CR-58	MCR CR-58	The BAMU pumps can provide boric acid to the charging pump suction header, RWST and VCT. They are also used to recirculate the BAMU tanks and transfer acid from one BAMU tank to the other.	Pump: Centrifugal, Horizontal, 480 VAC, 3 Phase, 25 hp, 3600 RPM, 31 FLA, 200 psig, 250°F, 231 ft., 145 gpm RUNOUT HEAD: 170 ft. FLOW: 220 gpm NPSH AVAILABLE: 7 ft. MINIMUM FLOW: 10 gpm **NORMALLY - 80°F, 10-13 psig**	SIAS START Selected BAMU Pump Starts when HS-0210 is selected to AUTO and a VCT Auto Makeup demand is present or HS-0210 is selected to BORATE
Emergency Boration Valve Block, 2(3)HV-9247	9' RWB	MCR CR-58	MCR CR-58 CR-57	Emergency Boration Block Valve, HV-9247, is designed to supply boric acid to the charging pumps suction header in an emergency.	3", motor operated gate valve **NORMALLY - CLOSED**	Fails: OPEN OPEN, CLOSE and OVERRIDE switchlight module SIAS OPENS
BAMU Tanks Gravity Feed Valves, 2(3)HV-9235 & 9240	26' RWB	MCR CR-58	MCR CR-58 CR-57	A gravity feed path is required, as defined in SO23-3-3.1, "Boric Acid Flow Path Testings," by TS LCO 3.1.9 and 3.1.10 Boration Systems Operating and Shutdown. These valves insure a boric acid path to the RCS.	3", motor operated gate valve **NORMALLY - CLOSED**	Fails: AS-IS OPEN, CLOSE and OVERRIDE switchlight module SIAS OPEN

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	015/17 AA2.01	
	Importance Rating	3.0	

RCP Malfunctions: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): Cause of RCP failure

Proposed Question: Common 43

The following annunciators are received in the Control Room:

- 56C24 - RCP P001 SEAL PRESS HI/LO.
- 56B57 - RCP BLEEDOFF FLOW HI/LO.

The Reactor Operator reports the following for Reactor Coolant Pump P-001:

- Vapor seal cavity pressure = 75 psia.
- Upper seal cavity pressure = 1162.5 psia.
- Middle seal cavity pressure = 2250 psia.

Which ONE (1) of the following describes the event in progress?

A. Lower Seal only has failed.

B. Middle Seal only has failed.

C. Both Lower and Middle Seals have failed.

D. Both Lower and Upper Seals have failed.

Proposed Answer: A

Explanation:

A. Correct. This is the correct seal failure diagnosis per SO23-13-6, Attachment 1.

B. Incorrect. Plausible because the Middle Seal is at RCS pressure, however, this is because the Lower Seal has failed. If this were true the Middle and Upper Seals would both be indicating 1162.5 psia.

C. Incorrect. Plausible because the Middle Seal is at RCS pressure it could be thought that both the Lower and Middle Seals have failed, however, if this were true the Upper Seal would be indicating 2250 psia.

D. Incorrect. Plausible because it could be thought that with the Middle Seal indicating RCS pressure it is the only seal currently working, however, if this were true the Upper Seal would be indicating 75 psia.

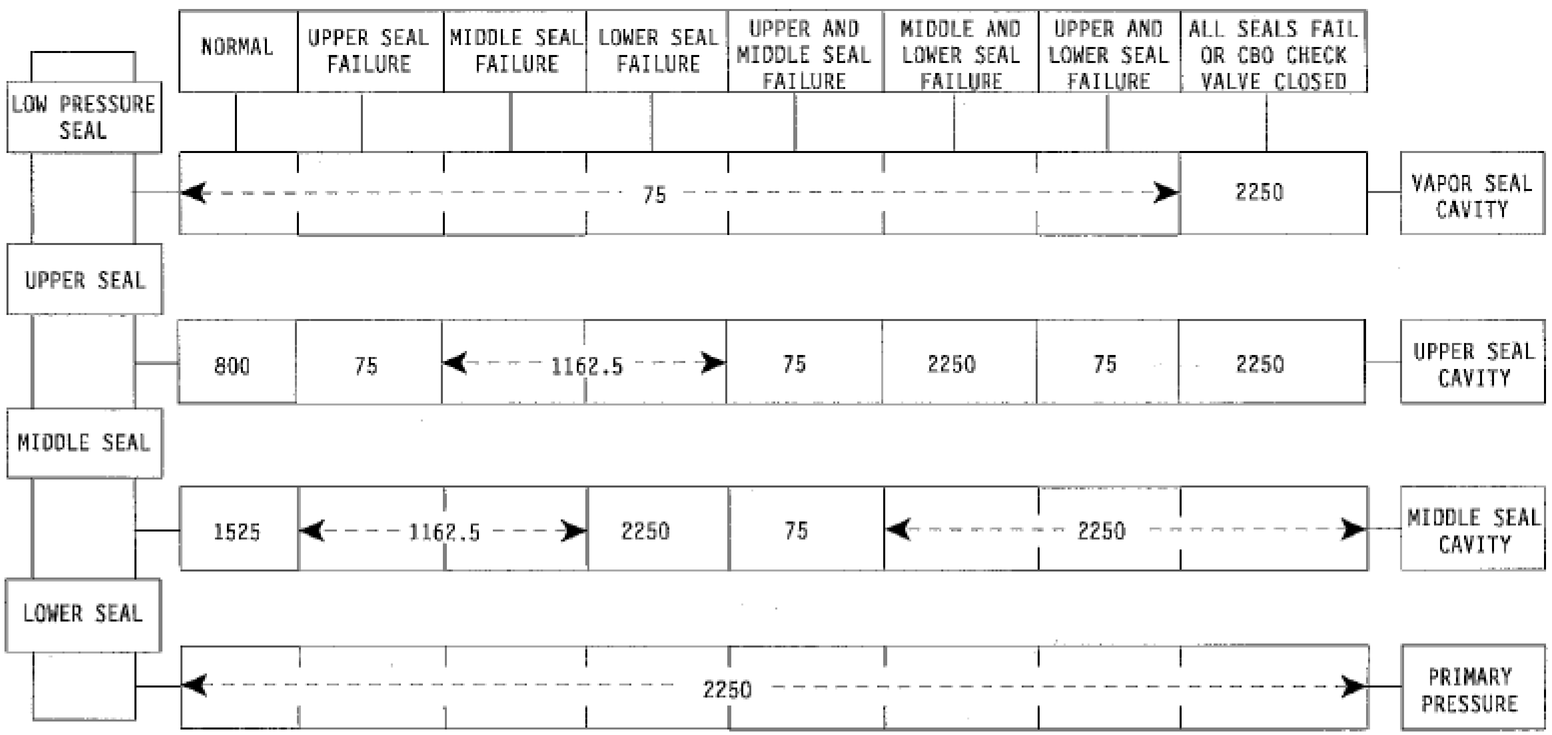
Technical Reference(s)	SO23-13-6, Attachment 1	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55452	Using SO23-13-6, Reactor Coolant Pump Seal Failure, DESCRIBE: The expected plant response for each Step.

Question Source:	Bank #	126991	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	3, 5	
	55.43		

Comments / Reference: From SO23-13-6, Attachment 1	Revision # 5



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	025 AA2.05	
Importance Rating	3.1	

Loss of RHR System: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Limitations on LPI flow and temperature rates of change

Proposed Question: Common 44

Given the following conditions:

- Unit 2 entered MODE 5 two (2) days ago for a Refueling Outage.
- Train A Shutdown Cooling Pump tripped on overcurrent.
- Train B Shutdown Cooling Pump is being placed in service.
- The Shutdown Cooling System has been secured for approximately 1 hour.

Which ONE (1) of the following identifies the:

- MINIMUM Shutdown Cooling System flow rate and
- MAXIMUM expected Shutdown Cooling System heatup rate when restoring the system to operation?

MINIMUM Shutdown Cooling System flow rate is __________ and MAXIMUM expected Shutdown Cooling System heatup rate is __________.

A. 2000 gpm
5°F per hour

B. 2000 gpm
50°F per hour

C. 2500 gpm
5°F per hour

D. 2500 gpm
50°F per hour

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that a 5°F/hr heatup rate is appropriate with the Unit in MODE 5, however, it has been demonstrated that a heatup rate can reach 50°F/hr.
B. Incorrect. Plausible because the expected heatup rate is correct, however, 2000 gpm is less than the Technical Specification minimum of 2200 gpm.
C. Incorrect. Plausible because the flow rate is correct, however, when securing Shutdown Cooling in this condition one would expect to see a heatup rate of up to 50°F/nr.
D. Correct. This is the minimum administrative limit for flow and maximum expected heatup rate when restoring the Shutdown Cooling System to service after temporary termination.

Technical Reference(s): SO23-3-2.6, Step 6.2.2 Caution; SO23-3-2.6, L&S 2.1 & 2.2; SO23-5-1.3, L&S 4.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52628 — Given an operational condition addressed by a procedural precaution, limitation, of administrative requirement, STATE the limiting condition and the basis for that limiting condition for the Shutdown Cooling System.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7, 10
55.43 ______

Comments / Reference: From SO23-3-2.6, Step 6.2.2 Caution	Revision # 26

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6.0 PROCEDURE (Continued)

6.2.2 Maintain **SHUTDOWN COOLING FLOW** within the following guidelines: (Ref. 2.2.1, 2.2.4 and 2.2.14) [LS-2.1 through LS-2.3, Attachment 7]

CAUTION

To ensure adequate core cooling, SDC flow **shall** be maintained at least 2500 gpm in all modes. (LS-2.2)

SINGLE LPSI PUMP OPERATION

RCS LEVEL	MAXIMUM FLOW (gpm)	RCS LEVEL	MAXIMUM FLOW (gpm)
> TOH	5500	24"	3700
		23"	3500
28" to TOH	4400	22"	3300
27"	4300	21"	3100
26"	4100	20"	2900
25"	3900	19"	2700

NOTE: Alarms should be set -/+ 200 gpm of actual flow (CFMS Page 314 is preferred). MAXIMUM FLOW values are administrative guidelines which are 200 gpm below actual flow limits to ensure the LPSI Pump will not exceed any limits.

OTHER SDC PUMP COMBINATIONS

SDC PUMP COMBINATIONS	RCS LEVEL CONDITION	ALLOWABLE RANGE (gpm)	ALARM SETPOINTS LO/HI
2 LPSI	Above TOH	5200 - 8100	± 200 gpm
1 CS	≥ 23' above the flange (38% PZR Level)	2500 - 2750 [4]	2550/2700
2 CS	≥ 23' above the flange (38% PZR Level)	2500 - 3875 [4] (< 1970 per Pump)	2550/3800
1 or 2 CS	< 23' above the flange (38% PZR Level)	2500 - 2750 [4]	2550/2700
CS and LPSI SDC/SFP Cooling Combinations	Refueling Water Level ≥ 23" (Transfer Tube open, SDC Pumps providing SDC <u>and</u> SFP cooling)	Refer to SO23-3-2.6.1 for guidance; individual Pump limitations apply	

[4] 2400 gpm lower limit is allowed if using CFMS, due to a lower TLU. (LS-2.2)

Comments / Reference: From SO23-3-2.6, L&S 2.1 & 2.2	Revision # 26

NUCLEAR ORGANIZATION
UNITS 2 AND 3

OPERATING INSTRUCTION
REVISION 26
ATTACHMENT 16

SO23-3-2.6
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SHUTDOWN COOLING SYSTEM LIMITATIONS AND SPECIFICS

OBJECTIVE

To provide a list of system/component limitations and specific operational details related to the steps in this procedure. Although the information presented here is not necessary to perform an evolution, it does provide supplementary information to enhance understanding and increase awareness. Some of this information may also be considered for Pre-job Brief subjects. Appropriate steps in this procedure will reference this attachment, for example ***(LS-2.2)*** for *Limitations and Specifics Item No. 2.2.*

Verify this document is current by checking a controlled copy or by using the method described in SO123-VI-0.9.

1.0 SDCS Pressure and Temperature Limitations

1.1 LIMIT: The LPSI System, SDCS and interconnected piping shall be maintained within the following RCS pressure and temperature limits (Not applicable for post-accident containment): (Ref. 2.1.3, 2.2.1, and 2.2.11)

- Pressure using Main Control Board instruments: ≤ 364 psia (both pressure instruments indicating within 38 psi)
- Pressure using PCS/QSPDS instruments: ≤ 370 psia (the largest ΔP between instruments within 24 psi)
- Temperature of ≤ 340°F

2.0 SDCS Flow Limitations

2.1 LIMIT: Shutdown Cooling flow shall be greater than or equal to 2200 gpm. (Tech. Spec. LCO 3.9.4, 3.9.5)

2.2 LIMIT: LPSI and SDCHX Flow Limits, as follows:

SDC CONFIGURATION	LIMITS	REFERENCES/NOTES
2 SDCHXs & 2 LPSI Pumps SDC flow range	5000-8300 gpm	Ref. 2.2.1 & AR 970400821
1 SDCHX maximum CCW flow		Refer to SO23-2-17 for limits based on system configuration.
1 SDCHX maximum SDC flow (both HXs required >5300 gpm)	≤ 5320 gpm	Ref. 2.2.1
1 LPSI Pump maximum flow	≤ 5500 gpm	Ref. 2.2.1
1 LPSI Pump minimum flow - Long Term (≥ 25" RCS Hot Leg)	> 3900 gpm	For Pump protection during long term use.
SDCS minimum flow: FI-0306	≥ 2500 gpm	LCO 3.9.4, 3.9.5 & Ref. 2.2.14
SDCS minimum flow: F306 (CFMS)	≥ 2400 gpm	LCO 3.9.4, 3.9.5 & Ref. 2.2.14

Comments / Reference: From SO23-5-1.3, L&S 4.4	Revision # 32

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3.0 SAFE SHUTDOWN OPERABILITY REQUIREMENTS

3.1 LIMIT: Exiting a 10CFR50 Appendix R Action Statement by entering a non-applicable Mode prior to the end of the 60-day Action will allow for termination of the compensatory measures. However, reentry into an applicable Mode without restoring the specific component/feature to OPERABLE status will cause the Action Statement to resume at the point in the 60-day period when it was exited.

3.2 LIMIT: Steam Generator Pressure Indication Channels A and B are required for Safe Shutdown. (Tech. Spec. LCS 3.7.113-1)

3.3 Steam Generator Pressure Indication may still be used to meet Safe Shutdown requirements while in bypass or tripped.

3.4 When requesting I&C to make "Live" Channel A or B Steam Generator Pressure Instruments, then any other "Live" pressure channel other than A or B will be simulated per the I&C procedure.

4.0 RCS HEATUP LIMITATIONS

4.1 The RCS **HEATUP** Administrative guideline is 50°F/hr. when Tc ≥70°F (this guideline is more conservative than Tech. Spec. LCO 3.4.3 and LCS 3.4.103 limit of 60°F/hour).

4.2 **UNIT 2 ONLY:** Due to a high rate of S/G tube failure, RCS nominal heatup is 20°F/HR when the RCS is >340°F. This is based on a theory that S/G sleeve collapse is caused by differential expansion rates between the sleeves and the tubes. When water is trapped in this gap, it causes deformation of the sleeve with resultant flow loss. This expansion rate is greater at higher temperatures. Consequently, controlling the heatup rate to a nominal 20°F/HR when the RCS is >340°F should reduce this failure rate. (AR 060102028)

4.3 An RCS Heatup or Cooldown evolution is defined as a planned change in RCS temperature from one plateau to another of ≥10°F total change. Unplanned (transient) temperature changes should be evaluated against T.S. Limits. (T.S. Bases SR 3.4.3.1)

4.3.1 Logging and Plotting is required for all heatup/cooldown evolutions.

4.4 Initial Heatup rate (during first 5 minutes) after securing SDC could exceed 50°F/hr; however, no experience in any outage has exceeded 50°F/hr after 15 minutes.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	026 AK3.02	
Importance Rating	3.6	

Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of ESFAS

Proposed Question: Common 45

Which ONE (1) of the following is the reason that the Containment Emergency Cooling Unit (ECU) Component Cooling Water (CCW) Inlet Valves are required to remain OPEN in MODES 1 through 4 even though they receive an open signal when a Containment Cooling Actuation Signal is initiated?

A. Assist Normal Containment Cooling during an accident.

B. Ensure sufficient Containment Cooling in the event Containment Spray fails to actuate.

C. Prevent thermally locking the ECU CCW Outlet Valves if Containment temperature rises during an accident.

D. Prevent Component Cooling Water Pumps from operating in a reduced flow condition.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that this system remains in operation until a Containment Isolation Actuation Signal is received, however, Normal Chill Water isolates on an SIAS.

B. Incorrect. The Containment Emergency Cooling Units provide the necessary protection in the event Containment Spray fails to actuate, however, that is not the reason why the CCW Inlet Valves are left open.

C. Correct. Per L&S 5.7 in SO23-2-17, there is a concern that the ECU Outlet Valves are susceptible to hydraulic locking if the CCW Inlet Valves are left closed in MODES 1 through 4.

D. Incorrect. Plausible because Component Cooling Water system alignment requires adequate flow to the pump, however, in an accident condition there is sufficient flow.

Technical Reference(s) SO23-3-2.22, Attachment 4; SO23-2-17, L&S 2.16 and 5.7 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 79748	Given an operational condition of the CSS, and SIS conditions addressed by a procedural precaution, limitation, or administrative requirement, STATE the limiting condition and the basis for that limiting condition.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 7, 8
	55.43 ______

Comments / Reference: From SO23-3-2.22, Attachment 4	Revision # 16

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2.0 PROCEDURE (Continued) — PERF. BY INITIALS

2.2 If a SIAS actuation occurs, then ENSURE SO23-3-3.23, Attachment for A. C. Sources Verification, is completed for both Units within one hour, due to the 1E 4kv Bus Tie Breaker Controls being in MANUAL. (Mark N/A if SIAS did not occur.) [Tech. Spec. LCO 3.8.1] ________

2.3 ENSURE the associated Emergency Chiller(s) have CCW flow. (Mark N/A if this is a spurious actuation.) ________

2.4 VERIFY the associated Emergency Chiller(s) operating properly per SO23-1-3.1, Section for Verification of Emergency Chiller Steady State Parameters. (Mark N/A if this is a spurious actuation or Emergency Chiller will be in-service < 30 minutes.) ________

2.5 **VERIFY SIAS/CCAS Train A component actuation at CR-57**:

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.5.1	HV-6370	CCW to Containment ECU E-399 Isolation	[1]	OPEN	________
2.5.2	HV-6371	CCW from Containment ECU E-399 Isolation	[1]	OPEN	________
2.5.3	HV-6366	CCW to Containment ECU E-401 Isolation	[1]	OPEN	________
2.5.4	HV-6367	CCW from Containment ECU E-401 Isolation	[1]	OPEN	________

Comments / Reference: From SO23-3-2.22, Attachment 4	Revision # 16

NUCLEAR ORGANIZATION UNITS 2 AND 3 — OPERATING INSTRUCTION REVISION 16 ATTACHMENT 4 — SO23-3-2.22 PAGE 42 OF 146

2.0 PROCEDURE (Continued)

2.4 **VERIFY SIAS/CCAS Train A component actuation at CR-57**: (Continued)

STEP	NUMBER OF COMPONENT	NOUN NAME	NOTE	REQUIRED POSITION	PERF. BY INITIALS
2.5.12	HV-7802	Containment Rad Mon Train A Isolation	[2][3]	CLOSED	________
2.5.13	HV-9920	Normal Chilled Water to Containment Isolation	[2][3]	CLOSED	________
2.5.14	HV-9921	Normal Chilled Water to Containment Isolation	[2][3]	CLOSED	________

Comments / Reference: From SO23-2-17, L&S 2.16	Revision # 27

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CCW SYSTEM LIMITATIONS AND SPECIFICS (Continued)

2.0 **SYSTEM GUIDELINES** (Continued)

2.15 The following table provides projected CCW flowrates through a SDCHX for various system alignments (actual flows may vary), however, being outside of the normal operating range is not an issue of operability, rather an issue of long term system reliability (excessive flow through a SDCHX may result in a mechanical failure of the heat exchanger):

CCW SYSTEM ALIGNMENTS AND SDCHX FLOWRATES

CCW FLOWPATH ALIGNMENTS				SDCHX Flowrate (gpm)	Condition
Noncritical Loop		ECU Return Valves (2)	Emergency Chiller		
ONE SFPHX	RCPs/CEDMs				
OPEN	OPEN	OPEN	OPEN	4900	CCW Pump Runout
OPEN	OPEN	OPEN	CLOSED	5200	
OPEN	CLOSED	OPEN	OPEN	5670	Normal Operating Range
OPEN	CLOSED	OPEN	CLOSED	5960	
CLOSED	CLOSED	OPEN	OPEN	6550	
OPEN	OPEN	CLOSED	CLOSED	6650	
OPEN	CLOSED	CLOSED	OPEN	6830	
CLOSED	CLOSED	OPEN	CLOSED	6970	
OPEN	CLOSED	CLOSED	CLOSED	7280	
CLOSED	CLOSED	ONLY 1 OPEN	CLOSED	7600	
CLOSED	CLOSED	CLOSED	OPEN	8000	Excessive SDCHX Flow
CLOSED	CLOSED	CLOSED	CLOSED	8500	

[1] Flowrates given assuming minor flowpaths are open, e.g., PACU, HPSI/LPSI/CS Pump Coolers.)

Comments / Reference: From SO23-2-17, L&S 5.7	Revision # 27

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CCW SYSTEM LIMITATIONS AND SPECIFICS (Continued)

5.0 **VALVE GUIDELINES** (Continued)

5.7 Containment ECU CCW Inlet Valves (HV-6366, HV-6368, HV-6370, HV-6372) must remain open in Modes 1, 2, 3, and 4 to minimize the differential pressure across the corresponding ECU Outlet Valve (HV-6367, HV-6369, HV-6371, HV-6373). Minimizing the differential pressure across the outlet valve ensures that it can be opened when required. If an ECU supply valve is closed (for surveillance testing, SDC flow balancing, etc.), then the associated ECU is inoperable. (AR 971201623, DBD 400, Sect. 4.4, Tech. Spec. 3.6.6.2 and Ref. 2.3.1.14)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AK1.02	
	Importance Rating	2.8	

Pressurizer Pressure Control System Malfunction: Knowledge of the operational implications of the following concepts as they apply to the Pressurizer Pressure Control Malfunctions: Expansion of liquids as temperature increases

Proposed Question: Common 46

Given the following conditions:

- Unit 2 is at 100% power with Reactor Coolant System T_{COLD} on program.
- Pressurizer pressure control setpoint is 2250 psia.
- Pressurizer pressure is 2250 psia.
- Both Pressurizer Proportional Heaters are in service.
- All Pressurizer Backup Heaters are secured.
- Both Pressurizer Spray Valves are closed.
- One (1) Charging Pump is in service.
- A step change in Turbine Control Valve position results in the following:
  - Reactor Coolant System T_{AVE} rises 5°F.
  - Pressurizer level rises 5%.
  - Pressurizer pressure rises 75 psia.

Which ONE (1) of the following would be indicative of initial plant response to the above conditions?

A. Letdown flow will rise to MAXIMUM.

B. The running Charging Pump will STOP.

C. Letdown flow will isolate.

D. Both Pressurizer Spray Valves will go full OPEN.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because Letdown flow will increase, however, it would take a +10.6% deviation for Letdown to be at MAXIMUM flow.
B. Incorrect. Plausible because the Backup Charging Pumps will STOP with a + 4% deviation, however, the Charging Pump in AUTO will continue to run.
C. Incorrect. Plausible if thought that this condition would isolate flow.
D. Correct. With a 50 psia rise in Pressurizer pressure, both Pressurizer Spray Valves go full OPEN.

Technical Reference(s) SD-SO23-360, Figures 3-9 & 3-16 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56419 DESCRIBE the cause/effect relationships associated with the Pressurizer Pressure and Level Control System and an increasing or decreasing Pressurizer pressure and/or level.

Question Source: Bank # 127315
Modified Bank # ____ (Note changes or attach parent)
New ____

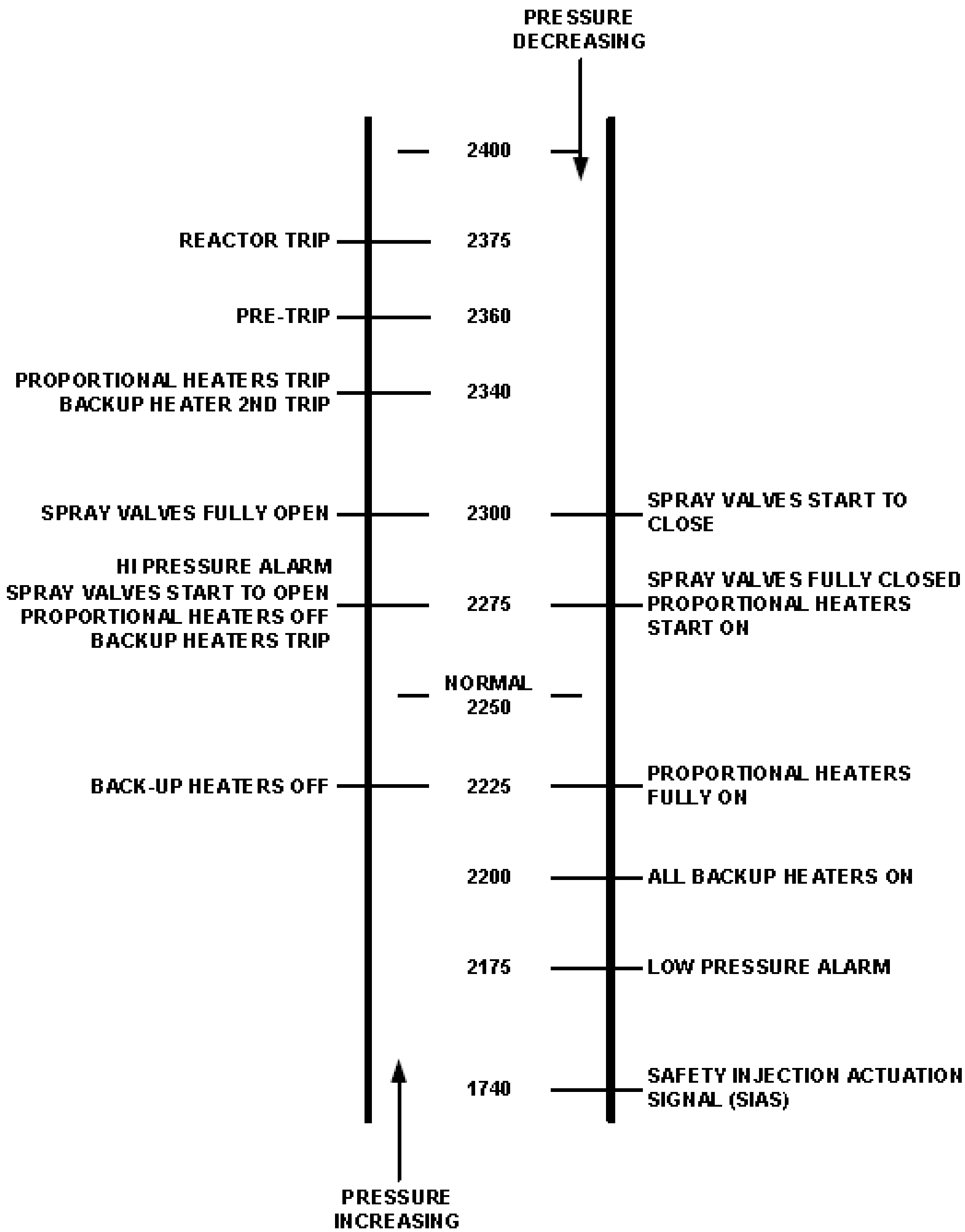
Question History: Last NRC Exam ____

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

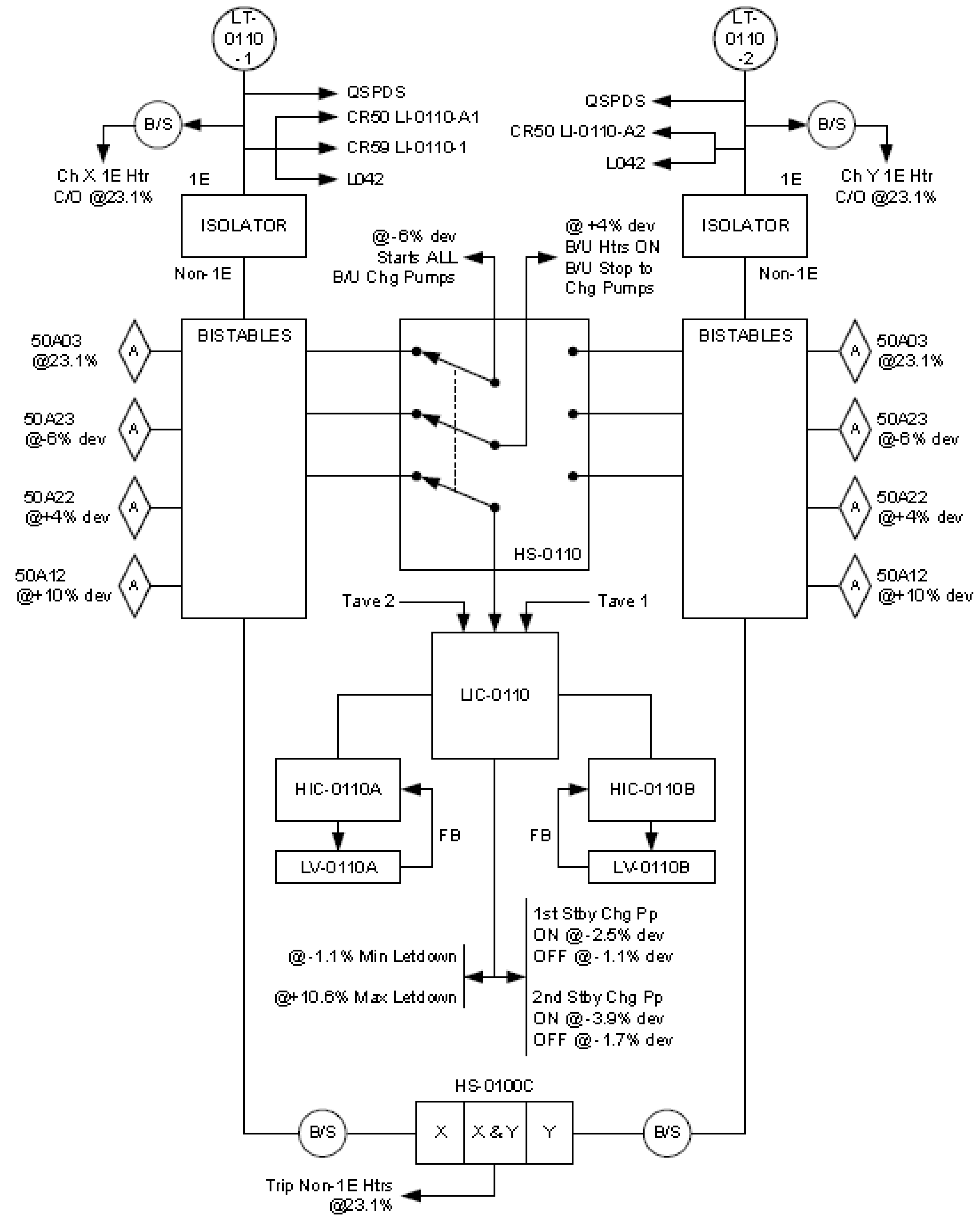
10 CFR Part 55 Content: 55.41 5, 7
55.43 ____

Comments / Reference: From SD-SO23-360, Figure 3-9	Revision # 17

FIGURE III-9: PRESSURIZER PRESSURE CONTROL SETPOINTS



Comments / Reference: From SD-SO23-360, Figure 3-16	Revision # 17



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	029 EA2.07	
Importance Rating	4.2	

ATWS: Ability to determine or interpret the following as they apply to a ATWS: Reactor trip breaker indicating lights

Proposed Question: Common 47

Given the following conditions on Unit 2 at 100% power:

- Several Reactor Protection System annunciators are received.
- Several Reactor Trip Circuit Breakers have opened.
- The Control Element Assemblies (CEAs) have NOT inserted.

If the plant is still at power with all CEAs fully withdrawn, which of the following OPEN trip breakers would indicate an Anticipated Transient without Scram (ATWS) condition exists?

Reactor Trip Circuit Breakers __________ indicate open.

A. 1, 2, 5, and 6

B. 1, 3, 5, and 7

C. 3, 4, 7, and 8

D. 1, 2, 3, and 4

Proposed Answer: B

Explanation:

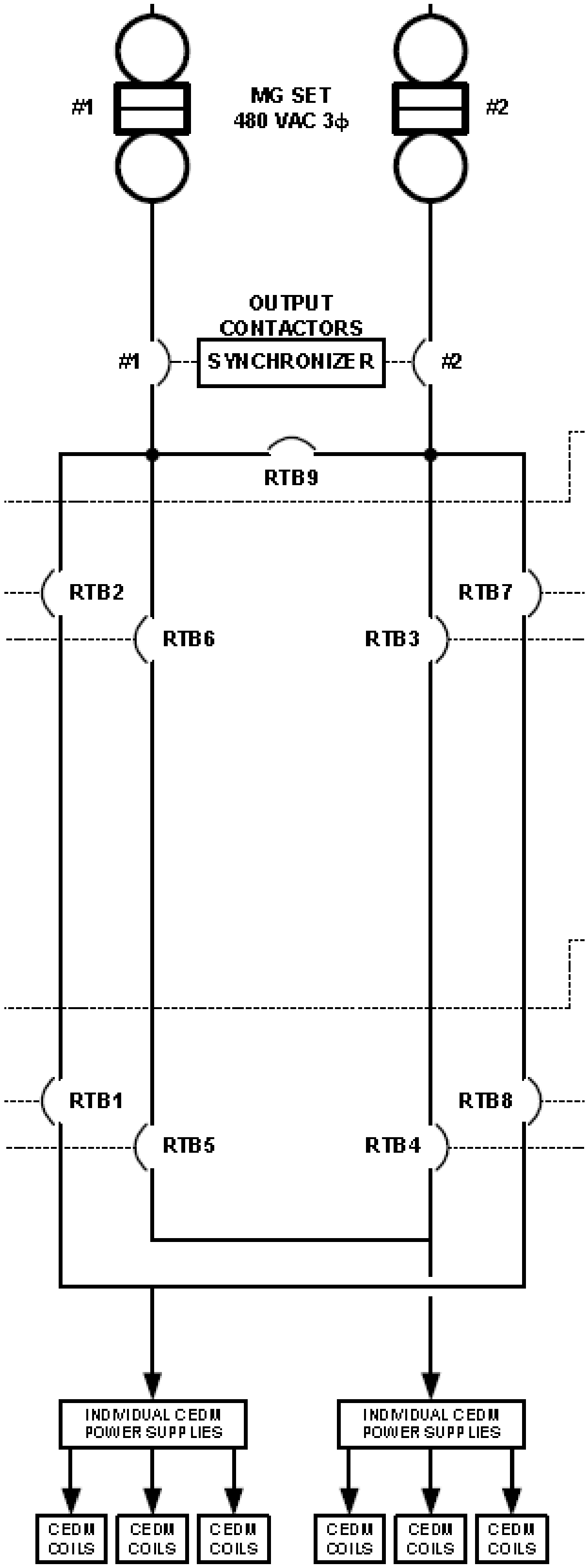
A. Incorrect. Plausible because it could be thought that this combination of trip breakers open should result in all rods being deenergized.
B. Correct. This combination of breakers being open should result in all rods being deenergized and the Reactor should trip.
C. Incorrect. Plausible because it could be thought that this combination of trip breakers open should result in all rods being deenergized.
D. Incorrect. Plausible because it could be thought that this combination of trip breakers open should result in all rods being deenergized.

Technical Reference(s) SD-SO23-710, Figure 2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55846 / 56627	IDENTIFY the plant parameter, including trip value, used to actuate the ATWS/DSS. DESCRIBE the operation of the Plant Protection System components and instrumentation, including function, location, design basis, interlocks, setpoints, special features and power supplies, where applicable.		
Question Source:	Bank #	127053	(Changed Distractor C)
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2005A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	6	
	55.43		

Comments / Reference: From SD-SO23-710, Figure 2	Revision # 7



Comments / Reference: From Exam Bank #127053	Revision # 10/20/06
Distractor C: Replaced RTCB #9 with RTCB #8 for better plausibility.	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	038 EK1.01	
Importance Rating	3.1	

Steam Generator Tube Rupture: Knowledge of the operational implications of the following concepts as they apply to the SGTR: Use of steam tables

Proposed Question: Common 48

Given the following conditions:

- A Steam Generator Tube Rupture has occurred and both Steam Generators are being used to lower Reactor Coolant System temperature and pressure.
- Reactor Coolant System T_{HOT} was initially lowered to less than 530°F.
- The ruptured Steam Generator has been isolated.
- Step 12 of SO23-12-4, Steam Generator Tube Rupture for lowering Pressurizer pressure is being implemented.
- Ruptured Steam Generator pressure is currently 800 psia.

Which ONE (1) of the following is the HIGHEST acceptable temperature for RCS T_{HOT} when lowering Pressurizer pressure to within 50 psia of ruptured Steam Generator pressure while maintaining at least 20°F subcooling?

A. 460°F

B. 480°F

C. 500°F

D. 520°F

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that it was desirable to lower temperature to the point where subcooled margin was increased.
B. Incorrect. Plausible because it could be thought that it was desirable to lower temperature to the point where subcooled margin was increased.
C. Correct. With saturation pressure for 850 psia at 525°F, a target temperature of 500°F would still allow for sufficient subcooled margin.
D. Incorrect. Plausible because this temperature corresponds to a value close to 850 psia, however, it does not allow for any subcooled margin.

Technical Reference(s) SO23-12-4, Step 12 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: 52660
Given plant conditions, PREDICT and EXPLAIN the response of major plant systems, equipment and parameters to a steam generator tube rupture.

Question Source:
Bank #
Modified Bank # 127066 (Note changes or attach parent)
New

Question History: Last NRC Exam SONGS 2005A

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

10 CFR Part 55 Content:
55.41 10, 14
55.43

Comments / Reference: From SO23-12-4, Step 12	Revision # 21

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STEAM GENERATOR TUBE RUPTURE

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

12 INITIATE Lowering PZR Pressure:

NOTE

SGTR depressurization strategy should be to reduce RCS pressure while maintaining RCP NPSH T_C requirements. This strategy should continue until RCS pressure is within 50 PSI of the ruptured S/G pressure or S/G level is not rising.

CAUTION

Keeping RCS pressure higher than S/G pressure is preferred to minimize RCS dilution due to backflow unless backflow is intended.

CAUTION

IF uncontrolled S/G level rise is occurring, THEN reducing RCS pressure to less than 1000 PSIA takes priority over maintaining RCP NPSH or 20°F Core Exit Saturation Margin. In this case stopping RCPs should be evaluated.

a. MAINTAIN RCS pressure requirements of SO23-12-11, Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS:

1) ESTABLISH RCS pressure:

– low in allowable band for SGTR (approximately equal to ruptured S/G pressure).

AND

– greater than RCP NPSH curve with RCPs running.

AND

– less than 160°F curve.

a. 1) IF RCP NPSH requirements
– NOT satisfied,

THEN

a) STOP all RCPs

b) INITIATE Auxiliary Spray

OR

INITIATE FS-32, ESTABLISH Manual Auxiliary Spray.

2) IF all RCPs stopped,

THEN MAINTAIN RCS Pressure above 20°F Saturation Margin curve of SO23-12-11, Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS.

Comments / Reference: From Exam Bank #127066	Revision # 10/20/06
A steam generator (SG) tube rupture has occurred and both SGs are being used to lower RCS temperature and pressure. All the RCPs have been stopped due to a saturation margin of 20°F. Step 12 (Lowering Pzr Pressure) of SO23-12-4, "Steam Generator Tube Rupture," is being implemented. If ruptured SG pressure is currently 600 psia, then an acceptable initial target temperature for RCS T_{hot} would be: A.525°F **B.475°F** C.425°F D.375°F	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	055 G 2.2.44	
Importance Rating	4.2	

Station Blackout: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions

Proposed Question: Common 49

Given the following conditions exist on Unit 3 two (2) hours after initiation of a Station Blackout:

- Reactor Coolant System pressure is 1850 psia.
- Core Exit Saturation Margin is 19°F.
- Reactor Coolant System T_{HOT} is 555°F and stable.
- Reactor Coolant System T_{COLD} is 535°F and stable.
- Representative Core Exit Thermocouple is 605°F.
- Reactor Vessel Level indicates 100%.
- Atmospheric Dump Valves are in service.
- P140, Turbine Driven Auxiliary Feedwater Pump is in service.

Which ONE (1) of the following describes the status of Core Heat Removal and required actions?

A. Natural Circulation Criteria are met.
MAINTAIN Atmospheric Dump Valve steaming rate to ensure subcooling between 10°F and 20°F.

B. Natural Circulation Criteria are NOT met.
REDUCE the steaming rate to MINIMIZE Reactor Coolant System inventory shrinkage and allow the Pressurizer pressure rise to recover subcooling.

C. Natural Circulation Criteria are NOT met.
INCREASE Atmospheric Dump Valve steaming rate to raise subcooling to greater than 20°F and lower REP_{CET} - T_{HOT} ΔT.

D. Two Phase Heat Removal Criteria are met.
RAISE Steam Generator narrow range levels to greater than 80% to MAXIMIZE heat removal capability.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that Natural Circulation is met, however, the correct minimum Core Exit Saturation Margin is 20°F.
B. Incorrect. Plausible because Natural Circulation is <u>not</u> met and it could be thought that the RCS shrinkage aspect could be the overriding factor for recovering subcooling.
C. Correct. Given the conditions listed, these are the correct actions per SO23-12-8.
D. Incorrect. Plausible because given the conditions listed, two phase heat removal is in progress per the RNO column of Step 10 (REP CET minus Thot is > 16°F). Heat removal is maximized by raising Steam Generator level up to, but not greater than, 80% narrow range.

Technical Reference(s) SO23-12-8, Step 10 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53297 / 53125

EXPLAIN the observable conditions used to verify the establishment and maintenance of single phase natural circulation.

Explain the operational impact of post trip Reactor Coolant System natural circulation relative to: RCS cooldown rate.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Comments / Reference: From SO23-12-8, Step 10	Revision # 20

STATION BLACKOUT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Low flow during Natural Circulation slows RCS response to temperature changes. Loop transit time rises to between 5 minutes and 10 minutes.

10 MAINTAIN Stable RCS Conditions With Natural Circulation:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. ENSURE AFW flow to at least one S/G.	a. GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.
b. ENSURE ADV – available on S/G with AFW flow.	
c. MAINTAIN available S/G(s) levels – between 40% NR and 80% NR.	

Comments / Reference: From SO23-12-8, Step 10	Revision # 20

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REVISION 20

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STATION BLACKOUT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE

10 MAINTAIN Stable RCS Conditions With Natural Circulation: (Continued)

d. OPERATE AFW and available ADVs to maintain Core Exit Saturation Margin – greater than or equal to 20°F:

QSPDS page 611.

e. VERIFY operating loop ΔT (T_H - T_C) – less than 58°F.

f. VERIFY T_H and T_C – NOT rising.

g. VERIFY operating loop T_H and REP CET – within 16°F.

QSPDS page 611
CFMS page 311.

h. VERIFY Reactor Vessel level – greater than or equal to 100% (Plenum):

QSPDS page 622
CFMS page 312
SO23-12-11, Attachment 4.

RESPONSE NOT OBTAINED

o IF single phase natural circulation per criteria of steps d. through h. – NOT satisfied,

THEN

- MAINTAIN two-phase heat removal:

  a) MAXIMIZE available S/G levels – less than 80% NR.

  b) OPERATE ADVs to raise available S/G steaming rates.

  c) RAISE Core Exit Saturation Margin – greater than or equal to 20°F:

  QSPDS page 611
  CFMS page 311.

- MONITOR REP CET temperature:

  QSPDS page 611
  CFMS page 311.

- IF REP CET temperature – greater than 700°F,

  THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY*

  AND

  INITIATE SO23-12-9, Attachment FR-5, RECOVERY - HEAT REMOVAL.

- IF after 2 hours of Station Blackout Core Exit Saturation Margin – lowering to 20°F,

  THEN GO TO step 13.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	056 AA2.20	
Importance Rating	3.9	

Loss of Offsite Power: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: AFW flow indicator

Proposed Question: Common 50

Given the following conditions on a trip from 100% power EOC:

- A Loss of Offsite Power has occurred.
- Train A Emergency Diesel Generator has failed to start.
- Emergency Feedwater Actuation Signals (EFAS) 1 and 2 have both actuated.
- HV-8200, Main Steam Supply from Steam Generator E089 to P140, Turbine Driven Auxiliary Feedwater Pump failed CLOSED.

Which ONE (1) of the following is the expected Auxiliary Feedwater flow indication to each Steam Generator as read on the CR-52 lumigraph?

Auxiliary Feedwater flow indication for Steam Generator E088 is indicating _____ gpm;
Auxiliary Feedwater flow indication for Steam Generator E089 is indicating _____ gpm.

A. 500
200

B. 500
500

C. 800
200

D. 800
500

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that Steam Generator E089 is receiving reduced low from P140 because its associated Main Steam Supply Valve is not open and Steam Generator E088 is receiving flow from P504 and reduced flow from P140.
B. Incorrect. Plausible because the flow value for Steam Generator E089 is correct, however, Steam Generator E088 is receiving flow from P504 and P140.
C. Incorrect. Plausible because Steam Generator E088 is receiving flow from P504 and P140, however, Steam Generator E089 is receiving full flow from P140.
D. Correct. Steam Generator E088 is receiving flow from P504 and P140. Steam Generator E089 is receiving flow from P140 only. Flow values were validated on the Simulator.

Technical Reference(s) SD-SO23-780, Figure 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52728 — DESCRIBE the instrumentation used to monitor the operation of the Auxiliary Feedwater System, including the name, function, sensing points, normal values for the parameters being measured and location of each instrument.

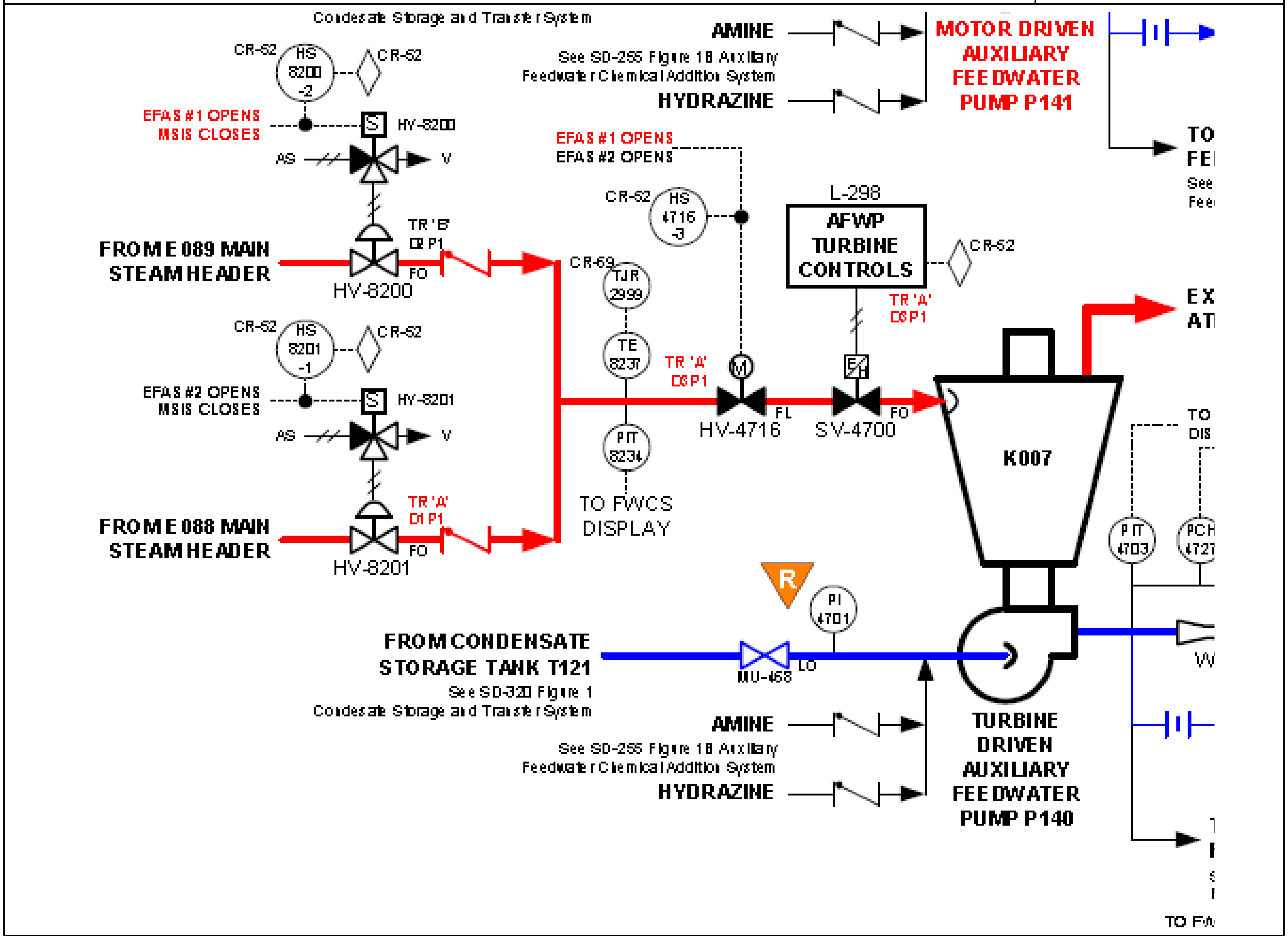
Question Source:
- Bank # ______
- Modified Bank # ______ (Note changes or attach parent)
- New X

Question History: Last NRC Exam ______

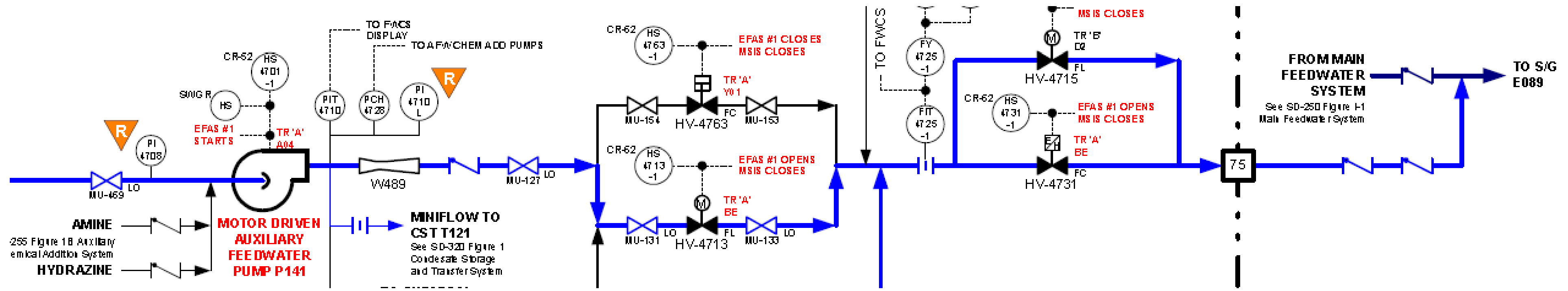
Question Cognitive Level:
- Memory or Fundamental Knowledge ______
- Comprehension or Analysis X

10 CFR Part 55 Content:
- 55.41 5, 7
- 55.43 ______

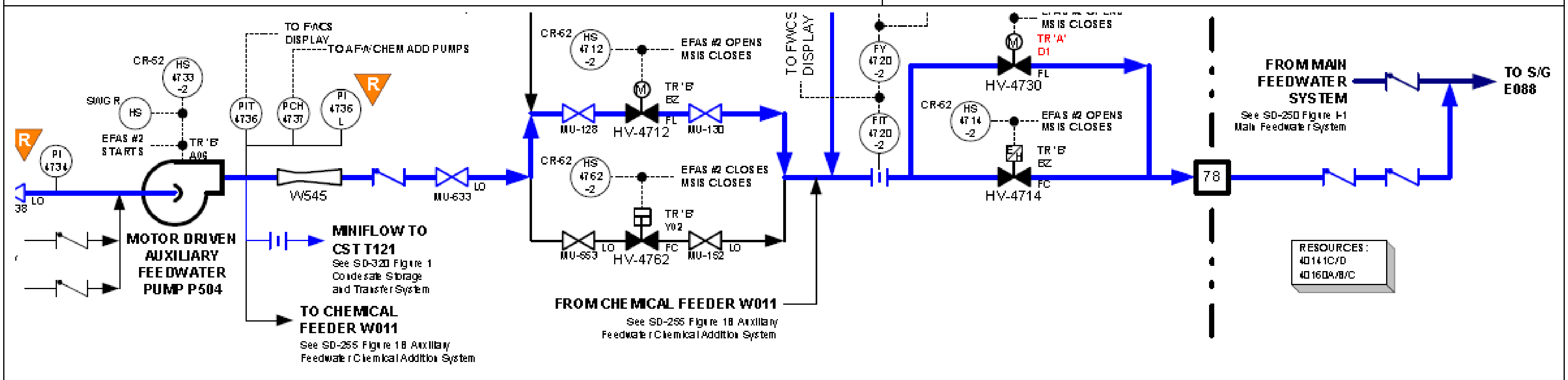
Comments / Reference: From SD-SO23-780, Figure 1	Revision # 10



Comments / Reference: From SD-SO23-780, Figure 1	Revision # 10



Comments / Reference: From SD-SO23-780, Figure 1	Revision # 10



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	057 AK3.01	
Importance Rating	4.1	

Loss of Vital AC Instrument Bus: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital AC electrical instrument bus

Proposed Question: Common 51

Given the following conditions:

- SO23-13-18, Reactor Protection System Failure/Loss of a Vital Bus is in progress.
- Vital AC Instrument Bus Y01 has been lost due to Inverter failure with the Unit operating at power.
- SO23-6-17, Class 1E 120 VAC Vital Bus Power Supplies System Operation is in progress to energize the Alternate Source.

Which ONE (1) of the following actions is required?

A. OPERATE Charging Pumps as necessary to control Pressurizer level.

B. ACTUATE Train B FHIS, TGIS, CRIS and CPIS due to Train A FHIS, TGIS, CRIS and CPIS actuation.

C. VERIFY SG wide/narrow range levels and pressures for Channel D auto transfer <u>into</u> service and Channel A auto bypass <u>out</u> <u>of</u> service on Feedwater DCS.

D. TRANSFER Pressurizer Level to Channel X to allow restoring the Pressurizer Level Controller to AUTO.

Proposed Answer: A

Explanation:

A. Correct. This action is required per SO23-13-18, Attachment 1.
B. Incorrect. Plausible because Train A FHIS, TGIS, CRIS and CPIS will actuate on a loss of Vital Bus Y01, however, the only action required is verification of proper alignment. Train B is not actuated.
C. Incorrect. Plausible because the Channel A Steam Generator levels and pressures must be bypassed on both Steam Generators, however, this action is performed manually by the operator.
D. Incorrect. Plausible because Pressurizer level must be transferred, however, Channel Y must be placed in service as it is powered from Vital Bus Y02.

Technical Reference(s) SO23-13-18, Attachment 1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55180	Per the Reactor Protection System Failure procedure, SO23-13-18, DESCRIBE: The expected plant response for each major step.		
Question Source:	Bank #		
	Modified Bank #	128051	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2008	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7, 10	
	55.43		

Comments / Reference: From SO23-13-18, Attachment 1	Revision # 8

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LOSS OF VITAL BUS Y01

CONTINUOUS USE

1.0 PREREQUISITES

None.

2.0 PROCEDURE

2.1 Review Tech. Spec. impacted LCO 3.4.9.b, 3.8.1, 3.8.2, 3.8.3, 3.8.7, 3.8.9 and 3.7.5.

2.2 **EFFECTS AND ACTIONS ON LOSS OF VITAL BUS Y01.**

2.2.1 Perform the following:

AFFECTED EQUIPMENT	INDICATIONS AND ASSOCIATED ACTIONS
.1 PPS A Status lights extinguished	☐ VERIFY protection system bistables NOT TRIPPED on PPS Channels B and D ROMs.
.2 Channels 1 & 3 Red ESFAS Function lights along the bottom of the ROM extinguished	☐ VERIFY all ESFAS function lights ILLUMINATED on PPS Channels B and D ROMs.
.3 Channel A Lumigraphs on CR56 extinguished	☐ VERIFY Safety Channel indications providing input to PPS Channels B, C, and D do not indicate that a Plant Protection Trip Setpoint has been exceeded.
.4 Charging Pumps P-190, P-191, and P-192.	☐ Operate Charging Pumps as necessary to control PZR level.
.5 PZR Pressure and Level Control	☐ ENSURE PZR Level Channel Y is SELECTED. ☐ If LIC-0110 is selected to setpoint LS1, then transfer Pressurizer level setpoint to LS2 per SO23-3-1.10, Attachment for Transferring Pressurizer Level and Pressure Controls. ☐ If an ACTUAL Pressurizer LO-LO level exists, then ENSURE all heaters DE-ENERGIZED. (AR 020900184)
.6 Vital Bus Inverter Y001 de-energized	☐ ENSURE SO23-6-17, Attachment for Re-energizing Vital Bus Y01 from the Alternate Source, in progress. (Tech. Spec. LCO 3.8.7 and LCO 3.8.9)

Comments / Reference: From SO23-13-18, Attachment 1	Revision # 8

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2.0 PROCEDURE (Continued)

2.2 **EFFECTS AND ACTIONS ON LOSS OF VITAL BUS Y01.** (Continued)

AFFECTED EQUIPMENT	INDICATIONS AND ASSOCIATED ACTIONS
.15 SIAS, CCAS, CIAS MSIS, CSAS, RAS	● Trip Paths 1 and 3 Actuated. ☐ VERIFY ESFAS alarms on CR-57 not ANNUNCIATED.
.16 SOES Alarm on Emergency Cooling System	● If Y01 is de-energized, then OES Alarm for Emergency Cooling System will alarm without delay only at the SOES (Sacramento) and not in the Control Room. ☐ INITIATE SO123-0-A7, Section for State Office of Emergency Services (SOES) Communication Guidelines.
.17 PT-0102-1, Wide Range PZR Pressure Transmitter	☐ IF PT-0102-1 is aligned as the WR PZR Pressure input to LIC-0103, PZR Compensated Level Indication, AND LIC-0103 is needed for level indication, THEN Per SRO direction, DISABLE the Inoperable PZR Pressure Input to LIC-0103 per SO23-3-1.10, Attachment for Transferring Pressurizer Level & Pressure Controls.
.18 PT-1013-1, S/G E-089 Pressure Transmitter PT-1023-1, S/G E-088 Pressure Transmitter LT-1113-1, S/G E-089 NR Level Transmitter LT-1123-1, S/G E-088 NR Level Transmitter LT-1115-1, S/G E-089 WR Level Transmitter LT-1125-1, S/G E-088 WR Level Transmitter	☐ At the Feedwater Digital Control System place in BYPASS the affected Channels for both S/G E-088 and S/G E-089: Channel "A" S/G PRESS Channel "A" NR LEVEL Channel "A" WR LEVEL per SO23-3-2.38, Section for Bypassing Selected Feedwater Signals.
.19 LT-5903-1, Diesel G002 Fuel Oil Storage Tank T-035 Level	● P-093 and P-096, Fuel Oil Transfer Pumps will not run in AUTO due to LT-5903-1 failing low. ● P-093 and/or P-096 can be operated in LOCAL from the Fire Isolation switch at L-160A.

Comments / Reference: From SONGS 2008 NRC Exam	Revision # N/A

Given the following conditions:

- Vital Bus Y-02 has been lost due to Inverter failure with the Unit operating at power.
- Vital Bus Y-02 has not been energized from the Alternate Source.

Which ONE (1) of the following actions is required and the reason for that action?

A. TRANSFER Pressurizer Level Setpoint to Setpoint LS1 to allow restoring the Pressurizer Level Controller to AUTO.

B. Manually ACTUATE Train A FHIS, TGIS, CRIS and CPIS due to Train B FHIS, TGIS, CRIS and CPIS actuation.

C. VERIFY Auxiliary Feedwater flow to Steam Generator E088 due to EFAS Trip Paths 2 and 4 actuation.

D. CLOSE the failed open Atmospheric Dump Valve (HV-8421) due to loss of the pressure input from the Main Steam pressure transmitter.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058 AK1.01	
	Importance Rating	2.8	

Loss of DC Power: Knowledge of the operational implications of the following concepts as they apply to Loss of DC power: Battery charger equipment and instrumentation

Proposed Question: Common 52

Given the following conditions:

- Unit 2 is operating at 100% power with all systems aligned for normal operations.
- DC Buses are aligned to the Dedicated Battery Chargers.
- A loss of 480 VAC Bus B06 occurs.

Which ONE (1) of the following describes the current status of DC Buses 2D2 and 2D4?

A. 2D2 and 2D4 DC Buses each continue to be supplied by their respective Dedicated Chargers and 1E Batteries.

Should be Charger here and below

B. D2 is being supplied only by the 1E Battery and D4 is still supplied by its Dedicated Chargers and 1E Battery.

C. D4 is being supplied only by the 1E Battery and D2 is still supplied by its Dedicated Chargers and 1E Battery.

D. 2D2 and 2D4 are being supplied only by their respective 1E Batteries.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the Dedicated Chargers were from an uninterruptible power source.
B. Incorrect. Plausible because it could be thought that only the Dedicated Charger to D2 was lost.
C. Incorrect. Plausible because could be thought that only the Dedicated Charger to D4 was lost.
D. Correct. Both Train B Dedicated Chargers are lost.

Technical Reference(s) SO23-13-26, Attachment 6
SD-SO23-140, Figure I-2B

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 80703 / 80682	DESCRIBE the configuration and operational characteristics of Non-IE 120 VAC and 125 VDC Power Supply System components. STATE the functions and design bases of the Non-IE 120 VAC and 125 VDC Power Supply System and its components.	
Question Source:	Bank #	128052
	Modified Bank #	(Note changes or attach parent)
	New	
Question History:	Last NRC Exam	SONGS 2008
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7, 10
	55.43	

Comments / Reference: From SO23-13-26, Attachment 6	Revision # 8

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ABNORMAL OPERATING INSTRUCTION REVISION 8 ATTACHMENT 6 | SO23-13-26 PAGE 28 OF 63

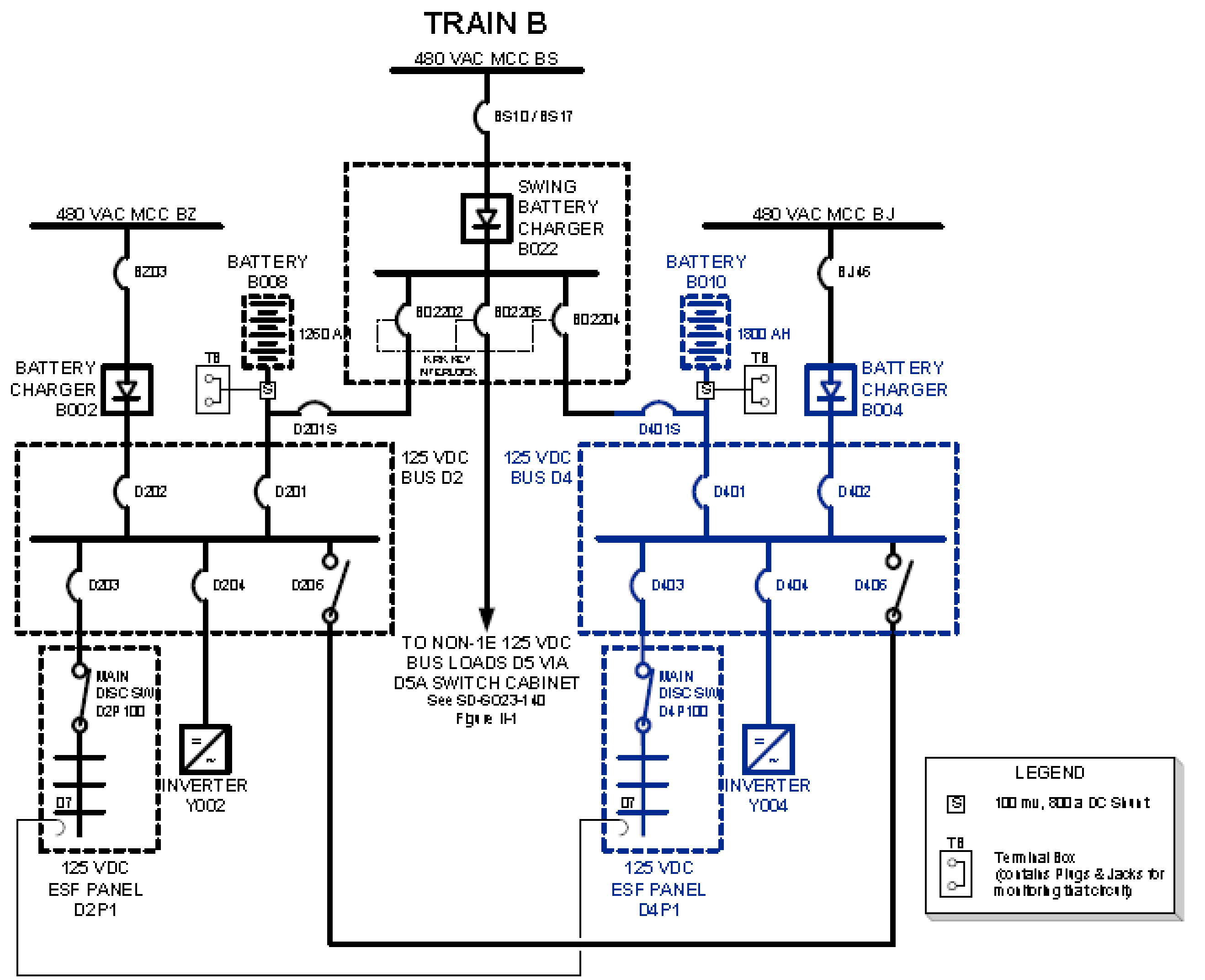
2.0 PROCEDURE (Continued)

2.1 **EQUIPMENT ACTIONS FOR LOSS OF BUS B06.** (Continued)

2.1.1 (Continued)

AFFECTED EQUIPMENT	ASSOCIATED ACTIONS
.2 B002, D2 Battery Charger, AND/OR B004, D4 Battery Charger, AND/OR B022, when aligned to the affected Unit (T.S. 3.8.4, 3.8.5)	☐ If the Required Battery Chargers aligned to **both** D2, and D4 are de-energized, then ENTER Tech. Spec. Action 3.8.4A, 3.8.4B, 3.8.5A, and/or 3.8.5B for loss of two 1E Battery Chargers on the same Train. ☐ Initiate Restoring D2 and then D4 Batteries. [2] ☐ ENSURE B022, Swing Charger, and MCC BS are powered from the unaffected Unit. [SO23-1-3.1, Transfer of MCC BS and Emergency Chiller ME-335 to the Unit 2(3) Power Source.], and ☐ Place B022, Swing Battery Charger, in service. **AND/OR** ☐ Place B017, Spare Charger(s) in service. **AND/OR** ☐ REQUEST Maintenance to supply Temporary Power to Required Charger(s). If required, then perform D2 Battery Load reduction, as follows: [1] (T.S. 3.3.1, 3.3.3)

Comments / Reference: From SD-SO23-140, Figure I-2B	Revision # 15



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 G 2.4.35	
	Importance Rating	3.8	

Loss of Nuclear Service Water: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects

Proposed Question: Common 53

Given the following conditions:

- SO23-13-2, Shutdown from Outside of the Control Room is in progress due to a fire in the Unit 2 Control Room.
- The Radwaste Operator ensures that the Saltwater Cooling / Component Cooling Water (SWC/CCW) Heat Exchanger Outlet Valve is open.

Which ONE (1) of the following describes the reason this action is required?

A. A hot short due to fire could have caused Heat Exchanger Outlet Valve automatic closure and the associated SWC Pump is running at shutoff head.

B. A hot short due to fire could have caused Heat Exchanger Outlet Valve closure and SWC Pump trip. Opening Heat Exchanger Outlet Valve restarts the SWC Pump.

C. The CRS will be unable to start the SWC Pump from the Second Point of Control due to the interlock with the Heat Exchanger Outlet Valve being closed.

D. The CRS will be starting the SWC Pump from the Second Point of Control and the automatic opening feature is defeated with the Fire Isolation Switch in LOCAL.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because hot shorts can result in spurious equipment operation and it could be thought that there were no operator actions to secure the SWC Pump.
B. Incorrect. Plausible because hot shorts can result in spurious equipment operation and an interlock does exist between these components but no interlocks exist with the Fire Isolation Switch in LOCAL.
C. Incorrect. Plausible because an interlock does exist between these components but no interlocks exist with the Fire Isolation Switch in LOCAL.
D. Correct. As part of the local operator actions the Fire Isolation Switch is placed in LOCAL in order to start the SWC Pump. The automatic opening feature is defeated with the FIS in LOCAL.

Technical Reference(s)	SD-SO23-410, Page 9 SO23-13-2, Attachment 2, Section 4.0 SO23-13-2, Attachment 6, Step 17.3 SO23-13-2, Attachment 10, Section 7.0	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56671 — DESCRIBE the purpose and operation of the Fire Isolation Switch for a given component.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
- 55.41 7, 8, 10
- 55.43

Comments / Reference: From SD-SO23-410, Page 9	Revision # 7

NUCLEAR ORGANIZATION SYSTEM DESCRIPTION SD-SO23-410
UNITS 2 AND 3 REVISION 7 PAGE 9 OF 40

2.0 DESCRIPTIONS (Continued)

2.2.2 Saltwater Cooling Pumps (P-112, P-113, P-114 and P-307) (Continued)

Each Saltwater Cooling Pump is equipped with a backlit switch-light module with START, STOP and OVERRIDE pushbuttons and an ammeter. The control switches are located on CR-64 and are numbered HS-6380-1 for P-112, HS-6382-1 for P-307, HS-6381-2 for P-113 and HS-6383-2 for P-114. Each pump can be controlled from panel CR-64 or from the associated 4 kV switchgear. A two-position fire/isolation switch is located in the associated fire isolation panel in the switchgear room. With the switch in the "LOCAL OR REMOTE" position, the pump can be operated from CR-64 or the 4 kV switchgear. With the switch in the "LOCAL" position, the pump can be operated only at the 4 kV switchgear and pump's valve interlocks are bypassed for pumps P-112, P-114 and P-307.

Comments / Reference: From SO23-13-2, Attachment 2, Section 4.0	Revision # 11

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ABNORMAL OPERATING INSTRUCTION REVISION 11 ATTACHMENT 2 — SO23-13-2 PAGE 19 OF 227

UNIT 2 CRS DUTIES (Continued)

PERF. BY INITIALS

4.0 In the Unit 2 Train B 1E Switchgear Room:

4.1 UNLOCK (93) and OPEN Fire Isolation Panel 2L-413. ________

CAUTION

When repositioning the Fire Isolation Switches in the following Step, DO NOT reposition the DG Crosstie Switches marked "Normal" and "50.54X"

4.1.1 SELECT all Fire Isolation Switches to LOCAL. ________

4.2 Open Second Point of Control Cubicle 2A06-01. ________

4.2.1 SELECT all Control Switches to STOP. ________

Comments / Reference: From SO23-13-2, Attachment 6, Step 17.3	Revision # 11

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ABNORMAL OPERATING INSTRUCTION REVISION 11 ATTACHMENT 6 — SO23-13-2 PAGE 49 OF 227

22 DUTIES (Continued)

PERF. BY INITIALS

17.3 Establish Cooling Systems:

NOTE: SWC Pump will start approximately 5 seconds after control switch is operated.

17.3.1 When 2HV-6497, SWC/CCW HX Outlet Valve has been verified Open, then coordinate with the Unit 2 CRS to start a SWC Pump from Second Point of Control: ________

☐ MP-307

☐ MP-112

17.3.2 If 2HV-6497 will not open, then coordinate with the Primary Operator to OPEN MCC Breaker 2BK23 and MANUALLY OPEN 2HV-6496, Overboard Block Valve to Seawall. ________

17.3.3 Coordinate with the Unit 2 CRS to start a CCW Pump from Second Point of Control: ________

☐ MP-024 ☐ MP-025 (If aligned to Train A.)

Comments / Reference: From SO23-13-2, Attachment 10, Section 7.0	Revision # 11

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ABNORMAL OPERATING INSTRUCTION
REVISION 11
ATTACHMENT 10

SO23-13-2
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RADWASTE OPERATOR DUTIES - UNIT 2

CONTINUOUS USE

PERF. BY INITIALS

1.0 At SSD Locker, obtain an emergency lantern, Alarming Dosimeter, TLD, Headset and Unit 2 Primary Operator Keyset (4, GMK, HR). **(SSD KIT: Pri. Op.)** ________

1.1 Obtain a set of security keys. ________

2.0 Performance Guidelines:

2.1 Do not delay these actions for Security, or any other concerns unless a delay is necessary to maintain personnel safety.

2.2 Due to the seriousness of the emergency, prompt completion of these actions overrides all other Procedures, Documents, Work Plans, Technical Specifications, Technical Manuals, and/or Verbal Directions given by any person or group other than the Operations Shift Manager.

2.3 If Card Readers are inoperative, then use Security Key No. 0 to pass through Card Reader Doors.

3.0 Proceed to Radwaste via Control Building Central Stairwell and Health Physics Control Point.

4.0 At 24' Radwaste:

4.1 OPEN 2HV-9235, BAMU Gravity Feed. ________

4.2 OPEN 2HV-9240, BAMU Gravity Feed. ________

5.0 At 37' Radwaste:

5.1 CLOSE 2LV-0227B, VCT Outlet, (Rm. 319A, Key No. HR). ________

6.0 At 9' Radwaste:

6.1 ENSURE CLOSED 2LV-0227C, RWST to Charging Pump suction. ________

7.0 In the SWC Pump Room:

7.1 ENSURE OPEN 2HV-6497, SWC/CCW HX Outlet Valve. ________

7.1.1 If 2HV-6497 will not open, then OPEN MCC Breaker 2BK23 and MANUALLY OPEN 2HV-6496, Overboard Block Valve to Seawall. ________

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	065 AA2.03	
Importance Rating	2.6	

Loss of Instrument Air: Ability to determine and interpret the following as they apply to the Loss of the Instrument Air: Location and isolation of leaks

Proposed Question: Common 54

Given the following conditions:

- Unit 2 and 3 are experiencing a major air leak.
- The following Annunciators are in alarm:
  - 61C19 - INST AIR HEADER PRESS LO.
  - 61B38 - N2 SUPPLY TO INST AIR HEADER ON.
  - 61C18 - SERVICE AIR HEADER PRESS LO.
  - 61B39 - INST AIR DRYER TEMP/LEVEL/DP HI.
  - 61B40 - INST AIR DRYER TROUBLE.
- The following indications are available:
  - 2PI-5344A, Instrument Air Header Pressure on CR-61 indicates 65 psig and is slowly lowering.
  - Two (2) Instrument Air Compressors are running loaded.
  - One (1) Instrument Air Compressor is in standby.
  - 2PI-5344B, Nitrogen Supply Header Pressure on CR-61 is steady.
  - 2HV-5343 and 3HV-5343, Instrument Air Supply to Containment Excess Flow Check Valves are OPEN.
  - Annunciator 57C58 - INSTRUMENT AIR TO CONTAINMENT is NOT in alarm on either unit.
  - Flow indication to Unit 2 and Unit 3 are approximately the same.

Which ONE (1) of the following describes the condition that exists?

There is a major leak...

A. at the outlet of the Instrument Air Dryers.

B. at the inlet of the Instrument Air Dryers.

C. on one of the Instrument Air Receivers.

D. on the Instrument Air header going into the Fuel Handling Building.

Proposed Answer: A

Explanation:

A. Correct. The leak location would cause all SA and IA compressors to run and cause the Hi D/P alarm on the dryer. The downstream pressure being constant indicates the N2 system is supplying.

B. Incorrect. Plausible because the leak before the IA Dryer would cause the dryer trouble alarm and other low pressure indications, however, it would not cause the IA Dryer Temp/Level/ΔP alarms.

C. Incorrect. Plausible because it could be thought there was not a check valve at each receiver outlet and all compressors would feed the leak.

D. Incorrect. Plausible because it could be thought that the leak on the Fuel Bldg supply would also create the high ΔP condition on the Dryer, however, the N2 Supply pressure would not be stable while IA header pressure was dropping; should indicate the same pressure for this leak.

Technical Reference(s): SD-SO23-570, Figure I-1; SD-SO23-570, Pages 6 to 9; SO23-13-5, Entry Conditions — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 72867 — ANALYZE normal and abnormal operations of the Instrument and Respiratory & Service Air Systems.

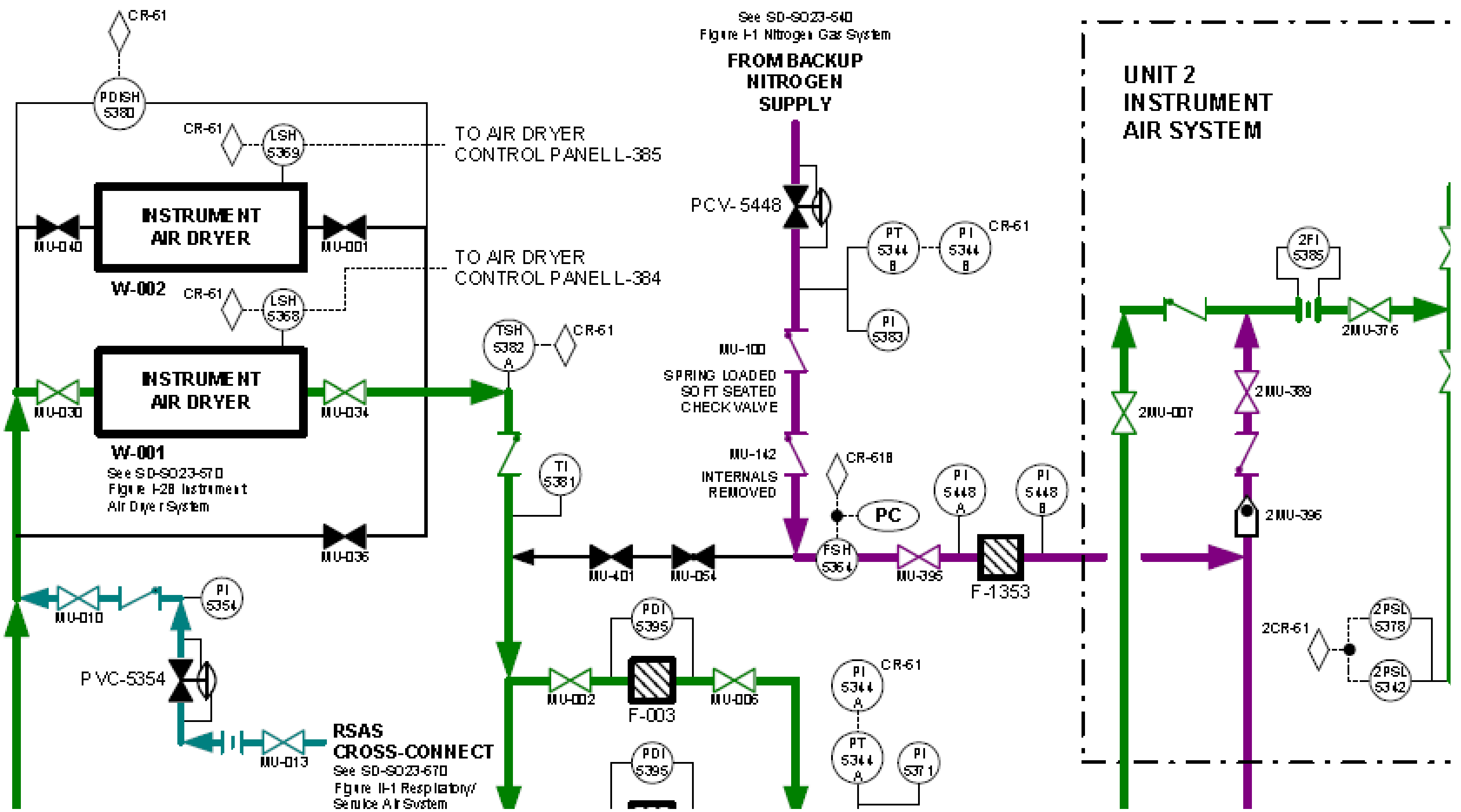
Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

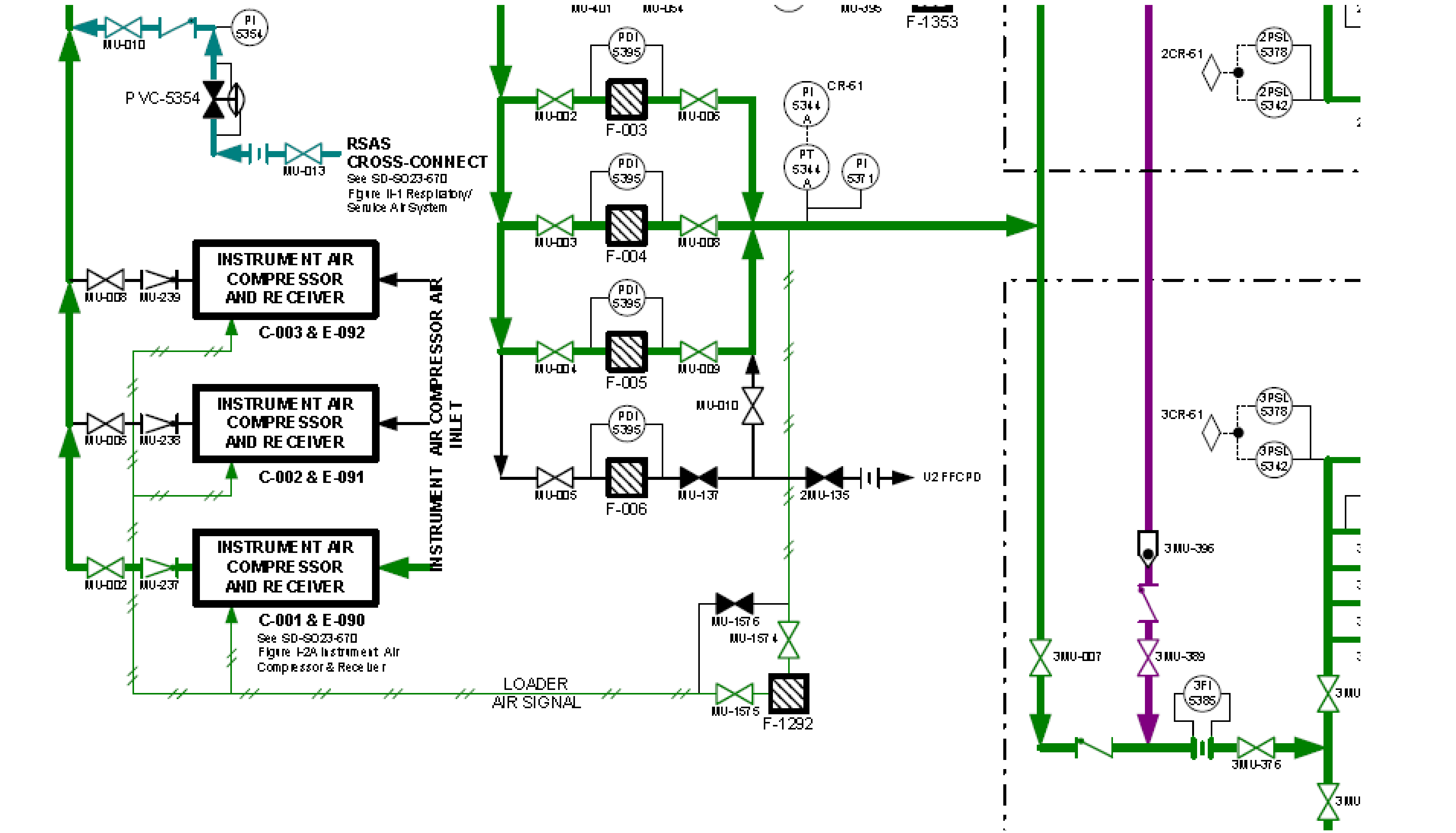
Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7
55.43 ____

Comments / Reference: From SD-SO23-570, Figure I-1	Revision # 16



Comments / Reference: From SD-SO23-570, Figure I-1	Revision # 16



Comments / Reference: From SD-SO23-570, Page 6	Revision # 16

PART I INSTRUMENT AIR SYSTEM

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

.2.5 In the event Instrument Air Compressors are not available Respiratory Service Air system can supply the Instrument Air header. Instrument Air is established through 2/3PCV-5354 Pressure Regulating Valve, and spring loaded soft seated check valve, SA2423MU1573. Located between the valves are a constantly open vent and 2/3PI-5354 pressure indicator. They are available to verify Instrument Air is not flowing into Respiratory Service Air system.

.2.6 Air from the common air header is directed to 1 of 2 Instrument Air Dryers. The Instrument Air Dryer separates and removes moisture and oil from the air. Downstream of the Air Dryers is a check valve followed by a branch line connection the Nitrogen Backup Supply System.

.2.7 Dry Instrument Air is then directed to 3 Air Filters, the 4th is in standby. The Air Filters remove particulate from the air with 5 micron filters. The Instrument Air is then sent to a line that directs it to each Units Instrument Air Header and the Unit 2 Full Flow Condensate Polishing Demineralizers.

.2.8 Individual instrument air lines tap off of each Units Instrument Air Header. Instrument air is directed to all instrument air connections throughout the plant including those within the Containments.

.2.9 There are isolating check valves in each Unit's Header downstream of the Instrument Air Filters. Back-up Nitrogen connects to each Unit's Header branch lines downstream of the isolating check valves. The Back-up Nitrogen is directed through a pressure regulator 2/3PCV-5448 set at 83 psig, then a 5 micron (maximum) Filter. Nitrogen supply lines then branches to supply each Unit's Instrument Air Header separately downstream of the isolating check valves. Each branch is provided with an 2(3)HV-5343 Reactor Instrument Air Supply Flow Excess Flow Check Valve, a check valve and an a control valve 2(3)HV-5388 Instrument Air to Containment Isolation Valve.

Comments / Reference: From SD-SO23-570, Page 7	Revision # 16

PART I INSTRUMENT AIR SYSTEM

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

.2.10 2(3)HV-5343 Reactor Instrument Air Supply Flow Excess Flow Check Valve will close given a sufficiently large line break, of ≥200 CFM, isolating the fault in instrument air piping. The isolating check valves in the other unit will ensure that the non-faulted piping can not feed the break. The backup Nitrogen would then feed only the non-faulted unit, keeping its Instrument Air System pressurized. A 0.030" orifice in the Excess Flow Check Valves' poppet allows pressure to equalize across the valve, automatically resetting it after system restoration.

.2.11 Flow indicators, 2(3)FI-5385 for the Containment Instrument Air Supply Excess Flow Check Valves are provided downstream of the Backup Nitrogen tie-in points to provide a means of monitoring system air consumption. The flow indicators are also used to troubleshoot system problems.

.2.12 Individual Instrument Air lines tap off of each Units Instrument Air Header. Instrument air is directed to all Instrument Air connections throughout the plant including those within Containment.

.2.13 Instrument Air to Containment is first directed through an Excess Flow Check Valve. The excess flow check valve will isolate instrument air to Containment upon abnormally high flow conditions.

.2.14 From the Excess Flow Check Valve, Instrument Air passes through 2(3)HV-5386 Instrument Air to Containment Isolation Valve and enters Containment through penetration #22.

.2.15 Inside Containment, instrument air is directed through a check valve and to the Containment Instrument Air loads.

Comments / Reference: From SD-SO23-570, Page 8	Revision # 16

PART I INSTRUMENT AIR SYSTEM

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

.3 The Instrument Air System is designed to maintain system air pressure under varying anticipated flow demands. Instrument Air System air pressure is controlled by: (Figure I-3)

.3.1 Air compressor(s) loading and unloading operations.

.3.2 Nitrogen Back-up Supply System. Nitrogen is supplied down stream of the Instrument Air Filters.

.3.3 Respiratory / Service Air Backup Supply. The air pressure is supplied through the cross connection upstream of the Instrument Air Dryers.

Comments / Reference: From SD-SO23-570, Page 9	Revision # 16

PART I INSTRUMENT AIR SYSTEM

2.0 DESCRIPTION (Continued)

2.1 System Overview (Continued)

.5 Upon decreasing system pressure, as a "full-load" pressure sensor set point is reached, each Air Compressor aligned to that pressure sensor fully-loads.

.5.1 As Instrument Air system pressure decreases, each of the 3 Instrument Air Compressors start and receive half-load and full-load control signals from the pressure sensors they are aligned to.

.5.2 That the flow resistance through one air dryer is too great to allow 3 Compressors to load at once.

.5.3 2/3PVC-5354 Service Air Backup TO Instrument Air System Pressure Control Valve is normally set at 88 psig.

.5.4 This set point will allow the first two instrument Air Compressors (LEAD and LAG1) to fully load, and the third (LAG2) Compressor to half load before starting to supply Service Air to the Instrument Air system. NOTE: that the Service Air pressure regulator can be set higher or lower at Operations/STEC direction to either carry the plant Instrument air load or to emulate the LAG1 or LAG2 Air Compressors.

.5.5 If pressure decreases to 83 psig, the Nitrogen Backup Supply System Isolation Valve 2/3PCV-5448 NITROGEN BACK-UP PRESS CONTROL VALVE to Instrument Air opens to allow nitrogen to assist in maintaining Instrument Air System pressure.

.5.6 As system pressure recovers to above 83 psig, the Nitrogen Backup flow stops and the Instrument Air System pressure is again maintained by the Instrument Air Compressors.

Comments / Reference: From SO23-13-5, Entry Conditions	Revision # 7

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ABNORMAL OPERATING INSTRUCTION REVISION 7 — SO23-13-5 PAGE 2 OF 34

LOSS OF INSTRUMENT AIR

PURPOSE

Specify actions to mitigate effects of excessive Instrument Air System leakage or loss of Instrument Air Compressors.

ENTRY CONDITIONS

This event may be caused by any of the following abnormal conditions:

1. Excessive Instrument Air System flow.
2. Unplanned Loss of all Instrument Air Compressors.

This event could be identified by one or more of the following alarms or indications:

1. 261C19, INST AIR HEADER PRESS LO
2. 361C19, INST AIR HEADER PRESS LO
3. 61B58, INST AIR COMPRESSOR CONTROL PANEL TROUBLE
4. 61B38, N2 SUPPLY TO INST AIR HEADER ON

NOTE: Excessive flow through the Instrument Air Dryer will increase Dryer delta pressure above the alarm setpoint.

5. 61B39, INST AIR DRYER TEMP/LEVEL/DP HI
6. 2/3PI5344A, Instrument Air Header Pressure (CR61)
7. 2/3PI5344B, Nitrogen Supply Header Pressure (CR61)
8. HV-5343, Instrument Air Supply to Containment Excess Flow Check Valve, closed. (CR-57)
9. Unanticipated start of backup Instrument Air Compressor(s) or RSAS Compressor(s).

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	007 AA1.05	
	Importance Rating	3.9	

Generator Voltage and Electric Grid Disturbances: Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Engineered safety features

Proposed Question: Common 55

Given the following conditions:

- Unit 2 is in MODE 1 at 100% power.
- Entry into SO23-13-26, Loss of Power to an AC Bus is required due to a loss of 4160 V Bus 2A08.

Which ONE (1) of the following identifies the action required per SO23-13-26, Loss of Power to an AC Bus?

A. ENERGIZE 1E Pressurizer Heaters as required due to loss of ALL Non-1E and Proportional Heaters.

B. ENSURE the standby Condensate Pump has started due to low Main Feedwater Pump suction pressure.

C. INITIATE a Containment Cooling Actuation Signal due to loss of the Containment Chillers.

D. REDUCE power to 65% due to loss of two (2) Circulating Water Pumps.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the 1E Pressurizer Heaters must be operated, however, not all Pressurizer Backup Heaters are lost.
B. Incorrect. Plausible because the standby Condensate Pump would start on low discharge pressure or trip of a running Condensate Pump, however, this loss is associated with Bus A03.
C. Correct. A Containment Cooling Actuation Signal (CCAS) must be initiated in the Control Room because Normal Containment Cooling is lost.
D. Incorrect. Plausible because this is the required action per SO23-5-1.7, Power Operations, however, the Circulating Water Pumps are lost on a loss of 4160 V Bus 2A03 or 2A07.

Technical Reference(s) SO23-13-26, Attachments 8 & 12; SO23-5-1.7, Attachment 5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54513	DESCRIBE plant and initial operator response to a loss of a Non-1E 4.16 KV bus while at power in accordance with SO23-13-26.		
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: From SO23-13-26, Attachment 12	Revision # 8

2.0 PROCEDURE (Continued)

2.1 **EQUIPMENT ACTIONS FOR LOSS OF BUS A08.** (Continued)

2.1.1 (Continued)

AFFECTED EQUIPMENT	ASSOCIATED ACTIONS
.3 (Continued) MCC B10, MP-158 and MP-159, Aux Bldg Chilled Water Pumps and 2Q0612, Instrument Bus #2. **(Unit 2 only)** and RT-7808, Plant Vent Stack Radmonitor and Waste Gas Analyzers	☐ 6) PLACE ME-331, Aux Building Chiller in service per SO23-1-3, Section for Starting a Chiller Unit. 7) If Aux Building Chiller restart is delayed, and ESF Room temperatures become unacceptably high, then: ☐ a. START Emergency Cooling in ESF rooms per SO23-3-2.27, Attachment for Emergency Cooler Exercise Run, as needed to maintain the following room temperatures: • SEB ESF Rooms < 104°F • Charging/BAMU Pp. Rooms < 97°F • ESF Switchgear/Battery Rooms < 90°F • Vital Power Dist. Rooms < 99°F 8) **(Unit 2 only)** If the transfer of B10 to Unit 3 was unsuccessful, then TRANSFER 2Q0612 to the Emergency power source by selecting 2VS612, 2Q0612 Instrument Bus #2 Transfer Switch to Emergency. 9) If 2RT7865 and 3RT7865 are **not** Operable, then perform required actions of SO23-3-2.24 Attachments for Waste Gas Holdup System 2/3RE7808 & 3RE7865 and Plant Vent Stack 2/3RE7808 & 2RE7865/3RE7865.
.4 ME-202, Containment Chiller MP-448, ME-201 Cont. Chiller Oil Pump MP-128, Containment Chilled Water Pump	PLACE the Containment Emergency Cooling System in service by manually initiating a Containment Cooling Actuation Signal using: ☐ HS-9138-1 and HS-9138-2 at CR56 **OR** ☐ HS-9138-3 and HS-9138-4 at CR53
.5 ME-403, CEDM Cooling Unit.	☐ ENSURE ME-404, CEDM Standby Cooling Unit is in service.

Comments / Reference: From SO23-13-26, Attachment 12	Revision # 8

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2.0 PROCEDURE (Continued)

2.1 **EQUIPMENT ACTIONS FOR LOSS OF BUS A08.** (Continued)

AFFECTED EQUIPMENT	ASSOCIATED ACTIONS
.6 MA-310, Continuous Exhaust Fan **(Unit 2 only)**	☐ **(Unit 2 only)** ENSURE MA-311 and MA-312 are in service per SO23-1-5, Section for Placing Continuous Plant Exhaust System in Service.
.7 MA-319 and MA-321, RX Cavity Cooling Fans	☐ START MA-320 and MA-322, RX Cavity Cooling Fans.
.8 E122, Proportional PZR Heaters E126 and E124 Backup PZR Heaters	☐ OPERATE the remaining PZR Backup heaters and 1E Heaters as necessary to Compensate for Proportional heater loss.
.9 MP-200 and MP-202, PMW Pumps	☐ Ensure MP-201, PMW Pump is aligned for operation on Unit 2 and that MP-203, PMW Pump is aligned for operation on Unit 3 per SO23-8-10.
.10 3L-395, Unit 3 Aircraft Beacon Panel **(Unit 3 only)**	☐ If both Unit 2 and 3 Aircraft Beacon Lights are out, then NOTIFY the US Federal Aviation Administration, San Diego Flight Services Station at 1-619-557-5280 or 5281.
.11 MA-317, FHB Normal Exhaust Fan MA-359, FHB Normal Supply Fan	☐ ENSURE Running MA-316, FHB Standby Exhaust Fan. ☐ ENSURE Running MA-360, FHB Standby Supply Fan.
.12 MA-192, Radwaste Bldg Normal Exhaust Fan	☐ ENSURE Running MA-193, Radwaste Bldg Normal Exhaust Fan.

Comments / Reference: From SO23-13-26, Attachment 8	Revision # 8

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LOSS OF 4kV BUS A03

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2 **Attempt Bus A03 Energization from Reserve Aux Transformer:**	
☐ a. CHECK Reserve Aux Transformer XR1 available for load.	☐ a. INITIATE Attachment 8, Equipment Actions for Loss of Bus A03.
	☐ 1) If Bus A03 return to service will be delayed, then evaluate supplying loads with temp. power per SO23-6-32.3.
	☐ 2) WHEN Bus A03 is re-energized as directed by the in use EOI, OR when **All** equipment actions of Attachment 8 have been completed, THEN GO TO Step 2f.
b. ENSURE OPEN the following Bus A03 Load Supply Breakers:	
☐ 1) MP-120 TPCW Pump	
☐ 2) MP-058 Heater Drain Pump	
☐ 3) MP-050 Condensate Pump	
☐ 4) MP-051 Condensate Pump	
☐ 5) MP-115 Circ. Water Pump	
☐ 6) MP-117 Circ. Water Pump	

Comments / Reference: From SO23-13-26, Attachment 8	Revision # 8

NUCLEAR ORGANIZATION
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2.0 PROCEDURE (Continued)

2.1 **EQUIPMENT ACTIONS FOR LOSS OF BUS A03.** (Continued)

2.1.1 (Continued)

AFFECTED EQUIPMENT	ASSOCIATED ACTIONS
.10 XM1, Main Transformer Normal Fans/Coolers	☐ TRANSFER Fan/Cooler Assembly Power from the normal to the alternate source per SO23-6-5, Section for Fan/Cooler Assembly Operations.
.11 MP-077, H2 Seal Oil Vacuum Pump	☐ TRANSFER the H2 Seal System Operation from Vacuum Treatment to Bypass Mode per SO23-6-21, Attachment for On-Line Transfer from Vacuum Treatment to Bypass Mode.
.12 Non-running Circ. Water Pumps	☐ Remove non-running Circulating Water Pumps and associated Condenser Sections from service per SO23-2-5.

Comments / Reference: From SO23-5-1.7, Attachment 5	Revision # 41

NUCLEAR ORGANIZATION UNITS 2 AND 3 | OPERATING INSTRUCTION REVISION 41 ATTACHMENT 5 | SO23-5-1.7 PAGE 46 OF 86

RECOMMENDED POWER PLATEAUS

INFORMATION USE

OBJECTIVE

To provide a list of examples to be used as an aid in determining the desired reduced Power Level Plateau. These are intended for extended periods of time- Not of short duration. (Ref. 2.4.2.1, 2.4.2.2, 2.4.2.3, 2.4.2.4)

1.0 This list is comprised of known long term acceptable power levels. Plant Power may be maintained at a higher level with Ops. Management's approval. Plant and Grid conditions may warrant further adjustment of these values.

2.0 Equipment Out of Service — Reactor Power

2.1 Circulating Water System

		Reactor Power
2.1.1	Operation With Reversed Tunnels	90-100% [1]
2.1.2	One Circulating Water Pump (Expected OOS duration is ≤ 12 hrs and OPS Management approval is obtained.)	100%
2.1.3	One Circulating Water Pump (Expected OOS duration is > 12 hrs and ≤ 3 days.)	85%
2.1.4	One Circulating Water Pump (Expected OOS duration is > 3 days.)	75% [2]
2.1.5	Heat Treatment of the Circulating Water System	80-100%
2.1.6	Circulating Water System Tunnel Reversal	80-100%
2.1.7	Two Circulating Water Pumps	65% [8]

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	E05 EK2.2	
Importance Rating	3.7	

Steam Line Rupture - Excessive Heat Transfer: Knowledge of the interrelations between the Excess Steam Demand and the following: Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.

Proposed Question: Common 56

Given the following conditions while at 100% power:

- An Excess Steam Demand Event has occurred inside Containment.
- SO23-12-5, Excess Steam Demand Event directs stabilizing Reactor Coolant temperatures using the Atmospheric Dump Valve on the intact Steam Generator once dry out has occurred on the affected Steam Generator.

Which ONE (1) of the following is the reason for stabilizing Reactor Coolant System temperature?

A. Ensure minimum Reactor Coolant System Core Exit Saturation Margin is maintained.

B. Ensure consistent Reactor Coolant System loop temperatures to prevent loss of Auxiliary Feedwater flow due to high Steam Generator differential pressure.

C. Prevent the post-accident Reactor Coolant System cooldown rate from exceeding the limits in Technical Specifications.

D. Limit Reactor Coolant System heatup which may result in rapid re-pressurization and the onset of Pressurized Thermal Shock conditions.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this is a desired condition during an ESDE, however, it is the heating of the Reactor Coolant System that is the major concern.
B. Incorrect. Plausible because an asymmetric cooldown is taking place, however, this situation is unavoidable and Auxiliary Feedwater flow can be overridden.
C. Incorrect. Plausible because the cooldown rate can be excessive dependent on ESDE severity, however, under most circumstances the Technical Specification cooldown limit is exceeded.
D. Correct. Limiting Reactor Coolant System heat up is paramount in avoiding Pressurized Thermal Shock conditions.

Technical Reference(s)	SO23-12-5, Step 7 Caution SO23-14-5, Step 7 Caution Bases	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54790 — Per the ESDE procedure SO23-12-5 DESCRIBE: The basis for each step, caution or note.

Question Source:
- Bank # 75360
- Modified Bank # ____ (Note changes or attach parent)
- New ____

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 10
- 55.43 ____

Comments / Reference: From SO23-12-5, Step 7 Caution	Revision # 21

NUCLEAR ORGANIZATION UNITS 2 AND 3 — EMERGENCY OPERATING INSTRUCTION REVISION 21 — SO23-12-5 PAGE 7 OF 25

EXCESS STEAM DEMAND EVENT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

7 PREVENT Pressurized Thermal Shock:

NOTE

WHEN excess steam demand remains NOT isolated and all RCPs are stopped, THEN T_C in loop with *least affected* S/G may be higher than REP CET temperature.

CAUTION

Failure to establish steaming flow path on least affected S/G before most affected S/G loses effective heat removal capabilities will result in rapid re-pressurization (PTS consideration).

a. INITIATE FS-30, ESTABLISH Stable RCS Temperature during ESDE.

Comments / Reference: From SO23-14-5, Step 7 Caution Bases	Revision # 8

EXCESS STEAM DEMAND EVENT BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.4.7 STEP 7 PREVENT Pressurized Thermal Shock

Intent
The main objective following an overcooling event is to minimize the stresses on the reactor vessel, return RCS temperature to within the Appendix G limits of Tech Specs, and establish stable RCS pressure and temperature until a cooldown to SDC entry conditions can be started. In general, a controlled cooldown should be started as soon as possible. The intent of this step is to direct the operator to take control of, and stabilize, RCS temperature following S/G dry out during an ESDE.

.1 NOTE prior to Step 7a

The NOTE discusses the situation in which the *least affected* S/G is not steamed during natural circulation conditions. For this situation, the *most affected* S/G provides the heat sink to cause natural circulation of coolant through the core. The *least affected* loop will have little or no flow through it. Reverse heat transfer will take place if some RCS flow remains, resulting in its RCS T_{COLD} possibly being higher than the temperature of the coolant passing through the core (CET, or REP CET). In such a case, if steaming of the *least affected* S/G is started after the *most affected* S/G is dry, a temporary loss of natural circulation driving head would result. Natural circulation will be re-established either when Core Exit temperature rise above the *least affected* loop RCS T_{COLD} as a result of plant heatup, or when the T_C in *least affected* loop is brought below Core Exit temperatures as a result of steaming the least affected loop.

.2 CAUTION prior to Step 7a

The CAUTION statement describes the expected plant response (rapid RCS repressurization due to heatup and expansion of the RCS coolant) if a steaming flow path is not established on the *least affected* S/G before the *most affected* S/G loses effective heat removal capabilities. Initiation of steaming of the *least affected* S/G prior to the *most affected* S/G losing effective heat removal capability will ensures that the heat sink for the RCS is maintained.

Method
At this point in the event there is reasonable probability the ESDE has been isolated. RCS temperature and pressure could now be expected to rise quickly. Action must be initiated at this point to prevent Pressurized Thermal Shock (PTS). The *least affected* S/G needs to be operated to stabilize RCS temperature and limit RCS repressurization.

When the isolated S/G dries out following an ESDE, RCS temperature will begin to rise unless a means of controlled steaming is established. If a method of heat removal is not established, the RCS heatup, in conjunction with the safety injection inventory added, could cause the plant to go solid and create a potential PTS concern.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	060 AA1.02	
	Importance Rating	2.9	

Accidental Gaseous Radwaste Release: Ability to operate and/or monitor the following as they apply to the Accidental Gaseous Radwaste: Ventilation system

Proposed Question: Common 57

Given the following conditions:

- A planned release of T086, Waste Gas Decay Tank is in progress.
- A high spike occurs on 2RE-7865, Wide Range Gas Monitor.
- 2/3HV-7202, Waste Gas Discharge Isolation Valve failed to close.

Which ONE (1) of the following would result in the closing of 2/3HV-7202, Waste Gas Discharge Isolation Valve?

A. LOSS of all Continuous Exhaust Fans (A310, A311 and A312).

B. INITIATE a Containment Purge Isolation Signal.

C. RAISE 2/3RE-7808, Plant Vent Stack Wide Range Monitor <u>setpoint</u> at the Data Acquisition System.

D. LOSS of both Radwaste Area Exhaust Fans (A192 and A193).

Proposed Answer: A

Explanation:

A. Correct. Loss of the Continuous Exhaust Fans will close 2/3 HV-7202, Waste Gas Discharge Isolation Valve.

B. Incorrect. Plausible because 2RE-7865, Wide Range Gas Monitor can be aligned to initiate a CPIS, however, this action will not secure 2/3 HV-7202.

C. Incorrect. Plausible because 2/3RE-7808 is interlocked to close 2/3 HV-7202, Waste Gas Discharge Isolation Valve, however, raising the setpoint will not close the valve.

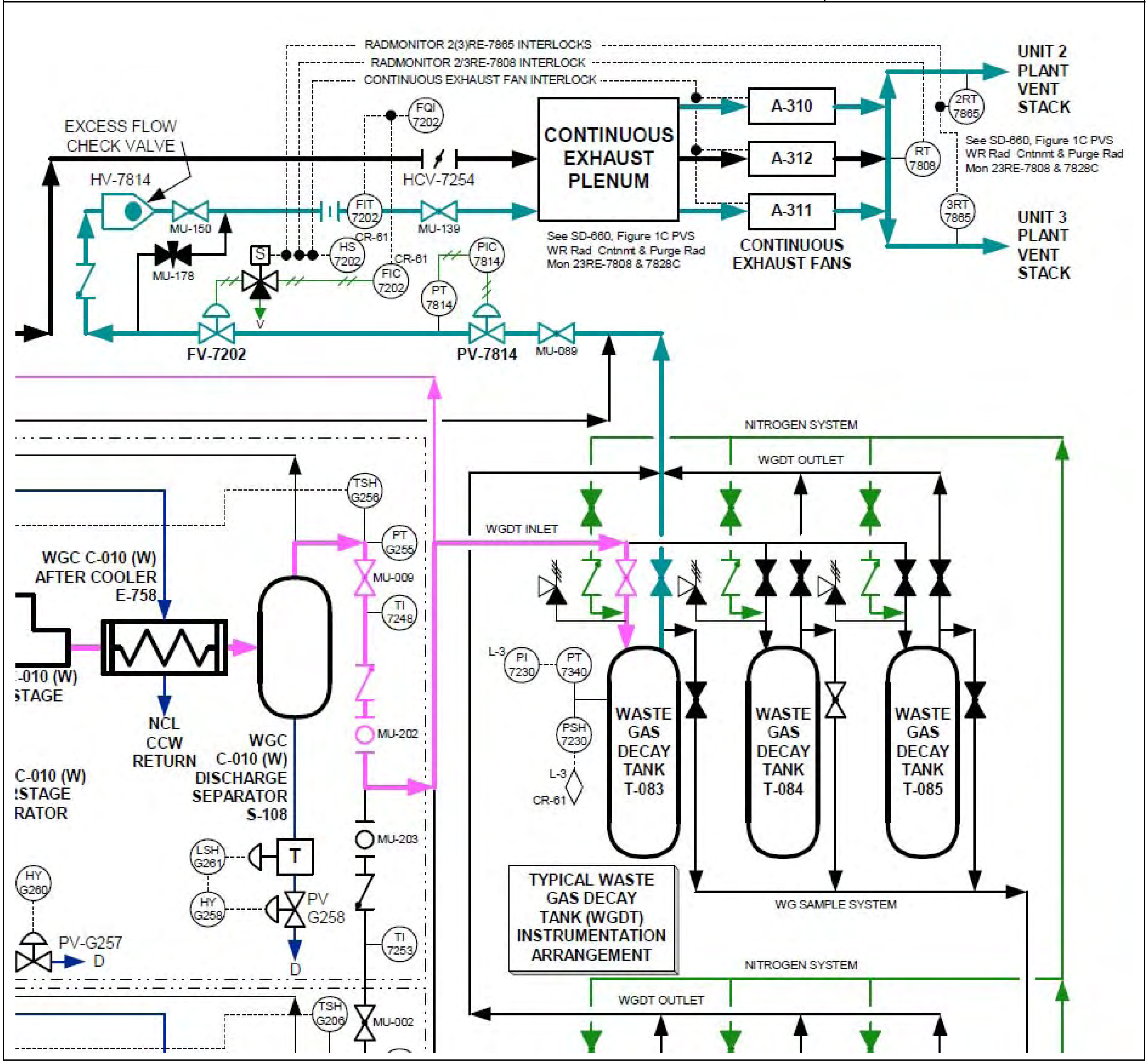
D. Incorrect. Plausible because the Radwaste Area Exhaust Fans discharge to the Continuous Exhaust Plenum and it could be thought that low flow would trip the Continuous Exhaust Fans.

Technical Reference(s)		
	SD-SO23-660, Figure 1	Attached w/ Revision # See Comments / Reference
	SD-SO23-660, Page 17	
	SD-SO23-622, Figure I-1	
	SD-SO23-690, Page 62	

Proposed references to be provided during examination: None

Learning Objective: 56619 / 56617	ANALYZE normal and abnormal operations of the Gaseous Radwaste System. DESCRIBE the configuration and operational characteristics of Gaseous Radwaste System components.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 7
	55.43 ______

Comments / Reference: From SD-SO23-660, Figure 1	Revision # 7



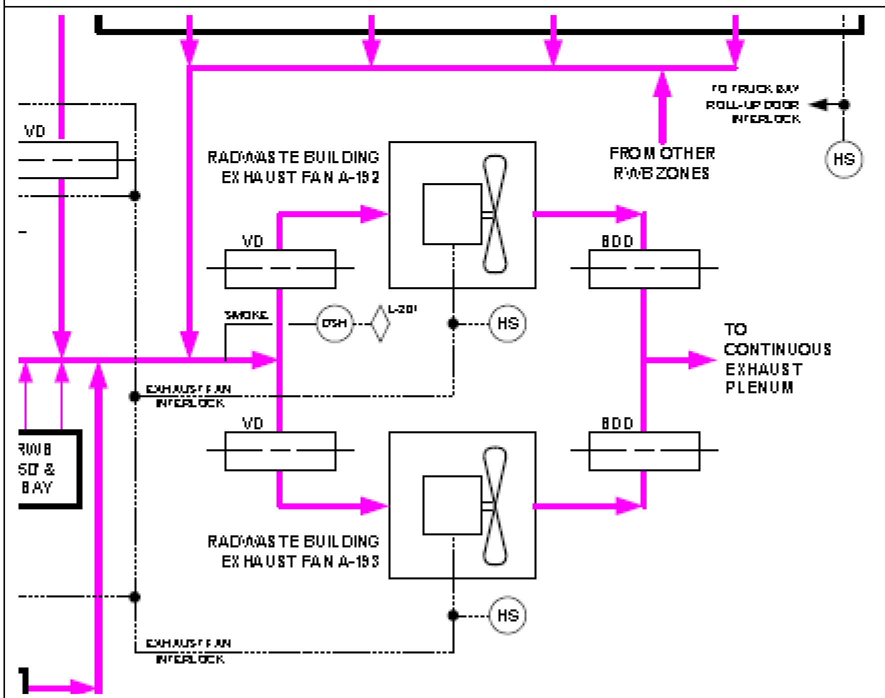
Comments / Reference: From SD-SO23-660, Page 17	Revision # 7

2.3.2 **Hydrogen and Oxygen Control**

The continuous hydrogen and oxygen analyzers, located on 63' Radwaste, initiate alarms should discharge concentrations (in the Surge Tank or in the Decay Tank) of oxygen reach > 4%, and a mixed gas concentration of oxygen and hydrogen reach > 1% and > 3%, respectively. The Surge Tank alarm is designed to initiate automatic nitrogen dilution of the surge tank but is procedurally isolated and requires is manual nitrogen injection. The Decay Tank discharge alarm initiates action by the plant operator. The release rate of waste gas from the Decay Tank is controlled by a flow controller set for a maximum discharge rate of 50 SCFM. The isokinetic air flow rate through the Discharge Plenum should be 66,000 to 113,000 SCFM with a minimum of 2-out-of-3 plant Vent Stack Fans operating.

The Continuous Exhaust Plenum is provided with redundant fans. If waste gas discharge is in progress, the gas release is automatically terminated upon receipt of a "loss of all fans" alarm. If discharge is not in progress, interlocks prevent the automatic isolation valve, FV-7202, in the Decay Tank discharge header from opening when an Exhaust Fan is not in operation.

Comments / Reference: From SD-SO23-622, Figure I-1	Revision # 2



Comments / Reference: From SD-SO23-690, Page 62	Revision # 16

2.3.4 **Gaseous Effluent Radiation Monitoring System**

.1 Plant Vent Stack Wide Range Radiation Monitor, 2/3RE-7808 (See Figure 3)

.1.1 The Plant Vent Stack Wide Range Radiation monitor is used to:

.1.1.1 Monitor Plant Vent Stack for radiation levels.

.1.1.2 Monitor Waste Decay Tank release and CLOSE the Waste Gas Isolation Valve, 2/3FV-7202, on high radiation or instrument failure.

.1.2 2/3RE-7808 is common to both Units, located on the plant vent stack at the 63 foot elevation of the Radwaste Building corridor. The monitor draws a sample from and returns the sample to the ventilation duct leading to the Plant Vent Stack, downstream of the Continuous Exhaust Plenum. The monitor can be aligned to sample either one or both discharge flow paths.

.1.3 The channels monitor for airborne gaseous activity in the exhausted air. 2/3RE-7808 functions to alarm and to STOP a Waste Decay Tank release by CLOSING the Waste Gas Isolation Valve, 2/3FV-7202, upon high radiation levels.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051 AK3.01	
	Importance Rating	2.8	

Loss of Condenser Vacuum: Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum

Proposed Question: Common 58

Which ONE (1) of the following states the reason the Steam Bypass Control System Valves close on Condenser low vacuum?

A. Prevent overfilling the Condenser Hotwells.

B. Protect the Main Turbine.

C. Prevent damage to the Condenser due to overheating.

D. Avoid damage due to differential pressure between Water Boxes.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this situation can occur if water box differential pressure is high, however, the reason is to avoid Condenser overheating.
B. Incorrect. Plausible if thought that Main Turbine protection were the concern, however, the reason is to avoid Condenser overheating.
C. Correct. This is the reason for blocking SBCS on low Condenser vacuum.
D. Incorrect. Plausible because differential pressure between Water Boxes does impact plant operation, however, this is not the correct reason.

Technical Reference(s): SO23-15-52A, 52A19; SD-SO23-175, Pages 27 & 32 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 59226 — INTERPRET instrumentation and controls utilized in the Main Steam System.

Question Source:
- Bank # 127199
- Modified Bank # ________ (Note changes or attach parent)
- New ________

Question History: Last NRC Exam SONGS 2005B

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

10 CFR Part 55 Content: 55.41 5, 7
55.43 ____

Comments / Reference: From SO23-15-52A, 52A19 | Revision # 12

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ALARM RESPONSE INSTRUCTION REVISION 12 ATTACHMENT 2 | SO23-15-52.A PAGE 45 OF 120

52A19 SBCS DISABLED

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-3	AMBER	N/A	99B07

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)HS-8429	SBCS Emergency/OFF/ Vac Trip Reset Pushbutton	Depressed	NONE	NONE	795/806
2(3)L-120 Relay	Main Condenser Vacuum Low	10.0" Hg Backpressure	2(3)PR-3205 2(3)PI-3202A 2(3)PI-3383A 2(3)PI-3395A		

1.0 REQUIRED ACTIONS:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Low Main Condenser Vacuum	2.1 Refer to SO23-13-10, Loss of Condenser Vacuum.

3.0 ASSOCIATED RESPONSES:

3.1 Refer to SO23-3-2.18.1, and use ADVs as necessary to maintain proper steam generator pressure.

Comments / Reference: From SD-SO23-175, Page 27	Revision # 6
To prevent SBCS actuation during times of insufficient condenser vacuum, three separate and independent pressure switches from the LP condenser sections must be satisfied in a two out of three configuration before any SBCV can open. An actuation from two out of any three swithes will cause a quick closing of any open valve(s) via the Condenser Interlock 1 and 2 Circuits. These circuits are the same ones actuated by the Emergency Off pushbutton. The actuation setpoint from condenser back pressure is ≥10" HgA. This interlock will automatically reset when sufficient condenser vacuum returns, provided all the individual valve controllers, and the Master Controller is in "AUTO". If any one of these controllers is in manual, the Condenser Interlock 1 and 2 signals must be manually reset via the Emergency Off/Condenser Interlock Reset switch, after sufficient condenser vacuum returns.	R

Comments / Reference: From SD-SO23-175, Page 32	Revision # 6

3.0 OPERATIONS (Continued)

3.2 Abnormal Operations

Some of the SBCS Abnormal Operations include inoperable SBCVs, one or more circulating water pumps secured, return to services following a power loss and return to service of a SBCV with steam leads hot.

3.2.1 Inoperable Steam Bypass Control Valves

The Turbine Trip-Reactor Trip setpoint is reduced proportionally for the number of inoperable SBCVs, if Reactor power is below 55%. Above 55%, the reactor automatically trips on a turbine trip and the setpoint need not be re-adjusted. The Plant Superintendent's permission is required to re-adjust the Reactor Loss of Load setpoint.

3.2.2 Operations With Secured Circulating Water Pumps

For each Circulating Water Pump that is secured, 1 of the 2 SBCVs serving that section of the condenser must be taken out of service. Failure to do this could result in turbine and tube sheet damage. Placing the appropriate SBCV permissive switch in "OFF" takes that valve out of service. If below 55% power, reset the Reactor Loss of Load permissive accordingly.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	032 G 2.1.27	
Importance Rating	3.9	

Loss of Source Range Nuclear Instrumentation: Conduct of Operations: Knowledge of system purpose and/or function

Proposed Question: Common 59

Given the following conditions:

- Unit 2 is in MODE 6 with core re-load in progress.
- JI-0005, Startup Channel B has failed low.
- JI-0006, Startup Channel A remains OPERABLE.

Which ONE (1) of the following describes operating functions that are affected?

A. Potential loss of audible count rate and loss of input to Core Vibration Monitor.

B. Loss of one channel of Plant Protection System High Log Power Bypass and input to the Main Control Board SUR indication.

C. Loss of one channel of Boron Dilution Monitoring and loss of input to Core Vibration Monitor.

D. Loss of one channel of Boron Dilution Monitoring and loss of Remote Shutdown Panel indication.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because loss of audible count rate is correct, however, the non-safety channels do not input into the Main Control Board SUR indication.
B. Incorrect. Plausible if thought that these inputs were provided by the Startup Channels, however, these indications are provided by the Safety Channels.
C. Incorrect. Plausible because loss of Boron Dilution Monitoring is correct, however, the non-safety channels do not input into the Core Vibration and Loose Parts Monitors
D. Correct. These functions are affected by failure of Startup Channel B.

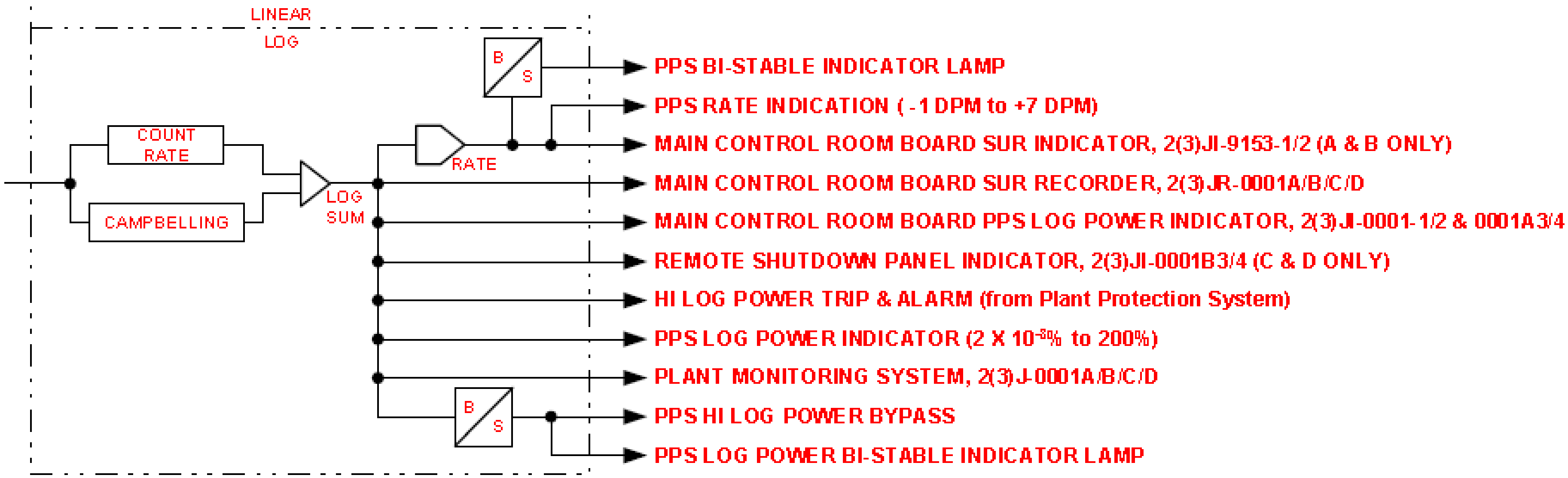
Technical Reference(s) SD-SO23-470, Figures 1A and 1B; SO23-3-2.15, L&S 4.2.1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

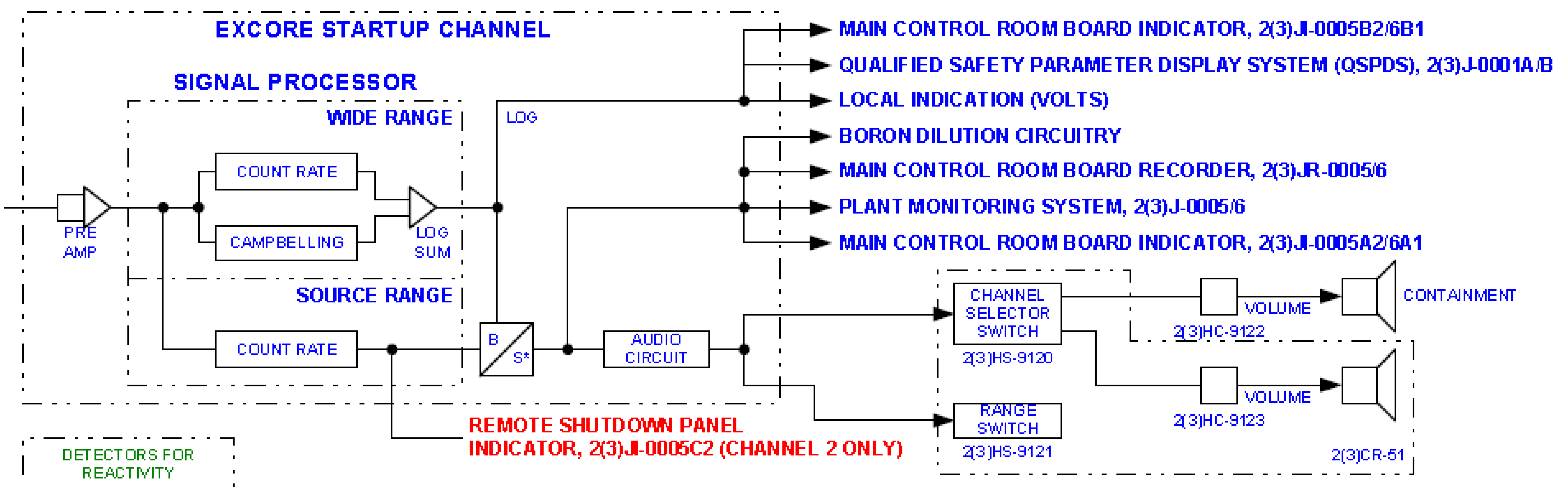
Learning Objective: DESCRIBE the Excore Nuclear Startup Channel System instrumentation

56475	and controls, including the name, function, location, interlock and power supplies, where applicable.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 6
	55.43 ______

Comments / Reference: From SD-SO23-470, Figure 1A (Safety Channel)	Revision # 7



Comments / Reference: From SD-SO23-470, Figure 1B (Non-Safety Channel)	Revision # 7



Comments / Reference: From SO23-3-2.15, L&S 4.2.1	Revision # 13

NUCLEAR ORGANIZATION
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OPERATING INSTRUCTION
REVISION 13
ATTACHMENT 4

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3.0 **SAFETY CHANNEL INDICATOR OPERATIONS** (Continued)

3.2 The 1E-4% lamp bistable energizes, providing contact output to the PPS for use in bypassing the High Log Power trip when Reactor power is >1E-4%. In addition, contact outputs are provided to the CPCs which permit bypassing the low DNBR and High Local Power trips in order to facilitate CPC testing when Reactor power is below the setpoint. When power is >1E-4%, then the local 1E-4% lamp will be illuminated.

3.3 The 55% lamp bistable supplies the PPS which automatically bypasses the Loss of Load trip when Reactor power is below the setpoint. When >55% power (or current Loss of Load setpoint), then the bistable de-energizes and the 55% lamp will be illuminated. The 55% lamp setpoint is equal to the current setpoint for the Loss of Load Trip.

3.4 The rate lamp illuminates when the rate circuitry of the safety channel exceeds 2.5 DPM. This lamp is normally extinguished.

4.0 **SOURCE RANGE NUCLEAR INSTRUMENT OPERATION**

4.1 JI-0005 S/U Channel B, JI-0006 S/U Channel A, XS539B, and XS539A, Nuclear Instrument Power Supply Transfer Switches, are Safe Shutdown Components. (LCS 3.7.113.1)

4.2 One source range neutron flux monitor <u>shall</u> be operable in Modes 1, 2 and 3, with readout capable of being displayed external to the control room. (Tech. Spec. LCO 3.3.11, Table 3.3.11-1 Item 1 and LCO 3.3.12, Table 3.3.12-1 Item 1.a)

4.2.1 The ONLY Startup Channel which meets Tech. Spec. 3.3.12, Remote Shutdown Instrumentation, is Channel B, JI-0005, because it is the only channel located remotely at L-042. Therefore, the loss of the channel results in entry into Action Condition A - restore within 30 days.

4.3 <u>When</u> Alternate Power is transferred, <u>then</u> either Alarm 57A14, FIRE ISOLATION SWITCH IN LOCAL <u>OR</u> 57B14, FIRE ISOLATION SWITCH IN LOCAL will annunciate.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	AK3.06	
Importance Rating	3.6	

Steam Generator Tube Leak: Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Normal operating precautions to preclude or minimize SGTR

Proposed Question: Common 60

Both Unit 2 and Unit 3 operate with reduced T_{COLD} bands at power and Turbine Governor Valves full open.

Which ONE (1) of the following describes the reason for operating at reduced T_{COLD}?

Reduced RCS temperatures...

A. result in lower energy release on a Main Steam Line Break at EOL which reduces the challenge to peak Containment pressure limits.

B. minimize the fouling of the Steam Generator U-tubes which optimizes secondary side heat transfer.

C. lower U-tube degradation caused by Stress Corrosion Cracking extending Steam Generator life.

D. maintain the Condenser ΔT limits associated with National Pollution Discharge Elimination Standards.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because there would be a reduction in the energy release on a MSLB but any change to prevent exceeding a design limit would be addressed by Technical Specifications not an administrative guideline.
B. Incorrect. Plausible because it could be thought that the corrosive film layer would be affected by the reduced temperature.
C. Correct. Due to Steam Generator metallurgy, operating with a reduced temperature minimizes Stress Corrosion Cracking.
D. Incorrect. Plausible because there is a limit on condenser differential temperature, however, reduced T_{COLD} is associated with stress corrosion cracking.

Technical Reference(s) SO23-5-1.7, LS-1.8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:	DESCRIBE the conditions that affect the severity of the steam generator tube rupture accident.	
Question Source:	Bank #	
	Modified Bank #	(Note changes or attach parent)
	New	X
Question History:	Last NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	10
	55.43	

Comments / Reference: From SO23-5-1.7, LS-1.8 (p77)	Revision # 41

NUCLEAR ORGANIZATION UNITS 2 AND 3	OPERATING INSTRUCTION REVISION 41 ATTACHMENT 15	SO23-5-1.7 PAGE 77 OF 86

1.0 **REACTOR COOLANT SYSTEM** (Continued)

1.8 The normal operating band for RCS Cold Leg Temperature at 100% Rated Thermal Power is 537°F and 541°F as identified in Attachment 13. The objective of the Tcold band is to maximize generation while minimizing long term Steam Generator tube degradation. Operation at the lower end of this band will prolong the life of the Steam Generators.

1.8.1 The best strategy at full power is to operate with Tcold low in the band and HP Stop and Governor Valves wide open (lower Thot). When approaching 100% power (>95%), then it is preferable to have Tcold on the program value. The goal is to operate with valves wide open (VWO) and dilute to achieve 100% power while remaining below the upper band of 541°F. For Unit 2 at full power, dilution at 540.7°F is recommended to maintain temperature high in the band.

1.8.2 Unit 2 is not able to achieve 100% power due to temperature being near the upper band of 541°F. <u>If</u> not able to stay above the minimum pressure specified by the Reload Ground Rules with temperature within the operating band, <u>then</u> Operations management should be consulted for guidance. (AR 040400300)

1.8.3 Unit 3 is able to achieve 100% power with Tc low in the operating band. However, this may cause the minimum pressure specified by the Reload Ground Rules to be challenged. <u>If</u> Steam Generator pressure approaches the minimum pressure, <u>then</u> pressure should be raised by diluting to RAISE Tc <u>and</u> throttling Turbine Governor Valves as necessary. This will result in operation slightly below 100% power, which is acceptable.

1.8.4 The Tcold Operating Band is not a limit and may be exceeded in certain cases to optimize plant performance. Engineering has stated that a Tcold of up to 545°F for periods of up to 2 weeks without having to reassess the impact of the RCS Materials is acceptable. Such cases include, but are not limited to, EOC power reductions when it is desirable to utilize MTC, during the performance of Turbine Valve Testing and/or required load reduction, and during transient conditions while attempting to maintain S/G pressure above MSIS pretrips and Reload Ground Rules limits. (AR 041101727)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	059 AK2.01	
Importance Rating	2.7	

Accidental Liquid Radwaste Release: Knowledge of the interrelations between the accidental liquid Radwaste release and the following: Radioactive liquid monitors

Proposed Question: Common 61

Given the following condition:

- The Preheater for 2/3E-354, Gas Stripper has developed a leak.

Which ONE (1) of the following Radiation Monitors would be the **first** to detect this leakage?

A. RE-7812, Radwaste Condensate Return Radiation Monitor.

B. RE-7819, Component Cooling Water Non-Critical Loop Radiation Monitor.

C. RE-7813, Liquid Waste Discharge Radiation Monitor.

D. RE-7870, Condenser Air Ejector Wide Range Radiation Monitor.

Proposed Answer: A

Explanation:

A. Correct. Although RE-7870 & RE-7813 could eventually detect the leakage (and RE-7819 if a leak were to occur in this location) this is the first monitor to sense the outflow.

B. Incorrect. Plausible because the Sample Cooler for the Radwaste Condensate Return Radiation Monitor is cooled by CCW via the Non-Critical Loop, however, a leak would need to develop in that Sample Cooler in order for this detector to sense the leak.

C. Incorrect. Plausible because Condensate Return from the Gas Strippers can be directed to the Miscellaneous Test Tank which would then be released past Radiation Monitor RE-7813, however, it is not the first detector in the flowpath.

D. Incorrect. Plausible because Condensate Return from the Gas Strippers can be directed to the Main Condenser which would then be released past Radiation Monitor RE-7870, however, it is not the first detector in the flowpath.

Technical Reference(s) SD-SO23-650, Figures 1 & 2; SD-SO23-690, Figures 8, 11, 12, & 13 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 103331 / 103328	ANALYZE normal and abnormal operations of the Radiation Monitoring System. EXPLAIN the interfaces between the Radiation Monitoring System and other plant systems.

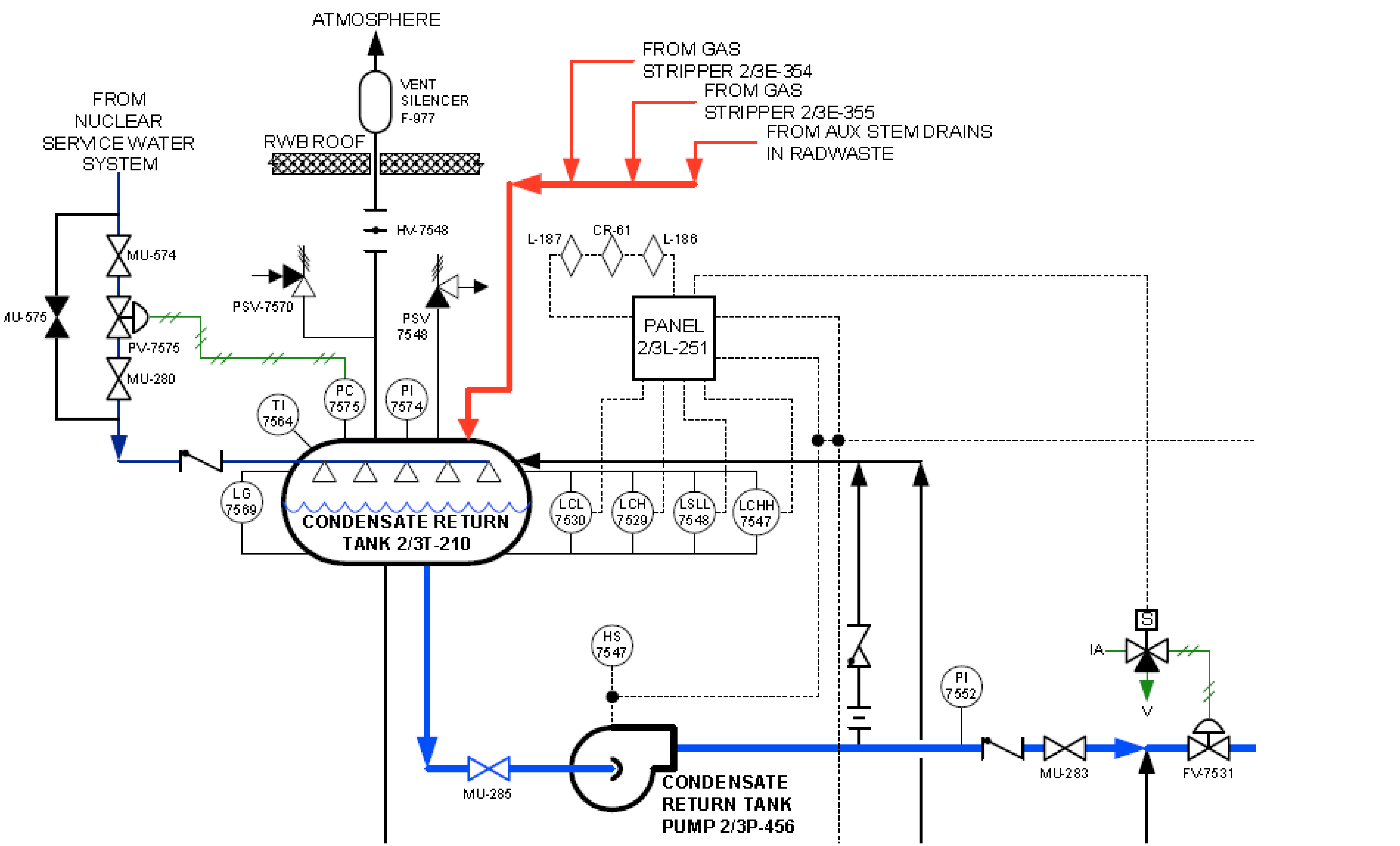
Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History: Last NRC Exam ______

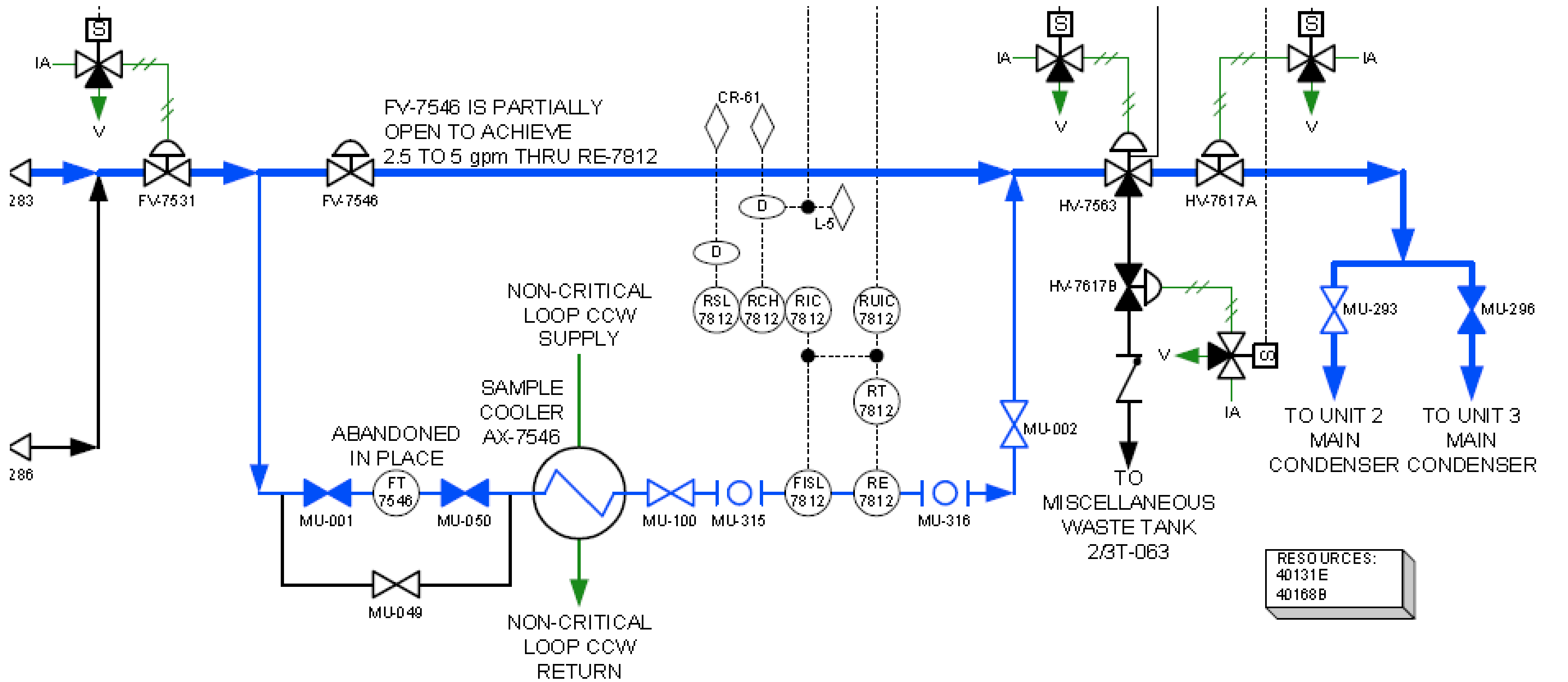
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 11
55.43 ______

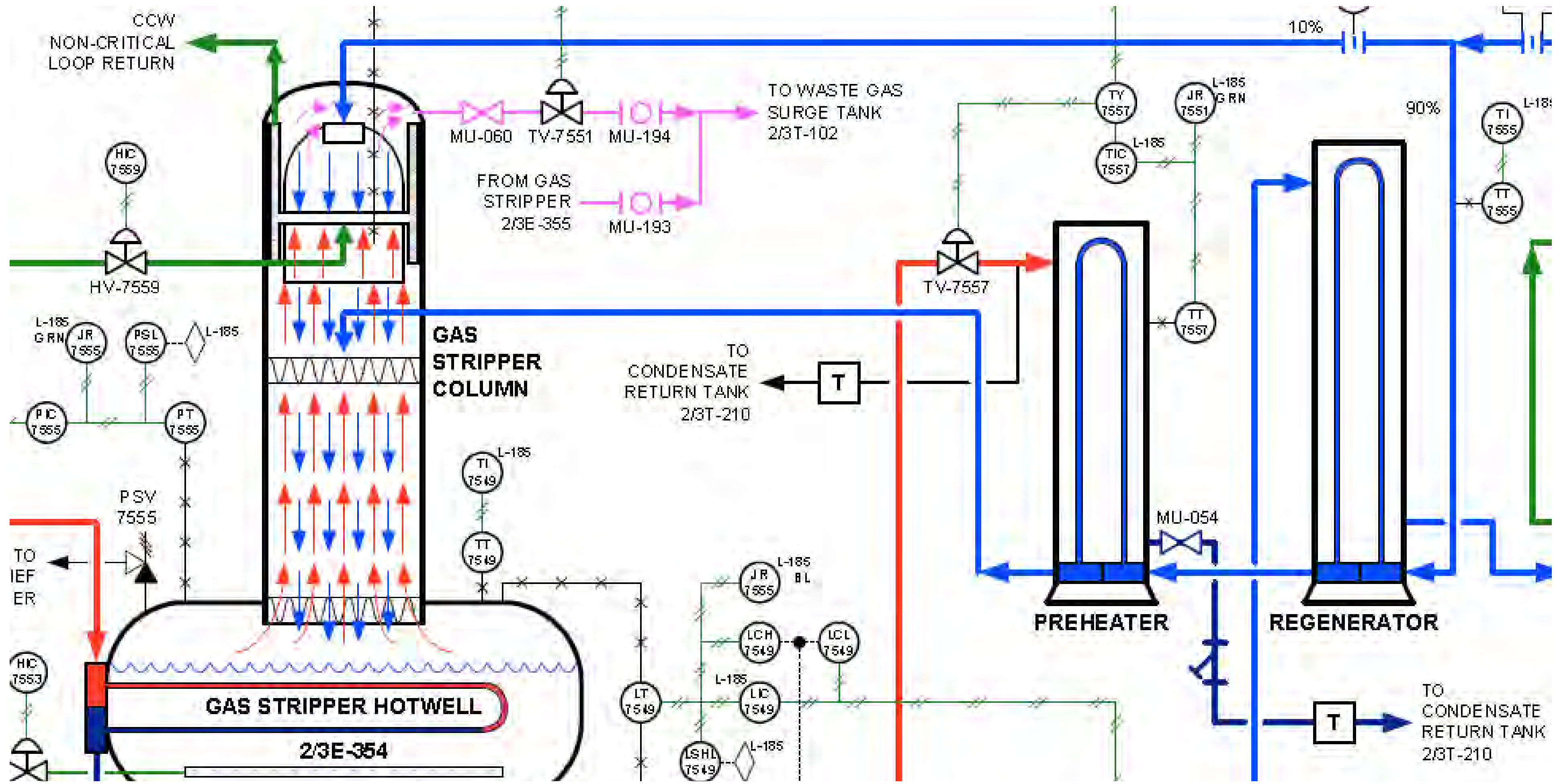
Comments / Reference: From SD-SO23-650, Figure 1	Revision # 12



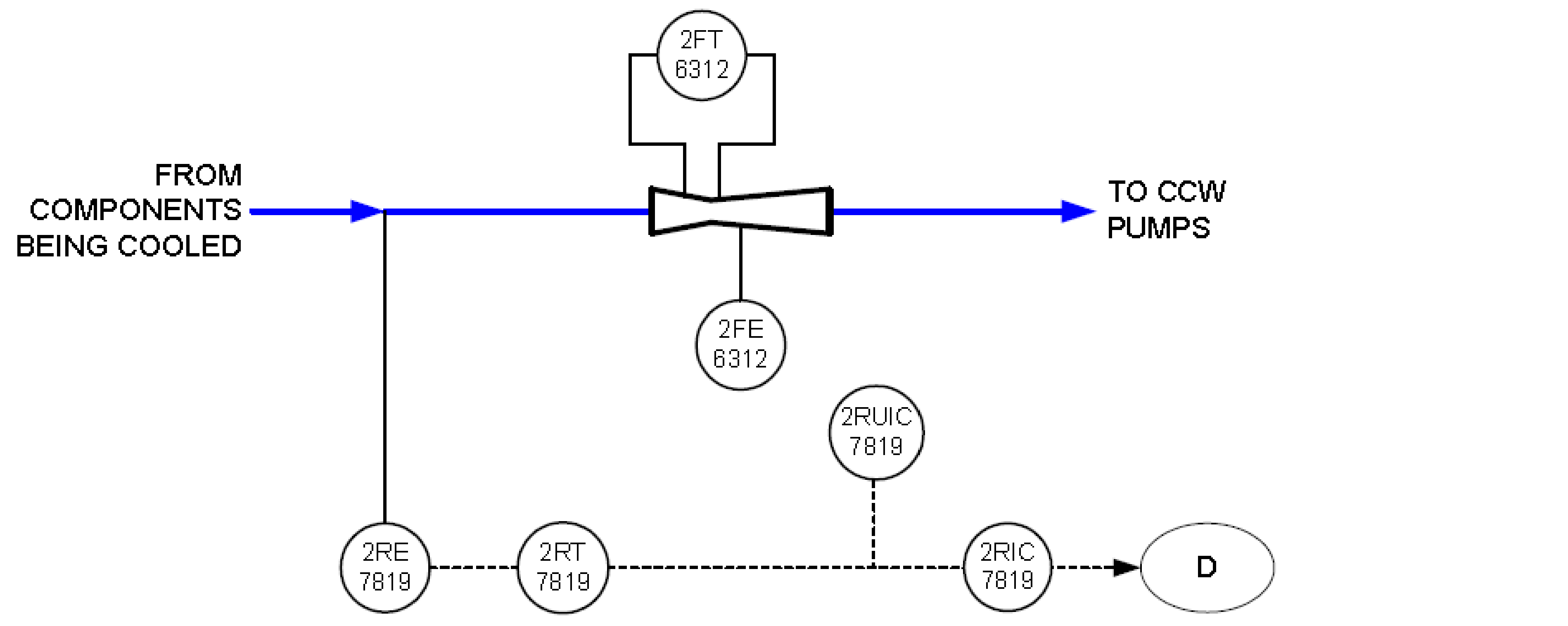
Comments / Reference: From SD-SO23-690, Figure 13	Revision # 11



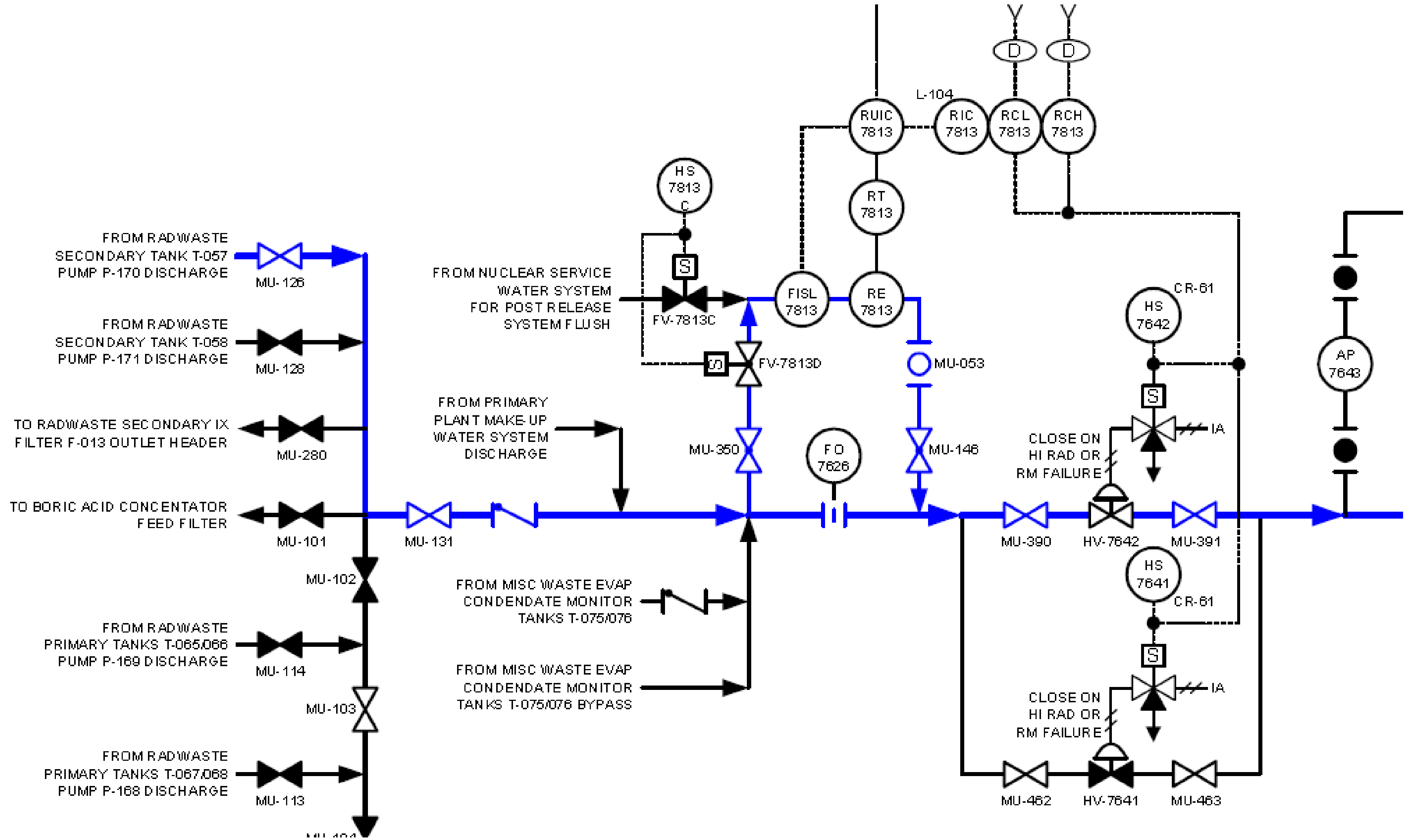
Comments / Reference: From SD-SO23-650, Figure 2	Revision # 12



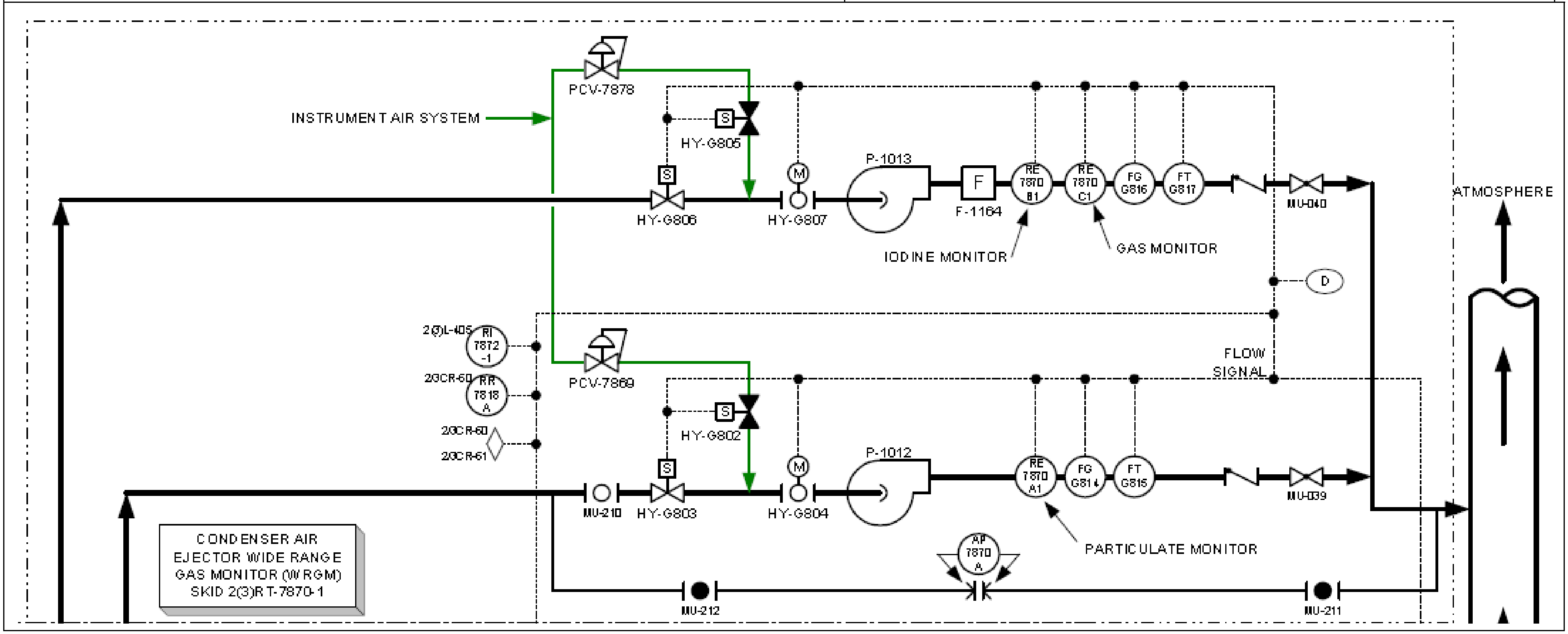
Comments / Reference: From SD-SO23-690, Figure 11	Revision # 16



Comments / Reference: From SD-SO23-690, Figure 12	Revision # 16



Comments / Reference: From SD-SO23-690, Figure 8	Revision # 16



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	067 AA2.05	
Importance Rating	3.2	

Plant Fire on Site: Ability to determine and interpret the following as they apply to the Plant Fire on Site: Ventilation alignment necessary to secure affected area

Proposed Question: Common 62

Given the following condition with Unit 2 in MODE 1:

- Annunciator 60A44 - COMPUTER ROOM SMOKE DETECTED is in alarm.

Which ONE (1) of the following describes the response to this alarm?

At Control Room Panel 2L-154, ensure the Smoke Exhaust Dampers between the Computer Room and the Control Room align to...

A. exhaust smoke from the Computer Room and that the Smoke Exhaust Fan is running.

B. isolate the Computer Room.

C. exhaust smoke from the Control Room and that the Smoke Exhaust Fan is running.

D. isolate the Control Room.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that the Smoke Exhaust Dampers would align to either the Control Room or Computer Room.
B. Correct. This is the correct response given the condition listed.
C. Incorrect. Plausible because it could be thought that the Smoke Exhaust Dampers would align to exhaust any smoke in the Control Room that might have escaped the Computer Room. This action would occur if the Control Room Smoke Detector alarmed.
D. Incorrect. Plausible because it could be thought that the Smoke Exhaust Dampers would align to isolate the Control Room from the source of smoke.

Technical Reference(s): SD-SO23-624, Page 10; SO23-15-60.A2, 60A44; SO23-15-60.B, 60B18

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 73024 / 73027	EXPLAIN the interfaces between the Fire Protection System and other plant systems.		
	ANALYZE normal and abnormal operations of the Fire Protection System.		
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7, 8	
	55.43		

Comments / Reference: From SD-SO23-624, Page 10	Revision # 1

CONTROL ROOM COMPLEX VENTILATION SYSTEM

2.1.3 **General Control Scheme** (Continued)

.1 NORMAL VENTILATION (Continued)

.1.5 The Computer Room Recirculation Units and the two air distribution fans are normally running continuously. A Temperature Controller (TC) in each computer room controls discharge temperature from the recirculation unit by positioning a chilled water, three-way valve that controls the flow of chilled water through two-thirds of the unit's cooling coil, and a solenoid valve that controls flow through the other one-third of the cooling coil. The TC also controls a heater in the humidifier to increase temperature. A moisture controller controls area humidity by controlling a humidifier and the chilled water solenoid valve in each unit.

.1.6 The Guard Room Unit has a local adjustable Temperature Controller to control unit capacity.

.2 SMOKE MODE

.2.1 If the Control Room complex smoke detector (DCH-9718) senses smoke, the smoke damper in the recirculation header shuts, the outside air smoke damper opens and the Smoke Exhaust Fan starts if outlet dampers for the fan are open.

.2.2 Each computer room has two smoke detectors. If detector DCH-9715 detects smoke, it shuts "B" train dampers HV-9715A, B, C and D (HV-9715 A and B - Unit 3), if they are in normal. Detector DCH-9734 detecting smoke, shuts "A" train dampers HV-9734A, B and C (HV-9734A, B, C and D - Unit 3), if they are in normal.

.2.3 A Halon system actuation signal from the Halon Fire Protection System (see SD-SO23-590), closes all the computer room smoke isolation dampers, regardless of control switch position.

Comments / Reference: From SO23-15-60.A2, 60A44	Revision # 14

60A44 COMPUTER ROOM SMOKE DETECTED

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	YES	61A15

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)DCH-9715	Control Building Computer Room A/C Isolation Smoke Detector Controller High	NONE	NONE	DH-9715	2172/1740 2173/1741
2(3)DCH-9734	Control Building Computer Room A/C Isolation Smoke Detector Controller High	NONE		DH-9734	

1.0 **REQUIRED ACTIONS:**

1.1 At the affected Unit, perform the following actions:

1.1.1 At Panel 2L-154, ENSURE CLOSED the following Dampers.

- 2HV-9715A, Control Room Computer Room A/C Isolation Damper
- 2HV-9715B, Control Room Computer Room A/C Isolation Damper
- 2HV-9715C, Control Room Computer Room A/C Isolation Damper
- 2HV-9715D, Control Room Computer Room A/C Isolation Damper
- 2HV-9734A, Control Room Computer Room A/C Isolation Damper
- 2HV-9734B, Control Room Computer Room A/C Isolation Damper
- 2HV-9734C, Control Room Computer Room A/C Isolation Damper

Comments / Reference: From SO23-15-60.B, 60B18	Revision # 16-1

NUCLEAR ORGANIZATION
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ALARM RESPONSE INSTRUCTION
REVISION 16 EC 16-1
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60B18 CONTROL ROOM SMOKE DETECTED

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	N/A	61A15

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2
2/3R4 [1]	2/3BU-28, 2/3MA-035 Smoke Exhaust Fan Autostart Relay	NONE	NONE	NONE	2229

1.0 REQUIRED ACTIONS:

1.1 Inspect Control Room 30 foot elevation to find the source of alarm.

1.2 At 2L-154 VERIFY the following:

1.2.1 2/3MA-035, Control Room Smoke Exhaust Fan running.

1.2.2 2/3HV-9718B, and 2/3HV-9718C, Smoke Dampers Open.

1.2.3 2/3HV-9718A, Smoke Damper Closed.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	A13 AK3.4	
Importance Rating	3.1	

Natural Circulation: Knowledge of the reasons for the following responses as they apply to the Natural Circulation Operations: RO or SRO function within the control room team as appropriate to the assigned position, in such a way that procedures are adhered to and the limitations in the facility license and amendments are not violated

Proposed Question: Common 63

Given the following conditions:

- Unit 2 has experienced a Steam Generator Tube Rupture on E088.
- All Reactor Coolant Pumps are secured due to a Loss of Offsite Power.
- Steam Generator E088 has been isolated per SO23-12-11, EOI Supporting Attachments, Attachment 27, SGTR Actions.
- Cooldown to Shutdown Cooling is required and the following parameters exist:
  - Pressurizer level is 5% and rising.
  - Reactor Coolant System pressure is 1200 psia.
  - Core Exit Saturation Margin is 21°F and rising.
  - Steam Generator E088 narrow range level is 45% and stable.
  - All ESFAS actuations occurred as designed.

Which ONE (1) of the following is required and the reason for this action?

A. Establish and maintain Pressurizer level at greater than or equal to 40% during cooldown to ensure adequate SHUTDOWN MARGIN.

B. Limit initial cooldown rates to 35-40°F per hour to optimize Natural Circulation flow in both loops.

C. Immediately reduce pressure using manual Auxiliary Spray to less than 1100 psia to prevent lifting Main Steam Safety Valves when E088 fills solid.

D. Throttle Safety Injection flow to prevent a Pressurized Thermal Shock condition with the ruptured Steam Generator isolated.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that this strategy is required for Natural Circulation cooldown, however, this is the strategy for a Natural Circulation cooldown during a Control Room Evacuation.

B. Correct. This strategy is required for Natural Circulation cooldown as identified in SO23-12-11, Attachment 3, Step 1g Note.

C. Incorrect. Plausible because for an uncontrolled increase in the isolated SG level this action would be directed, however, this pressure is not low enough to keep a Main Steam Safety Valve from lifting. The first MSSV lifts at 1105 psig.

D. Incorrect. Plausible because with the SG isolated RCS conditions are recovering and it could be thought that throttle conditions had been established and actions to prevent PTS conditions are warranted.

Technical Reference(s): SO23-12-11, Attachment 27, Step 7 Caution; SO23-13-2, Attachment 16, Step 1 Note; SO23-12-11, Attachment 3, Step 1d Caution; SO23-12-11, Attachment 3, Step 1g Note — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53298 — EXPLAIN how each of the following can affect single-phase natural circulation: Primary-to-secondary heat transfer rate.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 10
55.43 ____

Comments / Reference: From SO23-12-11, Attachment 27, Step 7 Caution	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
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ATTACHMENT 27

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EOI SUPPORTING ATTACHMENTS

SGTR ACTIONS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

7 INITIATE Lowering PZR Pressure:

NOTE

SGTR depressurization strategy should be to reduce RCS pressure while maintaining RCP NPSH T_C requirements. This should continue until RCS pressure is within 50 PSI of the ruptured S/G pressure or S/G level is not rising.

CAUTION

Keeping RCS pressure higher than S/G pressure is preferred to minimize RCS dilution due to backflow unless backflow is intended.

CAUTION

IF uncontrolled S/G level rise is occurring, THEN reducing RCS pressure to less than 1000 PSIA takes priority over maintaining RCP NPSH or 20°F Core Exit Saturation Margin. In this case stopping RCPs should be evaluated.

Comments / Reference: From SO23-13-2, Attachment 16, Step 1 Note	Revision # 11

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ATTACHMENT 16

NATURAL CIRCULATION COOLDOWN GUIDELINES

REFERENCE USE

UNIT ________ PERF. BY INITIALS

NOTE

Shutdown Margin will be ensured to be above the following limits by maintaining PZR actual (corrected) level greater than 40% with the Charging makeup water sources as specified in Section 1.0:

- ≥5.15% ΔK/K when >200°F
- ≥4.0% ΔK/K when ≤200°F (AR 080100439)

1.0 During the plant cooldown operate the RCS makeup sources, as follows:

1.1 When the first BAMU Tank reaches 20% level, then Coordinate with the Primary Operator to transfer Charging Pump suction to the to the RWST(s). ________

Comments / Reference: From SO23-12-11, Attachment 3, Step 1d Caution	Revision # 6

NUCLEAR ORGANIZATION
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ATTACHMENT 3

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EOI SUPPORTING ATTACHMENTS

COOLDOWN / DEPRESSURIZATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 INITIATE RCS Cooldown and Depressurization: (Continued)	

NOTE

Tech. Spec. Shutdown Margin may not be met during an ESDE. Frequent monitoring of Reactor Power and SUR during cooldown is desired.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
d. VERIFY adequate Shutdown Margin:	d.
1) Boration in progress – at greater than or equal to 40 GPM	1) ESTABLISH Boration – at greater than or equal to 40 GPM OR 2) MAINTAIN Shutdown Margin within limits:

Comments / Reference: From SO23-12-11, Attachment 3, Step 1g Note	Revision # 6

NUCLEAR ORGANIZATION UNITS 2 AND 3
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EOI SUPPORTING ATTACHMENTS

COOLDOWN / DEPRESSURIZATION

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

1 **INITIATE RCS Cooldown and Depressurization:** (Continued)

NOTE

IF Natural Circulation conditions exist and an Asymmetric Cooldown must be performed, THEN slower RCS cooldown rates are needed to maintain RCS flow in both loops. Normally a guideline rate of 35-40°F/HR may be established initially, then slowly increased further provided T_C ΔT does not continuously diverge. For low decay heat conditions, an initial guideline rate of 10-20°F/HR should be used.

NOTE

RCS cooldown rate determination should consider Condensate Inventory per Attachment 16, DETERMINE TIME UNTIL SHUTDOWN COOLING REQUIRED, if no Demineralized Water Storage Tanks are available.

g. DETERMINE and RECORD desired RCS cooldown band on Figure 2:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	A16 AA1.1	
Importance Rating	3.4	

Excess RCS Leakage: Ability to operate and monitor the following as they apply to the Excess RCS Leakage: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Proposed Question: Common 64

Added MISALIGNED to correct title of alarm window

Given the following conditions on Unit 2:

- The following Annunciators are in alarm:
  - 57C43 - RCS LEAKAGE ABNORMAL/RECIRC SYS VLV.
  - 56A56 - CONTAINMENT SUMP LEVEL HI.

Which ONE (1) of the following statements identifies the operation of the Containment Sump Pumps in this condition?

A. A Pump is running and will AUTO Stop on low level or SIAS/CIAS actuation.

B. A Pump will AUTO start when level reaches the HI-HI alarm and will AUTO Stop on low level or SIAS/CIAS actuation.

C. They will AUTO start when both Discharge Line Isolation Valves are opened and will AUTO Stop on low level.

D. They must be started manually after opening both discharge valves and will AUTO Stop on low level.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the pumps automatically stop on low level and it could be thought that the pumps would start on a high level alarm.
B. Incorrect. Plausible because the pumps automatically stop on low level and it could be thought that the pumps would start on hi-hi level alarm.
C. Incorrect. Plausible because the pumps automatically stop on low level, however, the pumps must be manually started.
D. Correct. There are no auto start features and the valves must be open to start the pumps. The pumps automatically stop on low level.

Technical Reference(s)	SD-SO23-670, Page 8	Attached w/ Revision # See Comments / Reference
	SO23-2-16, Step 6.19.3	
	SO23-15-57.C, 57C43	
	SO23-15-56.A, 56A56 and 56A55	

Proposed references to be provided during examination: None

Learning Objective: 81446 / 81448

DESCRIBE the configuration and operational characteristics of Containment System components.

ANALYZE normal and abnormal operations of the Containment System.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	7, 9
	55.43	

Comments / Reference: From SD-SO23-670, Page 8	Revision # 10

2.0 DESCRIPTION (Continued)

2.2.3 **Containment Normal Sump and Pumps, 2(3)P-007 & 008** (see Figure 1)

PUMP TYPE:	Vertical, centrifugal, single stage
PRIME MOVER:	480 VAC, 3 phase, 3 HP, 1800 RPM
DESIGN HEAD:	34 ft.
DESIGN NPSH:	11 ft.
DESIGN FLOW:	50 gpm
BEARING LUBRICATION:	Nuclear Service Water, 1-3 gpm
SUMP: ELEVATION:	17 ft. 6 in.
DIMENSION:	4.5 ft. x 4.5 ft. x 6.0 ft. deep
CAPACITY:	~900 gals.
INTERLOCKS:	2(3)HV-5803 & 5804

.1 The Containment Normal Sump and Pumps are located in the Containment inside the bio-shield and collect equipment and floor drainage from Containment.

.1.1 The pump discharge is directed to the Radwaste Sump.

.2 The Containment Sump Pumps are operated manually from handswitches in the Control Room Panel, 2(3)CR-56. The Containment Sump Pump(s) will auto stop on a Sump level of <18" from 2(3)LCHL-5801.

.3 The Containment Normal Sump is equipped with a level transmitter which provides sump level indication in the Control Room.

.3.1 Two redundant channel level transmitters send separate signals to the Control Room for post-Loss of Coolant Accident (LOCA) level indication and recording.

.3.2 The sump level signal from the 'B' Train Level Transmitter, 2(3)LT-5853-2, also inputs to the "Plant Monitoring System" (PMS) and the "Critical Functions Monitoring System" (CFMS).

Comments / Reference: From SO23-2-16, Step 6.19.3	Revision # 23

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6.0 PROCEDURE (Continued)

6.19.3 Initiate pumping Containment Sump to the Radwaste Sump, as follows:

.1 Obtain the Radwaste Operator's concurrence to pump to the Radwaste Sump. (See Attachment 34) ☐

.2 OPEN HV-5803, Sump Pump Containment Isolation Valve. ☐

.3 OPEN HV-5804, Sump Pump Containment Isolation Valve. ☐

.4 OPEN HV-7911, Nuclear Service Water to Containment. ☐

.5 START the Containment Sump Pumps. (One or both) ☐

☐ MP-007 (HS-5801A) ☐ MP-008 (HS-5801B)

.6 VERIFY receipt of 56A45 (56A46), CONTAINMENT SUMP PUMP P007 (P008) RUNNING, annunciator. ☐

6.19.4 When desired Containment Sump level is reached:

.1 STOP the running Containment Sump Pump(s). ☐

☐ MP-007 (HS-5801A) ☐ MP-008 (HS-5801B)

.2 VERIFY 56A45 (56A46), CONTAINMENT SUMP PUMP P007 (P008) RUNNING, annunciator resets. ☐

.3 CLOSE HV-7911, Nuclear Service Water to Containment. (N/A if being maintained open per SO23-5-1.8 for outage support.) ☐

.4 CLOSE HV-5803, Sump Pump Containment Isolation Valve. ☐

.5 CLOSE HV-5804, Sump Pump Containment Isolation Valve. ☐

Comments / Reference: From SO23-15-57.C, 57C43	Revision # 18

NUCLEAR ORGANIZATION
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SO23-15-57.C
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57C43 RCS LEAKAGE ABNORMAL/RECIRC SYS VV MISALIGNED

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	YES	57C10, 57C20

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
[1]	Containment Sump In Leakage	1 GPM or 0.5 GPM increase	C_KFCSTCHG [2]	NONE	208/208
	RCDT In Leakage	5 GPM	C_KFRDTCHG [2]		
	Containment Emergency Sump Valves	Not fully Closed with no RAS	2(3)HS-9302-2 2(3)HS-9303-1 2(3)HS-9304-2 2(3)HS-9305-1	ZL9302-2 ZL9303-1 ZL9304-2 ZL9305-1	
	Safety Injection and Containment Spray Mini-Flow Valves	Not fully Opened with no RAS	2(3)HS-9306-1 2(3)HS-9307-1 2(3)HS-9347-2 2(3)HS-9348-2	ZH9306-1 ZH9307-1 ZH9347-2 ZH9348-2	
	48V Power Failure on L096	N/A	N/A	NONE	
N/A	Main Containment Purge Valves	Closed when in Midloop	C_ZL99502 [2] C_ZL99511 [2]		

1.0 **REQUIRED ACTIONS**:

NOTES:

1. "Reactor Coolant Pressure Boundary" is initiated by a high flow rate from one or more of the normal inputs to the Reactor Coolant Drain Tank.
2. "Unidentified Leakage Abnormal" is initiated by a high flow rate into the Normal Containment Sump.

1.1 Check CFMS Page 122 to determine the source of this alarm.

1.2 Check PCS message logs for cause of this alarm.

1.3 If alarm is due to draining a component to the Containment Sump during an outage, then no response is necessary.

[1] Alarm is CFMS generated.

[2] PCS Point ID (AR 061001335)

Comments / Reference: From SO23-15-56.A, 56A56	Revision # 7

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56A56 CONTAINMENT SUMP LEVEL HI

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	WHITE	N/A	56A55

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)LCHL-5801	Containment Sump Level Control	4'0" [1]	2(3)LR-5853 CFMS Page 122	NONE	436/463

1.0 **REQUIRED ACTIONS**:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Normal Inleakage	2.1 Monitor 2(3)LR-5853, Containment Sump Recorder and/or CFMS Page 122 for indication of sudden or unexplained changes in rate of level change or flow. 2.1.1 If CFMS Delta Flow is >1 gpm, then perform SO23-3-3.37, Reactor Coolant System Water Inventory Balance. 2.1.2 If CFMS 24 hour or 30 minute average is >1 gpm, then Notify Maintenance Engineering. 2.1.3 If leakage is determined to be normal, then Pump the Containment Sump per SO23-2-16, Section for Pumping the Containment Normal Sump.
2.2 Sudden Excessive Inleakage	2.2 Attempt to Identify the source of the inleakage. 2.2.1 Notify Chemistry to sample the Containment Sump for Boron, Nitrates, Hydrazine, and Amines.

[1] The 4'0" setpoint corresponds to 66.5% on 2(3)LI-5839 (CR-56), and 14'6" on 2(3)LI-5853-1 and 2(3)LI-5853-2 (CR-57)

Comments / Reference: From SO23-15-56.A, 56A55	Revision # 7

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56A55 CONTAINMENT SUMP LEVEL HI-HI

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes ALL	AMBER	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)LSHH-5800	Containment Sump HI-HI Level	5'0" [1]	2(3)LR-5853	NONE	435/462

1.0 **REQUIRED ACTIONS**:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Containment Sump Inleakage	2.1 Pump the Containment Sump per SO23-2-16, Section for Pumping the Containment Normal Sump. 2.1.1 Verify HI/HI level alarm resets. 2.1.2 If the frequency of pumping the Containment Sump has increased, then monitor the Containment Sump Recorder 2(3)LR-5853 and/or CFMS on a more frequent basis and coordinate with Chemistry to identify the source of inleakage.

3.0 ASSOCIATED RESPONSES:

3.1 Notify the CRS/SM and the STA to review Tech. Specs. LCO 3.4.13, LCO 3.4.14, and initiate an EDMR/LCOAR, as required.

3.2 Notify the CRS/SM and the STA to review the EPIPs and SO123-0-A7, and perform notifications as required.

[1] The 5'0" setpoint corresponds to 85% on 2(3)LI-5839 (CR-56), and 15'6" on 2(3)LI-5853-1 and 2(3)LI-5853-2 (CR-57)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	068 AA1.28	
Importance Rating	3.8	

Control Room Evacuation: Ability to operate and / or monitor the following as they apply to the Control Room Evacuation: PZR level control and pressure control

Proposed Question: Common 65

Given the following condition:

- The Control Room has been evacuated due to a seismic event.

Which ONE (1) of the following represents the location and reason for the monitoring of Pressurizer level?

A. Evacuation Shutdown Panel (L-042) because the instrument range is adequate to maintain the plant in a HOT STANDBY condition.

B. Essential Plant Parameters Monitoring Panel (L-411) because the instrument range is adequate to place the plant in a COLD SHUTDOWN condition.

C. Essential Plant Parameters Monitoring Panel (L-411) because the panel is located in the seismically qualified Penetration Building.

D. Evacuation Shutdown Panel (L-042) because the Pressurizer level instrument retains its seismic qualification at this location.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Evacuation Shutdown Panel is the correct location, however, the reason is because its seismic qualification is maintained.
B. Incorrect. Plausible because achieving Cold Shutdown is a desired condition during a seismic event, however, this panel is used when a fire is the reason the Control Room was evacuated.
C. Incorrect. Plausible because the EPPM is located in the Penetration Building which has seismic qualifications, however, the EPPM itself is not seismically qualified.
D. Correct. Pressurizer level is monitored from the Evacuation Shutdown Panel during a seismic event given the design criteria associated with this indication. The Essential Plant Parameters Monitoring Panel would be used if a fire had occurred.

Technical Reference(s) SO23-13-2, Step 4.10.5 Note Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56667 As the RO, DESCRIBE: The basis for each step, note and caution, per the applicable sections and attachments of the Shutdown from Outside the Control Room procedure, SO23-13-2.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Comments / Reference: From SO23-13-2, Step 4.10.5	Revision # 11

NUCLEAR ORGANIZATION ABNORMAL OPERATING INSTRUCTION SO23-13-2
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4.0 SUBSEQUENT OPERATOR ACTIONS (Continued) PERF. BY INITIALS

NOTES

1. Due to design criteria, PZR level monitoring location is dictated by the type of shutdown event. (Ref. 5.2.2)
2. If a ***seismic event*** has occurred, then PZR level monitoring must be done at the EVSD (L-042).
3. If a ***fire event*** has occurred, then PZR level monitoring must be done at the EPPM (L-411).

4.10.5 Ensure Pressurizer actual (corrected) level is between 10% and 70%, and trending to between 40% and 50%: (Ref. 5.2.2)

.1 Direct the 21/31 to coordinate with the 22/32 and control PZR level by operating Charging Pumps using the Auxiliary Control Switches. ________

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G 2.1.21	
	Importance Rating	3.5	

Conduct of Operations: Ability to verify the controlled procedure copy

Proposed Question: Common 66

Which ONE (1) of the following activities does NOT require verification that the procedure is the most current revision?

Performing...

A. an evolution in the Main Control Room with a goldenrod controlled copy of the procedure.

B. an evolution at the FFCPD Control Room with a field controlled copy of the procedure.

C. a frequently performed evolution in the Main Control Room with a laminated copy of the procedure.

D. an evolution at the FFCPD Control Room with a multiple use procedure in a plastic jacket.

Proposed Answer: A

Explanation:

A. Correct. This type of procedure is considered a controlled document.
B. Incorrect. Plausible because it could be thought that these types of procedure copies were controlled documents.
C. Incorrect. Plausible because it could be thought that these types of procedure copies were controlled documents.
D. Incorrect. Plausible because it could be thought that these types of procedure copies were controlled documents.

Technical Reference(s) SO123-0-A3, Section 6.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55120	STATE the guidelines for procedure use during normal operations including: "Procedure-In-Use" file per Operation's Administrative Procedures.
Question Source:	Bank # ______
	Modified Bank # ______ (Note changes or attach parent)
	New X
Question History:	Last NRC Exam ______
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ______
10 CFR Part 55 Content:	55.41 10
	55.43 ______

Comments / Reference: From SO123-0-A3, Section 6.3	Revision # 8

6.0 PROCEDURE (Continued)

6.3 User Controlled and Field Controlled Documents

6.3.1 Operations has classified the following types of procedures and activities as User-Controlled documents per SO123-VI-0.9:

- Procedure pages that are posted locally at plant locations. These are printed on buff (yellow) colored paper and stamped "OPERATIONS USER CONTROLLED DOCUMENT."
- Procedure steps and activities performed on Hand-held computers

.1 Posted procedure pages and hand-held computer steps/activities are maintained up-to-date with CDM controlled procedures by the Operations Procedure Group. Consequently, field verification that these documents are up-to-date is not required.

.2 Posted procedure page locations are identified in PRO-16, Posted User Controlled and Field Controlled Procedures.

6.3.2 Operations has Field Controlled procedures located in the FFCPD Control Rooms, and HFMUD Control Room.

.1 Field Controlled procedures are ensured to be up-to-date with CDM controlled procedures by the Operator prior to use.

.2 Field Controlled procedures are copies of controlled documents which are laminated or placed inside plastic jackets to allow multiple use.

.3 Authorized Field Controlled procedures for the FFCPD and HFMUD Control Rooms are identified in PRO-16, Posted User Controlled and Field Controlled Procedures.

6.3.3 Operators in the Main Control Room may maintain high use procedure Sections in plastic jackets or laminated "hard cards" for convenience.

.1 Control Room jacketed or laminated procedures are ensured to be up-to-date with CDM controlled procedures by the Operator prior to use.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	1	
K/A #	G 2.1.15	
Importance Rating	2.7	

Conduct of Operations: Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

Proposed Question: Common 67

Given the following conditions:

- SO23-3-3.19, 4 kV Emergency Bus Transfer Test, Attachment 5, SSD Second Point of Control Tests - 2G002 Output Breaker 2A0413 is in progress.
- 2G002 Breaker 2A0413 fails to close when operated at the Train A Second Point of Control.

In accordance with SO23-3-3, Operations Surveillance Program Requirements, which ONE (1) of the following is the **first** person that needs to be notified about the breaker failure?

A. Shift Manager.

B. Operations Manager.

C. Cognizant System Engineer.

D. SRO Operations Supervisor.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Shift Manager is concerned especially if it affected component OPERABILITY, however, it is the SRO Operations Supervisor that is responsible per SO23-3-3.
B. Incorrect. Plausible because at some point the Operations Manager would be informed, however, it is the SRO Operations Supervisor that is responsible per SO23-3-3.
C. Incorrect. Plausible because at some point this individual would be informed, however, it is the SRO Operations Supervisor that is responsible per SO23-3-3.
D. Correct. As defined in SO23-3-3, Operations Surveillance Program Requirements.

Technical Reference(s) SO23-3-3, Steps 6.1.5 and 6.5.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55161	Given a plant situation involving the performance of a surveillance test, DETERMINE if administrative requirements for surveillances have been met and what corrective action, if any, is required.		
Question Source:	Bank #	127272	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: From SO23-3-3, Step 6.1.5 and 6.5.1	Revision # 12

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6.0 PROCEDURE (Continued)

6.1.3 The Surveillance/Compliance Coordinator is responsible for:

- Generating the daily Surveillance Control Sheets for the operating shifts
- Producing the Technical Specification Surveillance Report for the cognizant Equipment Control and Operations Supervisors
- Auditing completed surveillance documents for deficiencies
- Implementing approved changes to the OSCAR Program

6.1.4 The SRO Ops. Supervisor is responsible for:

- Authorizing Surveillance Testing after assessing its impact on Unit reliability or Generating Capacity
- Taking action to maintain/restore equipment Operability
- Reporting schedule deviations to the Surveillance/Compliance Coordinator
- Reporting unsatisfactory surveillance results to the Shift Manager
- Initiating a LCOAR/EDMR if applicable
- Reviewing the completeness, accuracy (where applicable), and results of Surveillance Testing procedures

6.1.5 The Balance of Plant Operator is responsible for:

- Completing the Surveillance Testing as scheduled
- Reviewing the completed surveillance procedures and informing the SRO OPS. Supervisor of deficiencies or difficulties encountered
- Reporting any reason why a scheduled surveillance will not be completed on time

Comments / Reference: From SO23-3-3, Step 6.5.1	Revision # 12

6.0 PROCEDURE (Continued)

6.4.5 The daily Surveillance Control Sheets will be completed, as follows:

.1 The Operator who actually performed the surveillance will initial the Surveillance control sheets, as applicable.

.2 After verifying all completed documents are placed in the CRS Surveillance Hold Box, and ensuring all uncompleted surveillances are rescheduled, then the NCO initials the Surveillance control sheets for the current shift.

.3 Just before turnover to DAY shift, the NIGHT shift NCO. will:

- Review the violation Date/Time to ensure the remaining completion time (plus any allowable extension time) is sufficient to complete the surveillance requirement
- Reschedule any uncompleted Technical Specification surveillances from the previous day's Surveillance Control Sheets to the current sheets
- Notify the SRO Ops. Supervisor of any scheduled surveillance which will not be completed on time
- Note on the Surveillance Control Sheets any incomplete Non-Technical Specification surveillances with reasons for non-completion
- Transfer all completed surveillances from the CRS Surveillance Hold Box to the Red Book for retrieval by the Surveillance/Compliance Coordinator

6.4.6 The Surveillance/Compliance Coordinator updates the OSCAR program then generates the computerized Technical Specification Surveillance Report.

6.5 Surveillance Performance Guidelines

INFORMATION USE

6.5.1 Performance of Surveillance instructions, except short term surveillances (≤72 hours) and passive surveillances (plant status not changed), shall be approved by the cognizant on-shift SRO Ops. Supervisor.

.1 The SRO Ops. Supervisor's final approval accounts for current plant status and equipment Operability. This ensures the surveillance will not violate any Technical Specification LCOs nor place the unit(s) in an unsafe condition.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G 2.1.31	
	Importance Rating	4.6	

Conduct of Operations: Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup

Proposed Question: Common 68

Which ONE (1) of the following will generate the Steam Bypass Control System Annunciator 50A07 - SBCS DEMAND PRESENT?

The Steam Bypass Control System...

A. AUTO Permissive Channel Circuitry is generating an output.

B. AUTO Permissive Channel Circuitry and the Master Controller (PIC-8431) are both generating an output.

C. Master Controller (PIC-8431) is generating an output.

D. AUTO Permissive Channel Circuitry and Rate of Change Circuitry are both generating an output.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this is 1 of 2 conditions required.
B. Correct. Both conditions must be present for the annunciator to be active.
C. Incorrect. Plausible because this is 1 of 2 conditions required.
D. Incorrect. Plausible because Auto Permissive Channel Circuitry is required, however, the Rate of Change Circuitry is not required.

Technical Reference(s) SO23-3-2.18, L&S 2.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 102466 / 102485 INTERPRET instrumentation and controls utilized in the Main Steam System.
ANALYZE normal and abnormal operations of the Main Steam System.

Question Source: Bank # 128167
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Comments / Reference: From SO23-3-2.18, L&S 2.1	Revision # 20

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STEAM BYPASS SYSTEM LIMITATIONS AND SPECIFICS

1.0 **SBCS GENERAL INFORMATION** (Continued)

1.7 When in Modes 1-2 with Generator Load < 200MWe or in Modes 3-5, and steam is dumped through the SBCS Valves, then to minimize the possibility of Condenser tube damage, all 4 Circulating Pumps should be in service. (ARs 030100307-7 and 060901029)

1.7.1 If one or more CWP is out of service, then SBCS operation should be used only if ADVs are not available.

1.7.2 If SBCS must be used with only 2 or 3 CWPs in service, then use only two SBCS valves, and dump only to the Condenser section that has two CWPs running (this results in less risk to the tubes than dumping steam to both ends of the Condenser).

1.8 Prior to changing the setpoint selector switch, PIC-8431, SBCS Master Controller, should be placed in MANUAL.

1.9 To prevent opening of more than one valve from a single component failure, minimize the time the SBCS is in MANUAL PERMISSIVE.

2.0 **SBCS RESPONSE**

2.1 When the SBCS Master Controller (PIC-8431) receives a demand signal, then 50A07, SBCS DEMAND PRESENT, annunciates. Typically, the initiation of steam bypass flow demand occurs as the power descent approaches 15%.

2.1.1 When both the AUTO PERMISSIVE Channel and the SBCS Master Controller (PIC-8431) are generating an output, then 50A07, SBCS DEMAND PRESENT, will annunciate.

2.2 SBCS Valves due to a known valve design deficiency may exhibit a jerky motion when valves are going closed between 45% to 20% of valve stroke. (AR 040100064)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	G 2.2.22	
Importance Rating	4.0	

Equipment Control: Knowledge of limiting conditions for operations and safety limits

Proposed Question: Common 69

Which ONE (1) of the following conditions requires entry into a Technical Specification ACTION STATEMENT while in MODE 3?

A. Pressurizer level is 30%.

B. Charging Pump P191 is cleared.

C. Train A Fuel Oil Storage Tank level is 39,000 gallons.

D. Reactor Coolant System T_{COLD} is 535°F.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 38% Pressurizer level is the minimum per the Pressurizer Level Control Program, however, there is no Technical Specification minimum value for level.
B. Incorrect. Plausible because two RCS boron injection flow paths must be OPERABLE, however, Charging Pump P191 is the Swing Pump and can be cleared without impact to boration systems.
C. Correct. This level meets the minimum requirements for MODE 5 or 6, however, level must be greater than 41,800 gallons in MODES 1, 2, 3 or 4.
D. Incorrect. Plausible because this temperature is less than the average temperature required at Hot Zero Power, however, the Technical Specification Minimum Temperature for Criticality is 522°F.

Technical Reference(s):
- Technical Specification LCO 3.8.3
- Technical Specification LCO 3.4.2
- SO23-3-1.10, Attachment 5
- Technical Specification LCO 3.1.9

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649

Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

Question Source:	Bank #		
	Modified Bank #	73789	(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	SONGS 2007	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: From Technical Specification LCO 3.8.3	Amendment # 211

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil, Lube Oil, and Starting Air

LCO 3.8.3 The stored diesel fuel oil, lube oil, and starting air subsystem shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with fuel volume < 48,400 gallons and > 41,800 gallons in storage tank during MODE 1,2,3 or 4.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory < TS min and ≥ TS inop.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One required DG with fuel volume in the storage tank < 43,600 gallons and > 37,400 gallons during MODE 5 or 6.	C.1 Restore fuel oil level to within limits.	48 hours
D. One or more DGs with stored fuel oil total particulates not within limits.	D.1 Restore fuel oil total particulates to within limits.	7 days

Comments / Reference: From Technical Specification LCO 3.4.2	Amendment # 127

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop cold leg temperature (T_c) shall be ≥ 522°F.

APPLICABILITY: MODE 1, THERMAL POWER ≤ 30% RTP and T_c < 535°F, and
MODE 2, K_{eff} ≥ 1.0 and T_c < 535°F.

ACTIONS

Comments / Reference: From SO23-3-1.10, Attachment 5	Revision # 21

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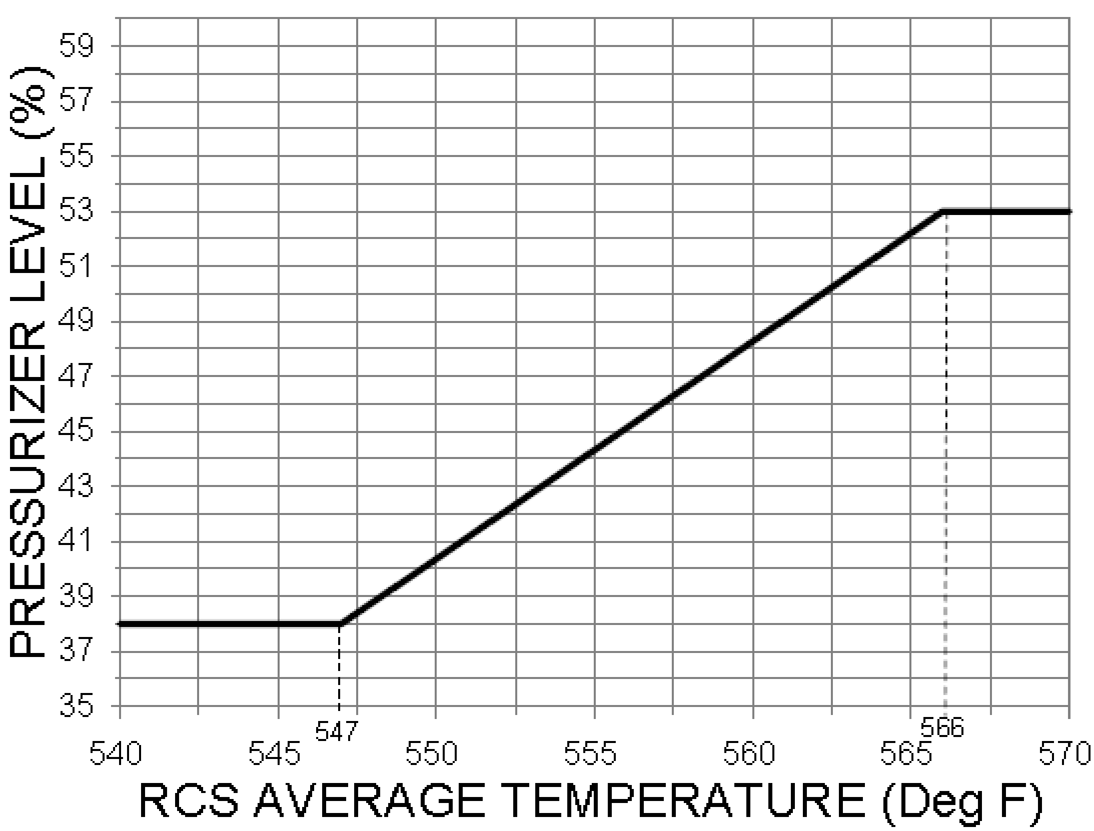
PRESSURIZER LEVEL CONTROL PROGRAM



Comments / Reference: From Technical Specification LCO 3.1.9	Amendment #127

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 Boration Systems - Operating

LCO 3.1.9 Two RCS boron injection flow paths shall be OPERABLE with the contents of the Boric Acid Makeup (BAMU) tanks in accordance with the LCS.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One boron injection flow path INOPERABLE.	A.1 Restore boron injection flow path to OPERABLE.	72 hours

Comments / Reference: From Exam Bank #73789	Revision # 05/22/01

Which ONE (1) of the following conditions requires entry into a Technical Specification ACTION STATEMENT while at 80% power?

A. <u>Diesel Generator 2G002 Fuel Oil Day Tank level is 29 inches.</u>

B. Fuel Oil Storage Tank 2T-035 level is 45,850 gallons.

C. Reactor Coolant System Tcold is 541ºF.

D. Charging Pump 2P-191 is tagged out for repairs.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G 2.2.39	
	Importance Rating	3.9	

Equipment Control: Knowledge of less than or equal to one hour Technical Specification action statements for systems

Proposed Question: Common 70

Given the following conditions:

- Unit 2 is in a Refueling Outage.
- Spent fuel is being moved inside Containment.
- Refueling Cavity level is greater than 23 feet above the fuel.

Which ONE (1) of the following is the required condition for the Personnel Air Lock (PAL) doors, Emergency Air Lock (EAL) doors, and Equipment Hatch?

A. The Equipment Hatch must be closed and held in place by 2 bolts; both doors of the PAL and EAL must be closed.

B. The Equipment Hatch must be closed with all bolts installed; both doors in the PAL and EAL must be closed.

C. The Equipment Hatch must be closed and held in place by 4 bolts; both doors in the EAL and PAL must be closed.

D. The Equipment Hatch must be closed and held in place by 4 bolts; one door in the EAL must be closed, and one door in the PAL must be closed or OPERABLE.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because if the Equipment Hatch were OPERABLE (requires 4 bolts) this condition would be correct, however, only one door in the PAL and EAL must be closed.
B. Incorrect. Plausible because the Equipment Hatch would be OPERABLE with all bolts installed, however, only one door in the PAL and EAL must be closed.
C. Incorrect. Plausible because the Equipment Hatch requirements are correct, however, only one door in the PAL and EAL need be closed.
D. Correct. Per Technical Specification LCO 3.9.3.

Technical Reference(s) Technical Specification LCO 3.9.3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649	Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).
Question Source:	Bank # 127083
	Modified Bank # ____ (Note changes or attach parent)
	New ____
Question History:	Last NRC Exam SONGS 2005A
Question Cognitive Level:	Memory or Fundamental Knowledge X
	Comprehension or Analysis ____
10 CFR Part 55 Content:	55.41 10
	55.43 ____

Comments / Reference: From Technical Specification LCO 3.9.3	Amendment # 193

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

---------------NOTE------------------

The equipment hatch may be open if all of the following conditions are met:

1) The Containment Structure Equipment Hatch Shield Doors are capable of being closed within 30 minutes,
2) The plant is in Mode 6 with at least 23 feet of water above the reactor vessel flange,
3) A designated crew is available to close the Containment Structure Equipment Hatch Shield Doors,
4) Containment purge is in service, and
5) The reactor has been subcritical for at least 72 hours.

----

b. One door in each air lock closed;

---------------NOTE------------------

Both doors of the containment personnel airlock may be open provided:

a. one personnel airlock door is OPERABLE, and

b1. the plant is in MODE 6 with 23 feet of water above the fuel in the reactor vessel, or

b2. defueled configuration with fuel in containment (i.e., fuel in refueling machine or upender).

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	2	
K/A #	G 2.2.41	
Importance Rating	3.5	

Equipment Control: Ability to obtain and interpret station electrical and mechanical drawings

Proposed Question: Common 71

Given the following condition:

- An overload condition has occurred on Reserve Auxiliary Transformer 2XR1

Per Drawing 30103, which ONE (1) of the following are automatic functions initiated by operation of Overcurrent Relay, 451-1?

Along with initiating an Annunciator,...

A. trips 230 kV CBs, trips 6.9 kV CBs, and initiates DDSMS.

B. trips 230 kV CBs, trips 6.9 kV CBs, and initiates DFR1 Channel 23.

C. trips 6.9 kV CBs, trips 4.16 kV CBs, and initiates DFR1 Channel 22.

D. trips 230 kV CBs, trips 480 V CBs, and initiates 230 kV Breaker Failure Backup.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because tripping of the breakers is correct, however, this device does not initiate DDSMS.
B. Correct. Per the Function Table on Drawing 30103 and overcurrent relay 451-1.
C. Incorrect. Plausible because tripping of the breakers is correct, however, this device actuates DFR1 Channel 23.
D. Incorrect. Plausible because the 230 kV Breakers and Breaker Failure Backup are actuated by the overcurrent device, however, there are no 480 V circuit breakers that are tripped.

Technical Reference(s): Drawing 30103, 451-1 Overcurrent Relay; Drawing 30103, Function Table — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Drawing 30103 (Size “C”)

Learning Objective: 54937 — Given the appropriate Elementary diagram, DETERMINE starting, stopping, overriding, indicating, protecting, and interlock criteria, as applicable, for any component.

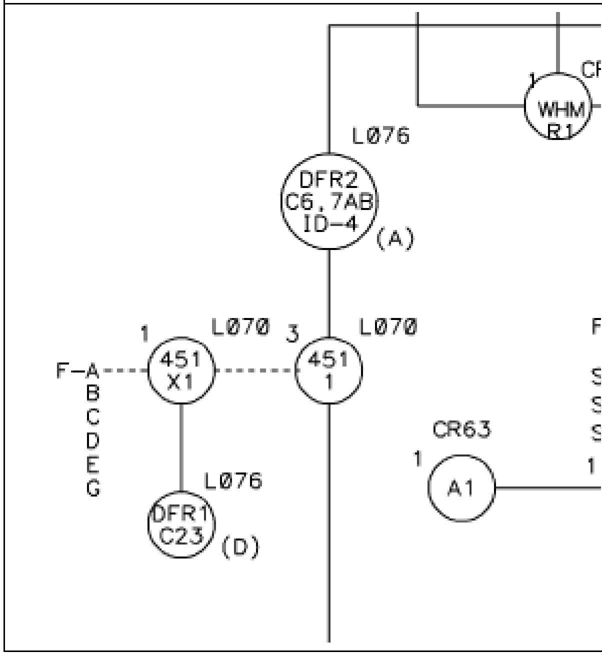
Question Source: Bank # 73728
Modified Bank # ________ (Note changes or attach parent)
New ________

Question History: Last NRC Exam ________

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

10 CFR Part 55 Content: 55.41 7
55.43 ________

Comments / Reference: From Drawing 30103, 451-1 Overcurrent Relay	Revision # 11



Comments / Reference: From Drawing 30103, Function Table	Revision # 11

FUNCTION TABLE	
F–A	ANNUNCIATION & COMPUTER INPUT
B	TRIP 230KV RES AUX XFMR PCB'S
C	TRIP 6 9KV RES AUX XFMR PCB'S
D	TRIP 4 16 KV RES AUX XFMR PCB'S
E	TRIP RES AUX XFMR COOLING
F	XFMR FAN CONTROL
G	230KV RES AUX PCB FAILURE B.U. PROT INITIATION
Q	COMPUTER INPUT
S	DDSMS

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G 2.3.12	
	Importance Rating	3.2	

Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: Common 72

Given the following conditions on Unit 3:

- Unit 3 is in a Refueling shutdown with a Reduced Inventory Condition.

Which ONE (1) of the following conditions requires that the Unit 3 Containment be evacuated?

A. Notification from Security that a Direct Armed Attack (Code RED) is in progress.

B. Annunciator CONTAINMENT SUMP HI LEVEL alarms in the Control Room.

C. An inadvertent Reactor Coolant System dilution of 100 gallons.

D. Receipt of a seismic alarm at less than Operating Basis Earthquake.

Proposed Answer: A

Explanation:

A. Correct. This is the required action per SO23-13-25, Operator Actions During Security Events.

B. Incorrect. Plausible because it could be thought that receipt of this alarm would require a Containment evacuation, however, it is not required for this condition.

C. Incorrect. Plausible because it could be thought that a dilution, especially in Reduced Inventory Condition would result in increasing count rate, however, AOI SO23-13-11 Emergency Boration of the RCS/Inadvertent Dilution or Boration does not require Containment Evacuation. An increase in radiation levels could require evacuation but this condition is not specified.

D. Incorrect. Plausible because it could be thought that evacuating Containment would be prudent for a seismic event but is not an action specified in SO23-13-3, Earthquake.

Technical Reference(s): SO23-13-25, Step 12a RNO, p6; SO23-13-11, Step 5a; SO23-13-3, Attachment 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54470 — Given a SONGS Security Event, DESCRIBE basis for each step, caution or note in the Abnormal Operating Instruction and the expected plant or operator response in accordance with SO23-13-25.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 12
- 55.43 ____

Comments / Reference: From SO23-13-25, Step 12a RNO	Revision # 12-1

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ABNORMAL OPERATING INSTRUCTION REVISION 12 EC 12-1 — SO23-13-25 PAGE 6 OF 43

OPERATOR ACTIONS DURING SECURITY EVENTS

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
12 PERFORM Unit 3 Containment Actions:	
☐ a. Verify Containment is CLOSED.	☐ a. Initiate Containment Evacuation <u>and</u> Closure.
☐ b. Verify Normal and Emergency Containment lighting DE-ENERGIZED.	☐ b. DE-ENERGIZE Normal and Emergency Containment lighting (SO23-3-2.34).
☐ c. VERIFY CLOSED both Personnel Hatch doors <u>and</u> VERIFY DE-ENERGIZED the Personnel Hatch.	☐ c. CLOSE both Personnel Hatch doors <u>and</u> DE-ENERGIZE the Personnel Hatch (SO23-3-2.34).

Comments / Reference: From SO23-13-11, Step 5a	Revision # 14

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ABNORMAL OPERATING INSTRUCTION REVISION 14 | SO23-13-11 PAGE 8 OF 19

EMERGENCY BORATION OF THE RCS / INADVERTENT DILUTION OR BORATION

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5 **Inadvertent Dilution Actions:**

> ***CAUTION***
>
> While in Mode 5, Reactor Criticality can occur if an RCS dilution event continues for greater than one hour. (Ref. 1.2.1)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a. VERIFY Refueling NOT in progress.	☐ a. 1) ENSURE operations involving core alterations or positive reactivity changes are suspended. ☐ 2) IF the Refueling Cavity is being filled, ☐ THEN STOP Cavity fill, AND ☐ VERIFY fill water Boron Concentration meets requirements. ☐ 3) GO TO Step b.

Comments / Reference: From SO23-13-3, Attachment 1	Revision # 11-1

NUCLEAR ORGANIZATION ABNORMAL OPERATING INSTRUCTION SO23-13-3
UNITS 2 AND 3 REVISION 11 EC 11-1 PAGE 9 OF 33
ATTACHMENT 1

POST OPERATING BASIS EARTHQUAKE INSPECTIONS

CONTINUOUS USE

UNIT _____ MODE ______ DATE ____________ TIME ___________

1.0 PREREQUISITES — PERF. BY INITIALS

1.1 This attachment has been directed by the OPERATOR ACTIONS. ________

2.0 PROCEDURE

NOTES

1. Steps in this attachment should be performed concurrently.
2. An OBE is 50% of the Design Basis Earthquake.

2.1 If in Mode 1 or 2, then initiate normal plant shutdown. (Mark N/A if already in Mode 3, 4, 5 or 6.) ________

2.2 When in Mode 3 or 4, then initiate normal plant cooldown to Mode 5 per SO23-5-1.5. (Mark N/A if already in Mode 5 or 6.) ________

2.3 ENSURE T-120 required inventory for plant cooldown:

2.3.1 SECURE 2(3)MP-049, Condensate Transfer Pump. ________

NOTE

An REP is normally required for entry into the RWST vault, but for expediency in closing S2(3)1414MU092 it may be necessary to notify HP and proceed directly to the vault.

2.3.2 ***WITHIN 30 MINUTES***: CLOSE S2(3)1414MU092, MT-120 and MT-121 Makeup Header Isolation. (T-005 RWST Vault Under Platform.) ________

2.3.3 ***WITHIN 90 MINUTES***: CLOSE 2(3)HV-5715, Condensate Transfer Pump MP-049 Suction from MT-120. (2HV-5715 located South of BPS Sluice Pump P-431, 3HV-5715 located outside T-121 vault, near MP-049.) ________

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	3	
K/A #	G 2.3.4	
Importance Rating	3.2	

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions

Proposed Question: Common 73

Given the following conditions:

- A room, accessible to individuals, has general area radiation levels of 50 mrem/hour.
- A valve on the far wall of the room has a contact radiation level of 2000 mrem/hour.
- The radiation level 30 centimeters from the valve is 80 mrem/hour.

Which ONE (1) of the following is the correct posting for the room?

A. "CAUTION, RADIATION AREA" with the valve identified as a "HOT SPOT."

B. "CAUTION, RADIATION AREA" with no "HOT SPOT" identified.

C. "CAUTION, HIGH RADIATION AREA" with the valve identified as a "HOT SPOT."

D. "CAUTION, HIGH RADIATION AREA" with no "HOT SPOT" identified.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, the posting is correct.
B. Incorrect. Plausible because the posting is a Radiation Area is correct, however, this area also has a hot spot.
C. Incorrect. Plausible because it could be thought that the hot spot radiation area level qualified as a High Radiation Area.
D. Incorrect. Plausible because it could be thought that the general area radiation levels qualify as a High Radiation Area.

Technical Reference(s): SO123-VII-20, Attachment 1; SO123-VII-20, Step 6.11.3.3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 52585 — Given a type of Radiation Area, DESCRIBE its radiation limits, as well as the access controls required by 10CFR20 and site procedures to enter this area. Include in this description any signs and barriers that define this area.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	12	
	55.43		

Comments / Reference: From SO123-VII-20, Attachment 1	Revision # 13

NUCLEAR ORGANIZATION
UNITS 1, 2 AND 3

HEALTH PHYSICS PROCEDURE
REVISION 13
ATTACHMENT 1

SO123-VII-20
PAGE 89 OF 98

DEFINITIONS, ABBREVIATIONS AND ACRONYMS

CONTAMINATED AREA means an accessible area with general area loose surface contamination levels greater than or equal to 1000 dpm/100 cm^2 beta-gamma activity or greater than or equal to 20 dpm/100 cm^2 alpha activity. |

CRITICAL STEP means a step in a work evolution which is important to the control of a worker's exposure or control in the release of radioactive material.

DOSE QUANTITY means the type of dose being measured including DDE, LDE, SDE/WB, or SDE/ME.

DECLARED PREGNANT WOMAN (DPW) means any woman who has voluntarily informed the Cognizant HP Technical Specialist in writing of her pregnancy or intent to become pregnant and the estimated date of conception.

DEEP DOSE EQUIVALENT (DDE) is the dose to one whole body location from external sources of radiation.

DERIVED AIR CONCENTRATION (DAC) means the concentration of a given radionuclide in air which, if breathed for a working year (2000 hours), results in an intake of one ALI. |

EFFECTIVE DAC means the DAC value (μCi/cc) which is calculated based on the anticipated mixture of radionuclides present in air.

EFFECTIVE DOSE EQUIVALENT (EDE) is the dose to the whole body from external sources of radiation.

EMBRYO/FETUS means the developing unborn child of a DECLARED PREGNANT WOMAN from the time of conception until the time of birth.

ENTRANCE or **ACCESS POINT** means any location through which an individual could gain access to radiation areas or to radioactive material (10CFR20.1003).

EXTERNAL DOSE ASSESSMENT (EDA) is the method of assigning SDE initiated when a whole body count report estimates greater than 100 mrem SDE due to noble gas submersion.

EXTERNAL DOSE REVIEW (EDR) is the method of assigning dose to replace dosimeter readings. Dosimeter readings are replaced when a dosimeter is lost, damaged, offscale or when a reading is invalid or suspect.

HEALTH PHYSICS (HP) means the protection of human and the environment from the harmful effects of radiation.

HIGH CONTAMINATION AREA (HCA) means an accessible area with general area loose surface contamination levels greater than or equal to 100,000 dpm/100 cm^2 beta-gamma distributed activity or greater than or equal to 100 dpm/100 cm^2 alpha.

HIGH RADIATION AREA (HRA) means an accessible area in which an individual could receive 100 mRem deep dose equivalent in 1 hour at 30 centimeters from the source (10CFR20.1003).

Comments / Reference: From SO123-VII-20, Attachment 1	Revision # 13

NUCLEAR ORGANIZATION
UNITS 1, 2 AND 3

HEALTH PHYSICS PROCEDURE
REVISION 13
ATTACHMENT 1

SO123-VII-20
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DEFINITIONS, ABBREVIATIONS AND ACRONYMS

MONITORING REQUIRED means an individual is required to be monitored for occupational exposure at SONGS. In this case, prior dose determination and occupational dose reporting are required.

NATIONALLY TRACKED SOURCE is a sealed source that exceeds the activity thresholds in 10CFR 20 Appendix E. This does not include material contained in a fuel assembly or waste encapsulated for disposal.

NRC FORM-4 is a report of an individual's lifetime occupational exposure history.

NRC FORM-5 is a report of an individual's occupational exposure for a monitoring period.

NVLAP is the National Voluntary Laboratory Accreditation Program conducted by National Institute of Science and Technology.

OCCUPATIONAL DOSE means the dose received by an individual during the course of employment in which the individual's assigned duties involve exposure to radiation (10CFR20.1003).

OCCUPATIONAL WORKER means an individual who enters a Restricted Area at San Onofre during employment.

OWNER CONTROLLED AREA (OCA) is SONGS SCE Property to include facilities and parking lots located on the west side of Interstate 5 freeway, extending westward from Old Highway 101 to the high-tide line bordered on the north and south by the State Beach Park.

PLANT ACCESS DATA SYSTEM (PADS) is a system for tracking security clearance, training and respirator qualifications, and occupational dose records for transient workers.

PERSONNEL CONTAMINATION RECORD (PCR) DOSE EVALUATION is method of assigning dose when a PCR indicates skin contamination greater than the personnel contamination investigation level of 10,000 cpm as measured with a standard frisker.

QUALIFICATION means the condition of being authorized to perform a radiological activity. Qualifications include area access, REP specific, and respirator use. Individuals are qualified when they have attained all necessary requirements.

QUALIFIED ESCORT means an escort who has the radiological qualification for access to the applicable area.

RADIATION AREA means an accessible area in which an individual could receive 5 mRem deep dose equivalent in 1 hour at 30 centimeters from the source (10CFR20.1003).

RADIATION EXPOSURE PERMIT (REP) means a document which identifies workplace radiological hazards, establishes radiological control requirements and provides radiological control instructions to workers.

RADIOACTIVE MATERIAL AREA means an area or room where licensed material exceeding 10 times Appendix C quantities is used or stored (10CFR20.1902).

Comments / Reference: From SO123-VII-20, Step 6.11.3.3	Revision # 13

NUCLEAR ORGANIZATION
UNITS 1, 2 AND 3

HEALTH PHYSICS PROCEDURE
REVISION 13

SO123-VII-20
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6.11.3.3 In addition to the 10CFR 20 posting requirements, other postings and posting inserts may be used to communicate radiological conditions to workers such as:

- Locked High Radiation Area
- Hot Spot,
- Contaminated Area,
- ALARA Cool/Cold Zone, and/or
- Hot Particles.

.4 In addition to posting entrances into large areas or rooms, discrete areas inside the larger area which contain substantially greater hazards are also posted.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	G 2.4.2	
Importance Rating	4.5	

Emergency Procedures/Plan: Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions

Proposed Question: Common 74

Given the following conditions:

- Unit 2 is operating at 100% power.

Which ONE (1) of the following conditions would require the operators to trip the Unit?

A. Stator Hot Gas temperature of 150°F.

B. Steam Generator E089 narrow range level of 90%.

C. Instrument Air header pressure of 65 psig.

D. Main Condenser ΔT is 27°F.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that this temperature would require a plant trip, however, the Turbine would trip automatically at 181°F.
B. Correct. The Turbine should automatically trip at 89% narrow range level.
C. Incorrect. Plausible because it could be thought that this pressure would require a plant trip, however, a Unit trip must be initiated at 50 psig.
D. Incorrect. Plausible because a 25°F ΔT is an NPDES violation, however, it does not require a Unit trip.

Technical Reference(s)
SO23-15-52.A, 52A11
SO23-15-99.C, 99C01
SO23-13-5, Step 2b
SO23-2-5, L&S 2.4

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53972
As the RO, RECOGNIZE and RESPOND to a reactor trip event and manipulate plant systems and equipment to perform the Standard Post Trip Actions per SO23-12-1.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: From SO23-15-52.A, 52A11	Revision # 12

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ALARM RESPONSE INSTRUCTION
REVISION 12
ATTACHMENT 2

SO23-15-52.A
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52A11 FWCS SG2 E088 LEVEL HI OVERRIDE

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	RED	NO	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
LAHLSG2HLO (5XL)	FWCS SG2 E088 LVL HI Override	$\geq$85%	2(3)LI-1125-1	NONE	784/795

1.0 **REQUIRED ACTIONS**:

NOTE

A S/G High Level Reactor trip is automatically initiated at 89%.

1.1 If a Reactor Trip has occurred, then GO TO SO23-12-1, Standard Post Trip Actions.

Comments / Reference: From SO23-13-5, Step 2b	Revision # 7

NUCLEAR ORGANIZATION UNITS 2 AND 3 — ABNORMAL OPERATING INSTRUCTION REVISION 7 — SO23-13-5 PAGE 8 OF 34

LOSS OF INSTRUMENT AIR

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2 **Pressure less than 50 psig: (continued)**	
b. Mode 1 or 2: PERFORM the following:	
☐ 1) TRIP the ***affected Unit*** Reactor.	
☐ 2) INITIATE SO23-12-1, Standard Post Trip Actions.	
☐ 3) LOCATE and ISOLATE the leak using Attachment 2.	
☐ 4) SRO Ops. Supv. EVALUATE isolating all Instrument Air to the ***affected Unit*** per Attachment 1.	

Comments / Reference: From SO23-15-99.C, 99C01	Revision # 14

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ALARM RESPONSE INSTRUCTION
REVISION 14
ATTACHMENT 2

SO23-15-99.C
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99C01 GENERATOR HOT GAS TEMP HI TURBINE TRIP

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-3	RED	NO	NONE

INITIATING DEVICE **[1]**	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)TSH-2910A	Hot Gas Inlet to Number 1 Cooler Top	181° F	NONE	TE2910A	1240/1240
2(3)TSH-2910B	Hot Gas Inlet to Number 2 Cooler Top	181° F		TE2910B	
2(3)TSH-2910C	Hot Gas Inlet to Number 3 Cooler Top	181° F		TE2910C	
2(3)TSH-2910D	Hot Gas Inlet to Number 4 Cooler Top	181° F		TE2910D	

1.0 **REQUIRED ACTIONS**:

1.1 Verify that the Turbine Generator has Tripped by observing the following:

- Turbine Speed < 2000 rpm and Lowering
- Stop and Control Valves Closed
- Main Generator Output Breakers Open

Comments / Reference: From SO23-2-5, L&S 2.4	Revision #

NUCLEAR ORGANIZATION UNITS 2 AND 3 — OPERATING INSTRUCTION REVISION 26 ATTACHMENT 18 — SO23-2-5 PAGE 150 OF 163

CIRCULATING WATER SYSTEM LIMITATIONS AND SPECIFICS (Continued)

2.0 BUMPING A CIRC WATER PUMP

2.1 Bumping a CWP takes 20 to 30 minutes including field manipulations, with the pump actually being off for ≈10 minutes.

2.2 Consideration should be given to the effect on Condenser Vacuum due to the order that pumps are bumped:

- Good pump next to bad pump: Stopping the bad pump results in the mildest transient because good waterbox can take up the slack
- Bad pump next to bad pump: Stopping either pump results in moderate transient because both waterboxes are degraded
- Good pump next to bad pump: Stopping good pump (NOT RECOMMENDED) results in worst transient because bad waterbox cannot take up the slack

2.3 If Circ Water ΔT is near the upper limit, then starting a second Saltwater Cooling Pump will provide ≈ 0.5°F reduction in ΔT and should be considered.

2.4 LIMIT: The differential temperature of the Circulating Water Discharge and Intake shall not exceed the NPDES ΔT limit of 25°F ΔT. Under normal conditions (Non Circ. Water Heat Treat) exceeding the 25°F ΔT limit **for any time period** is a NPDES violation. The NPDES requirement for maintaining less than 25°F ΔT is exempted during Circ. Water Heat Treatment Process **only**. (Ref. 2.4.7)

2.4.1 The permit endnotes state "insignificant figures shall be rounded to the nearest significant figures." Therefore, maximum indicated ΔT is 25.4°F which rounds down to 25.

2.4.2 The Data Logger receives input from eight RTDs located in the Intake Structure. Only one RTD or PCS is required for NPDES monitoring and reportability.

2.4.3 The warmer the inlet temperature the closer to the 25°F instantaneous ΔT you will become when the CWP is secured. If additional Circ Water Pumps will be bumped, then you may have to wait a few minutes to allow the Circ Water ΔT to recover. Action should be taken to reduce load if the 25°F instantaneous ΔT is approached.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Group #	4	
K/A #	G 2.4.39	
Importance Rating	3.9	

Emergency Procedures/Plan: Knowledge of RO responsibilities in emergency plan implementation

Proposed Question: Common 75

Given the following conditions:

- An emergency event has been declared as an ALERT at 1030.
- You have assumed the duties of Operations Leader.

Which ONE (1) of the following is the LATEST time that you should activate the Emergency Response Data System (ERDS)?

A. 1045.

B. 1100.

C. 1130.

D. 1145.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Siren and PA Coordination must be performed within 15 minutes by the Operations Leader.
B. Incorrect. Plausible because personnel must be briefed by the operations leader every 30 minutes as to the status of the plant, Emergency Response Priorities, and any Protective Actions Recommendations initiated by the Emergency Coordinator
C. Correct. The Emergency Response Data System must be activated within one hour of declaration of an ALERT or higher classification.
D. Incorrect. Plausible because it could be thought that 15 minutes was allocated to Siren and PA Coordination and then one hour later the ERDS would be activated.

Technical Reference(s) SO23-VIII-30, Step 6.1.1.11.12 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56184 DESCRIBE the responsibilities of the Operations Leader during an emergency event per SO23-VIII-30, including: Providing plant status information.

Question Source:	Bank #	130572	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments / Reference: From SO23-VIII-30, Step 6.1.1.11.12	Revision # 15

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EPIP
REVISION 15

SO23-VIII-30
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6.0 PROCEDURE

6.1 Activation

NOTES: (1) See sections indicated in parentheses for detailed instructions.

(2) The Units 2/3 SM retains operational control for Unit 1 and the ISFSI.

(3) PA announcements made by Security are in addition to the requirements of this procedure.

6.1.1 Upon event declaration, or as directed by the EC, assume the Operations Leader Duties.

.1 Immediately, (within 15 minutes of event declaration) perform PA and siren coordination per Attachment 1 before continuing with this procedure.

.2 Evacuate personnel from hazardous areas (see Section 6.6).

.3 Notify EC/SED of Unit 1/ISFSI changing conditions and make recommendations in accordance with SO123-VIII-1.

.4 Use the Ivory Phone to provide plant status to other emergency facilities on the circuit (see Section 6.2).

.5 Maintain a log of decisions and actions required by EPIPs. Provide documentation of conditions, events, and communications wherever appropriate. Ensure a complete and adequate record to minimize misunderstanding and to identify items requiring followup actions.

.6 Retain operators needed for immediate in-plant response and direct auxiliary operators to report to the Operations Support Center (OSC).

.6.1 For security events, direct auxiliary operators to report to the Control Room Lunch Room.

.7 Track location and Security Badge numbers of on-shift operators (see Section 6.3).

.8 When OSC Operations Coordinator is ready, transfer tracking of location and Security Badge numbers of on-shift operators (see Section 6.3).

.9 Inform OSC Operations Coordinator when sending operators from the Control Room to the field (see Section 6.4).

.10 Contact Health Physics (HP) for in-plant radiological conditions and inform them of changes in plant conditions which may affect radiological conditions (see Section 6.5).

Comments / Reference: From SO23-VIII-30, Step 6.1.1.11.12	Revision # 15

NUCLEAR ORGANIZATION EPIP SO23-VIII-30
UNITS 2 AND 3 REVISION 15 PAGE 5 OF 16

6.1.1.11 If a Site Area Emergency (SAE) or General Emergency (GE) is declared, then prohibit eating and drinking in the Control Room until clearance has been given by HP (see Section 6.5).

.12 Within one (1) hour of an Alert or higher classification, activate Emergency Response Data System (ERDS) to NRC Operations Center in accordance with SO23-3-2.32.

.13 If emergency exposure authorization is required prior to Technical Support Center (TSC) or OSC activation, then obtain volunteers and complete the form (see Section 6.7).

.14 Brief Operations personnel assigned to Unit 1 (if applicable) approximately every 30 minutes or as conditions warrant. Include the following:

- Plant Status
- Emergency Response Priorities (i.e., repair activities)
- Onsite protective actions and Offsite PARs made by the EC

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	040 AA2.05	
Importance Rating		4.5

Steam Line Rupture - Excessive Heat Transfer: Ability to determine or interpret the following as they apply to the Steam Line Rupture: When ESFAS systems may be secured

Proposed Question: SRO 76

Given the following conditions during an Excess Steam Demand Event outside Containment and upstream of the Main Steam Isolation Valves on Steam Generator E089:

Should be Valve

- Steam Generator E089 has been isolated.
- T_{HOT} is 480°F and stable.
- Lowest T_{COLD} is 469°F and stable.
- Representative Core Exit Thermocouple is 495°F and stable.
- Pressurizer level is 39% and rising.
- Pressurizer Pressure is 1690 psia and rising.
- Auxiliary Feedwater is in service supplying E088.
- Reactor Vessel level indicates 100%.
- All CEAs inserted on the trip and power is less than $1X10^{-4}$ and lowering.

Which ONE (1) of the following describes the NEXT action of SO23-12-5, Excess Steam Demand Event for these conditions?

A. Refer to SO23-3-2.2, Makeup Operations and ensure a boration of greater than or equal to 40 gpm is maintained.

B. Refer to SO23-12-11, EOI Supporting Attachments, FS-31, Establish CVCS Letdown Flow and override and open Letdown Isolation Valves and restore Letdown to control Pressurizer level.

C. Refer to SO23-12-11, EOI Supporting Attachments, FS-7, Verify SI Throttle / Stop Criteria and stop Charging Pumps one at a time to establish Pressurizer level control.

D. Refer to SO23-12-11, EOI Supporting Attachments, FS-1, Verify Pressurizer Pressure and initiate Pressurizer Spray flow to reduce RCS and Pressurizer differential temperature.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that Shutdown Margin is in jeopardy due to the Excess Steam Demand Event.
B. Incorrect. Plausible because for Pressurizer level greater than 80% the EOI directs action to place Letdown in service, however, implementing FS-7 is a more direct way to address this concern.
C. Correct. Throttle conditions are met and with RCS pressure above SI Pump shutoff head the next throttling steps require securing Charging Pumps.
D. Incorrect. Plausible because at 160°F the EOI does direct actions to initiate spray flow for PTS mitigation, however, this action is not performed by this Floating Step.

Technical Reference(s): SO23-12-5, Steps 7 & 13; SO23-12-11, Attachment 2, FS-7; SO23-14-5, Step 7 Bases; SO23-12-10, Attachment SF-5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54789 — STATE the major recovery actions in response to an ESDE event.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 5

Comments / Reference: From SO23-12-5, Step 7	Revision # 21

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EXCESS STEAM DEMAND EVENT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

7 PREVENT Pressurized Thermal Shock:

NOTE

WHEN excess steam demand remains NOT isolated and all RCPs are stopped, THEN T_C in loop with *least affected* S/G may be higher than REP CET temperature.

CAUTION

Failure to establish steaming flow path on least affected S/G before most affected S/G loses effective heat removal capabilities will result in rapid re-pressurization (PTS consideration).

a. INITIATE FS-30, ESTABLISH Stable RCS Temperature during ESDE.

b. INITIATE FS-7, VERIFY SI Throttle/Stop Criteria.

Comments / Reference: From SO23-12-5, Step 13	Revision # 21

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REVISION 21

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EXCESS STEAM DEMAND EVENT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13 LIMIT RCS Re-pressurization:	
a. VERIFY PTS Subcooling: CFMS page 311.	a. DETERMINE PTS Subcooling using lowest RCS temperature and highest PZR pressure.
b. VERIFY PTS Subcooling – greater than 160°F AND – rising.	b. GO TO step e.

CAUTION

IF a harsh environment (escaping steam) exists inside containment, THEN changing the position of HV-9202 and HV-9203 for Auxiliary Spray control should be avoided.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. VERIFY environment in Containment – NOT harsh.	c. INITIATE FS-32, ESTABLISH Manual Auxiliary Spray.
d. INITIATE PZR spray operation.	
e. INITIATE SO23-12-11, Attachment 5, CORE EXIT SATURATION MARGIN CONTROL using the value of PTS Subcooling in place of Core Exit Saturation Margin.	

Comments / Reference: From SO23-12-11, Attachment 2, FS-7	Revision # 6

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

FS-7 VERIFY SI Throttle/Stop Criteria

Applicability: ALL

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. VERIFY at least one S/G operating: 1) SBCS – available OR ADV – available. AND 2) Feedwater – available.	a. GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.
b. VERIFY PZR level – greater than 30% AND – NOT lowering. c. VERIFY Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311. d. VERIFY Reactor Vessel level – greater than or equal to 100% (Plenum): QSPDS page 622 CFMS page 312 Attachment 4.	IF any criteria of steps b. through d. – NOT satisfied, THEN ▪ OPERATE Charging and SI systems as necessary to maintain Throttle/Stop criteria – satisfied. ▪ THROTTLE Loop Injection valves – as required. ▪ ENSURE auxiliaries to SI Pumps: a) Electrical power to pumps and valves. b) Proper system alignment. c) CCW flow. d) HVAC.

Comments / Reference: From SO23-14-5, Step 7 Bases	Revision # 8

EXCESS STEAM DEMAND EVENT BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.4.7 STEP 7 PREVENT Pressurized Thermal Shock (Continued)

At ~50% WR level in the affected S/G, actions are taken to begin steaming the generator in the unaffected loop. Opening the unaffected S/G ADV supplements the remaining affected S/G inventory to ensure Heat Removal is not lost and initiates the transition to the unaffected S/G. Transferring the ADV to AUTO/MODULATE to maintain the least affected S/G pressure approximately 200 PSIA above the most affected S/G pressure will result in heat transfer to the unaffected S/G with little or no temperature rise in the RCS. After the affected S/G reaches dryout, the unaffected S/G ADV is again adjusted for the lowest RCS T_{COLD} to address any additional cooldown associated with the remaining affected S/G inventory. The unaffected S/G should now be controlling RCS T_{COLD} with little or no RCS heatup. These strategies were developed via simulator experiences.

Step a.: Actions are initiated to establish stable RCS temperature anticipating dryout of the S/G with an unisolable steam line break or after isolation of the ESDE. RCS heat is primarily being removed by the steam line break, with little or none by the *least affected* S/G. Action must be taken to expeditiously transfer this heat removal to the *least affected* S/G that acts to prevent PTS. See SO23-14-11 for more detail.

Step b.: Initiates floating step, VERIFY SI Throttle/Stop Criteria. The ESDE results in a rapid drop in RCS temperatures. This effect could be aggravated by Safety Injection (SI) flow which adds cold water (approximately 70°F) from the RWST to the RCS; continued SI flow at full SI Pump discharge pressure could result in increased RCS pressures. Such a condition could result in exceeding design pressure of the RCS for the existing temperature. This could lead to brittle fracturing of RCS components, including the Reactor Vessel (which would result in unisolable loss of coolant flow paths). It is therefore desirable to throttle/stop SI as soon as the SI Throttle/Stop Criteria are met. See SO23-14-11 for more detail.

Comments / Reference: From SO23-12-10, Attachment SF-5	Revision # 3

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ATTACHMENT SF-5

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SAFETY FUNCTION STATUS CHECK

EXCESS STEAM DEMAND EVENT

1. VERIFY at least one Safety Function Acceptance Criteria for each Safety Function – satisfied at intervals of less than 15 minutes.

2. IF any Safety Function Criterion – NOT satisfied,

   THEN immediately inform SRO-in-charge.

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
1 Reactivity Control	
a. Reactor Power: 1) Lowering OR 2) Less than 10^{-4} % AND – stable or lowering. b. Maximum of one full length CEA – NOT fully inserted. OR Boration in progress – at greater than or equal to 40 GPM OR Shutdown Margin established – greater than 5.15% ΔK/K.	o RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT Excess Steam Demand Event, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Excess Steam Demand Event OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-1, RECOVERY – REACTIVITY CONTROL.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	E02 EA2.2	
	Importance Rating		4.0

Reactor Trip - Stabilization - Recovery: Ability to determine and interpret the following as they apply to the Reactor Trip Recovery: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.

Proposed Question: SRO 77

Given the following conditions:

- A Reactor trip has occurred on Unit 3.
- A transition to SO23-12-2, Reactor Trip Recovery has been completed.
- All four Reactor Coolant Pumps are running.

Subsequently:

- Pressurizer Pressure is 1870 psia and slowly lowering with all heaters energized.
- Pressurizer level is 30% and slowly lowering with all Charging Pumps operating.
- Containment sump level is rising.
- Steam Generator E088 Blowdown Radiation monitor shows a rising trend.
- Reactor Coolant System T_{COLD} is 520°F and stable.
- Reactor Coolant System T_{HOT} is 522°F and stable.

Which ONE (1) of the following identifies the action that should be taken in this situation?

A. Enter SO23-12-04, Steam Generator Tube Rupture and isolate Steam Generator E088.

B. Re-diagnose the event per SO23-12-10 Safety Function Status Checks, Attachment SF-1, Recovery Diagnostics and enter the identified EOI.

C. Enter SO23-12-03, Loss of Coolant Accident and initiate SIAS and CIAS.

D. Enter SO23-12-09, Functional Recovery and isolate Steam Generator E088.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because an indication of secondary activity is present, however, there is also leakage into Containment indicated and re-diagnosis is required.
B. Correct. The SFSCs of Reactor Trip Recovery for Pressurizer level and pressure and secondary activity are not met and re-diagnosis is specified.
C. Incorrect. Plausible because of RCS pressure and temperature and Containment sump level rise, however, there is also secondary activity indicated and re-diagnosis is required.
D. Incorrect. Plausible because there are indications of more than one event, however, confirmation via the re-diagnosis process is required.

Technical Reference(s) SO23-12-10, Steps 2, 4, and 7 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54780 — As the SRO, ANALYZE plant conditions to select the appropriate Emergency Operating Instruction and direct and coordinate the activities of shift personnel to recover from a Reactor Trip Event per SO23-12-2.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments / Reference: From SO23-12-10, Step 2	Revision # 3

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EMERGENCY OPERATING INSTRUCTION
REVISION 3
ATTACHMENT SF-2

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SAFETY FUNCTION STATUS CHECK

REACTOR TRIP RECOVERY

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
3 RCS Inventory Control	
a. PZR level: – between 10% and 70% AND – trending to between 30% and 60% b. Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311. c. Reactor Vessel level – greater than or equal to 48% (Head): QSPDS page 622 CFMS page 312 Attachment SF-10.	RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT uncomplicated Reactor Trip, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Uncomplicated Reactor Trip OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-3, RECOVERY – RCS INVENTORY CONTROL.

Comments / Reference: From SO23-12-10, Step 4	Revision # 3

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ATTACHMENT SF-2

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SAFETY FUNCTION STATUS CHECK

REACTOR TRIP RECOVERY

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
4 RCS Pressure Control	
a. PZR pressure (NR and WR) – between 1740 PSIA and 2380 PSIA AND – trending between 2025 PSIA and 2275 PSIA b. Core Exit Saturation Margin – between 20°F and 160°F: QSPDS page 611 CFMS page 311.	RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT uncomplicated Reactor Trip, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Uncomplicated Reactor Trip OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-4, RECOVERY – RCS PRESSURE CONTROL.

Comments / Reference: From SO23-12-10, Step 7	Revision # 3

NUCLEAR ORGANIZATION UNITS 2 AND 3	EMERGENCY OPERATING INSTRUCTION REVISION 3 ATTACHMENT SF-2	SO23-12-10 ISS 2 PAGE 11 OF 100

SAFETY FUNCTION STATUS CHECK

REACTOR TRIP RECOVERY

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
7 Containment Isolation	
a. Containment pressure – less than 1.5 PSIG. b. Containment Area Radiation Monitors – NOT alarming or trending to alarm R7845 Access Hatch R7848 General Area R7820-1 Containment (High) R7820-2 Containment (High). c. Secondary Radiation Monitors – NOT alarming or trending to alarm. R7870 Air Ejector, WRGM. R7818 Air Ejector R6759 E088 Blowdown R7874A/R7875A E088 Steamline R6753 E089 Blowdown R7874B/R7875B E089 Steamline.	◘ RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT uncomplicated Reactor Trip, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Uncomplicated Reactor Trip OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-6, RECOVERY – CONTAINMENT ISOLATION.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	015/17 G 2.1.20	
Importance Rating		4.6

RCP Malfunctions: Conduct of Operations: Ability to interpret and execute procedure steps

Proposed Question: SRO 78

Given the following conditions while in MODE 1:

- Annunciator 50A51 - VIBRATION AND LOOSE PARTS MONITORING SYSTEM TROUBLE is in alarm due to Reactor Coolant Pump P-004 vibration.
- Containment Sump inlet flow indicates 1.1 gpm.
- Charging and Letdown flow mismatch has risen by about 1 gpm.
- Reactor Coolant Pump P-004 Seal Cavity pressures are all normal.
- Reactor Coolant Pump P-004 Controlled Bleedoff flow is 1.5 gpm.
- All other seal parameters are normal.
- Reactor Coolant System Inventory Balance identified the leakage as 1.2 gpm.
- A Containment entry has identified leakage via a crack in the RCP casing.

Which ONE (1) of the following identifies the type of leakage and the action(s) required?

A. This is UNIDENTIFIED LEAKAGE greater than 1 gpm and must be corrected or the plant taken to COLD SHUTDOWN.

B. This is IDENTIFIED LEAKAGE less than 10 gpm and continued operation is allowed with no restrictions.

C. This is PRESSURE BOUNDARY LEAKAGE and the Unit must be placed in MODE 5.

D. This is leakage from a known source that has been specifically located and does not interfere with leakage detection systems. Establish controls to monitor for changes in the leakage rate.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that due to the location this would not be PRESSURE BOUNDARY LEAKAGE (PBL), however, pump casing leaks are unisolable from the RCS and meet the requirement for PBL.
B. Incorrect. Plausible because it could be thought that this was considered UNIDENTIFIED LEAKAGE based on being visually identified and not interfering with the leakage detection systems; however, this is PRESSURE BOUNDARY LEAKAGE which is not allowed in any quantity.
C. Correct. This is PRESSURE BOUNDARY LEAKAGE and the Unit must be placed in MODE 5.
D. Incorrect. Plausible because this particular criterion is used in the Tech Specs to differentiate IDENTIFIED and UNIDENTIFIED LEAKAGE, however, no PRESSURE BOUNDARY LEAKAGE is allowed.

Technical Reference(s): Technical Specification LCO 3.4.13; SO23-13-6, Step 3; SD-SO23-360, Figures I-13 & I-17A; Technical Specification Definitions

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 — Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 ____
55.43 2

Comments / Reference: From Technical Specification LCO 3.4.13	Amendment # 204

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: From SO23-13-6, Step 3	Revision # 5

NUCLEAR ORGANIZATION
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ABNORMAL OPERATING INSTRUCTION
REVISION 5

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REACTOR COOLANT PUMP SEAL FAILURE

OPERATOR ACTIONS

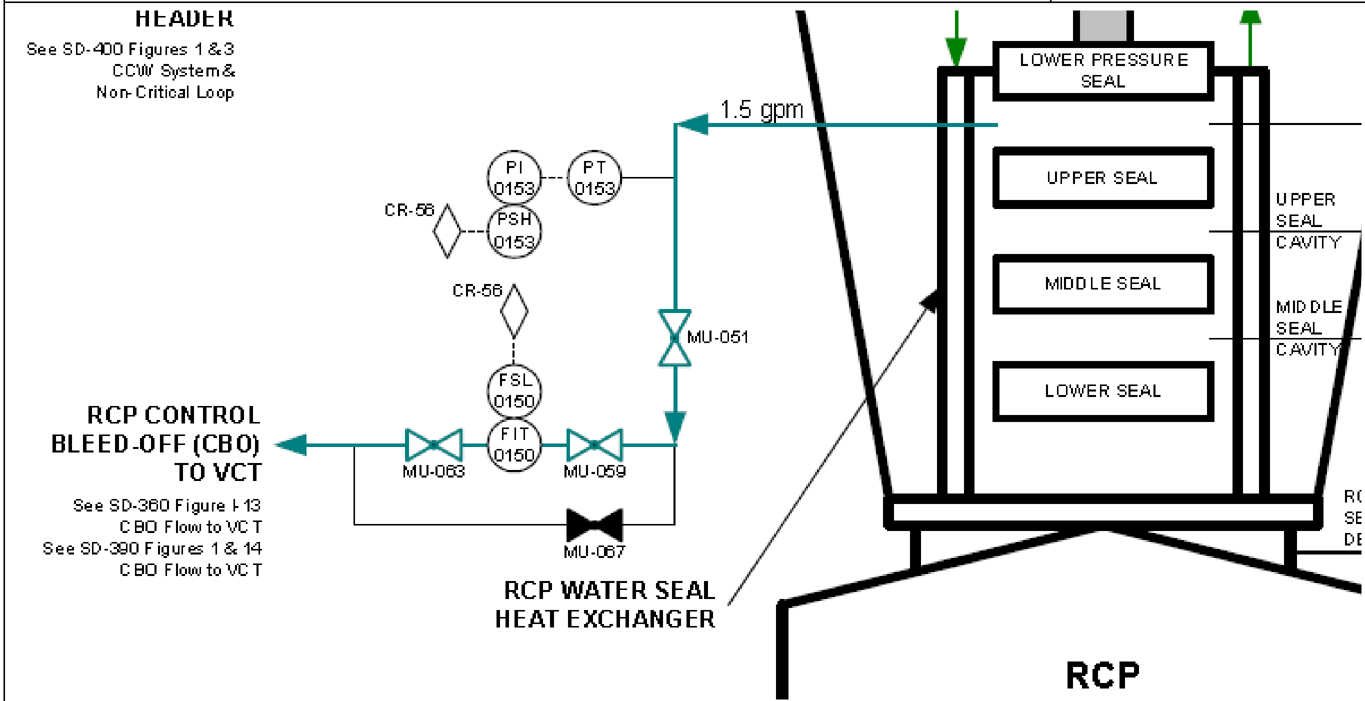
ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3 **Subsequent Diagnosis/actions:**	

NOTE

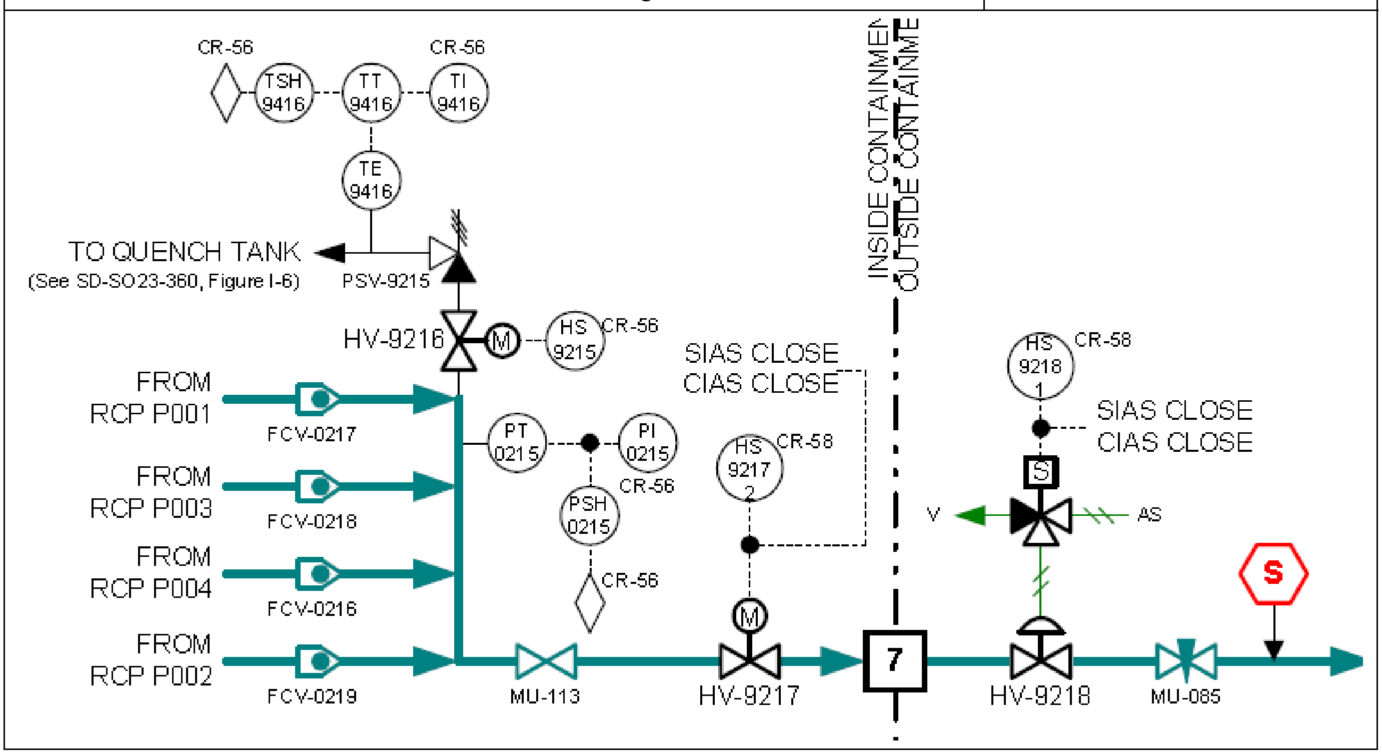
If there is no indicated CBO flow, and Vapor Seal Cavity Pressure is low, but other seal parameters are trending normally, then the vapor seal has failed and CBO flow is blowing into Containment. (Tech. Spec. LCO 3.4.13)

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐a. PERFORM SO23-3-3.37 to determine leakage into Containment.	
☐b. VERIFY CBO leakage into Containment - ≤ 10 gpm.	☐ b. TRIP the RX. ☐ 1) 5 seconds after CEA rod bottom lights are illuminated, TRIP the affected RCP(s). ☐ 2) GO TO SO23-12-1.
☐c. VERIFY CBO leakage into Containment - ≤ 4 gpm.	☐ c. INITIATE a controlled Plant Shutdown per SO23-5-1.7. 1) AFTER Reactor is tripped, **AND** CEAs have been inserted 5 seconds, ☐ THEN SECURE the affected RCP(s).
☐d. EVALUATE with Management the need to make a Containment entry to locate leak.	

Comments / Reference: From SD-SO23-360, Figure I-17A	Revision # 17



Comments / Reference: From SD-SO23-360, Figure I-13	Revision # 17



Comments / Reference: From Technical Specification Definitions	Amendment # 204

1.1 Definitions

ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME (Continued)

measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

All LEAKAGE that is not identified LEAKAGE.

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	009 G 2.2.37	
	Importance Rating		4.6

Small Break LOCA: Equipment Control: Ability to determine operability and/or availability of safety related equipment

Proposed Question: SRO 79

Given the following conditions:

- A Small Break Loss of Coolant Accident is in progress on Unit 2.
- A total loss of Component Cooling Water has occurred on Unit 2.
- The Recirculation Actuation Signal will actuate in 5 minutes.
- Bus 2A04 has tripped and locked out due to a bus fault.
- High Pressure Safety Injection Pump P-018 is running on Train B.
- Actions of SO23-12-3, Loss of Coolant Accident are in progress.
- Steam Generator levels are being maintained by Auxiliary Feedwater.

Which ONE (1) of the following actions must be taken to mitigate the situation?

A. Remain in SO23-12-3, Loss of Coolant Accident and cross-connect Unit 2 Component Cooling Water with Unit 3 per SO23-13-7, Loss of Component Cooling Water and invoke 10CFR50.54.X.

B. Start Train B P-019, HPSI Pump following the Recirculation Actuation Signal in order to provide additional flow and transition to SO23-12-9, Functional Recovery, Attachment FR-5, Recovery - Heat Removal if HPSI Pump performance becomes unstable.

C. Transition to SO23-12-9, Functional Recovery, Attachment FR-5, Recovery - Heat Removal and perform SO23-12-11, EOI Supporting Attachments, Attachment 23, Cross-Connecting Class 1E 480V Buses Between Units.

D. Perform actions of SO23-12-11, EOI Supporting Attachments, Attachment 14, RAS Operations to raise RWST level in order to flood Containment above the 23’ (foot) level to improve Net Positive Suction Head to the operating HPSI Pump.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, remaining in SO23-12-3 is appropriate and cross connecting CCW places the Unit in a 50.54.X notification.

B. Incorrect. Plausible because the pump is available (P-018, the 3rd of a kind Pump will start on an SIAS, therefore, P-019 is available) and might be considered, however, according to Step 5 EOI Bases, it will only increase flow marginally, if at all, and one operating train is sufficient at his time. Transitioning to the FR is plausible as this is the RNO action in Floating Step 22 if HPSI Pump flow becomes unstable.

C. Incorrect. Plausible because it could be thought that the FR is the procedure required for this condition, however, one Train is all that is required given the conditions listed.

D. Incorrect. Plausible because refilling the RWST would be a desired action given the loss of CCW and raising level does improve HPSI Pump NPSH, however, flooding above the 22’5” level will impact the Emergency Cooling Unit ductwork and could complicate the loss of CCW that already exists.

Technical Reference(s)	SO23-12-3, Step 5	Attached w/ Revision # See Comments / Reference
	SO23-14-3, Step 5 Bases	
	SO23-12-11, FS-22	
	SO23-13-7, Attachment 4	
	SO23-12-11, Attachment 14	

Proposed references to be provided during examination: None

Learning Objective: 52757 / 53682

STATE the major recovery actions in response to a LOCA event.

EVAULATE Component Cooling Water System conditions against Administrative and Technical Specification requirements and determine what action, if any, is required.

Question Source:	Bank #	129077	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam SONGS 2008

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

10 CFR Part 55 Content:	55.41	
	55.43	2, 5

Comments / Reference: From SO23-12-3, Step 5	Revision # 20

NUCLEAR ORGANIZATION
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EMERGENCY OPERATING INSTRUCTION
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LOSS OF COOLANT ACCIDENT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5 ESTABLISH Optimum SI Alignment:	
a. ESTABLISH two train operation:	a. REQUEST Shift Manager/Operations Leader to direct plant resources to establish the following support systems for non-operating/ unavailable equipment:
1) All available Charging Pumps – operating.	1) Electrical power to pumps and valves.
2) One HPSI and one LPSI per train – operating.	2) Proper system alignment.
3) All Cold Leg flow paths – aligned.	3) CCW flow.
4) VERIFY SI flow required:	4) HVAC.
SI flow – indicated	
OR	
RCS pressure – greater than 1250 PSIA.	
OR	
b. VERIFY FS-7, VERIFY SI Throttle/Stop Criteria – satisfied.	

Comments / Reference: From SO23-14-3, Step 5 Bases	Revision # 8

NUCLEAR ORGANIZATION
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EOI SUPPORT DOCUMENT SO23-14-3
REVISION 8 PAGE 17 OF 55
ATTACHMENT 1

LOSS OF COOLANT ACCIDENT BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.4.5 STEP 5 ESTABLISH Optimum SI Alignment

Intent
The intent of this step is to ensure that SI flow is within the limits of the design basis SI flow. That is, SI flow should be in accordance with the SI delivery curves. Required SI flow for given RCS pressure can be found in *SO23-12-10, Safety Function Status Checks* and *SO23-12-11, EOI Supporting Attachments*. The SI system design provides two redundant trains of SI, but only one train operation is necessary to meet the intent of the this step. Therefore, it may be optimum for RCS inventory recovery purposes to have two SI trains in operation delivering flow in accordance with the two pump curve, but one train in operation (one SI pump running with flow in accordance with the one pump curve) is acceptable.

4.4.5 STEP 5 ESTABLISH Optimum SI Alignment (Continued)

The addition of a second HPSI pump (third-of-a-kind or standby) to one train of HPSI injection will increase flow only 35-40% of the single pump's run-out flow. The increase in flow due to the second pump, at the lower end of single pump operation, will be minimal or none at all. Starting the second HPSI pump on one train will not make up for the loss of the redundant train of HPSI[1]. In addition, two pumps operating in parallel will normally not split flow evenly due to differences in the installed impellers; the higher head pump will dominate. This is particularly important at lower flows where the higher head pump can effectively stop flow from the other pump.

The RNO establishes the necessary support systems for pumps not operating so they can be made available to provide additional water to satisfy RCS inventory control sooner.

1 When you add two pump curves to form a generic curve for the two in parallel, you always start the curve with the highest head pump and follow that curve out until its head drops to the shutoff head of the other pump; from that point on they become additive. Gary Johnson, Email dated 7/10/98, Subject: LOCA EOI Bases for HPSI use.

Comments / Reference: From SO23-13-7, Attachment 4	Revision # 13

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REVISION 13
ATTACHMENT 4

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SUPPLYING UNIT 2 CCW SYSTEM FROM UNIT 3 TRAIN A CCW SYSTEM

CONTINUOUS USE

OBJECTIVE

Supply the Unit 2 CCW System from the Unit 3 CCW System by cross connecting through the Radwaste CCW Supply and Return Headers when Unit 2 is in Mode 5 or 6. This attachment would only be used when Unit 2 has lost all Saltwater Cooling or Component Cooling Water. **This attachment invokes 10CFR50.54.X on Unit 2 only.**

GUIDELINES

1. During the use of this Attachment, Unit 3 remains within its design and licensing basis analysis. The CCW Train being used to supply Unit 2 remains Operable. Therefore, it is not necessary to invoke 10CFR50.54.X on Unit 3. (AR 001001740)
2. In order to provide miniflow protection for the CCW Pump in the event that the Noncritical Loop is lost, CCW flow to the Unit 3 Emergency Cooling Units on the Train supplying Unit 2 should be maintained.
3. Train A, Train B, and 55 Lock keys are required.
4. CCW to the Emergency Chiller for the Train that will supply Unit 2 is isolated by this Attachment. This places Unit 3 in a 14 day action statement.

Comments / Reference: From SO23-12-11, FS-22	Revision # 6

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
FS-22 MONITOR ECCS Pump(s) Suction After RAS **Applicability: ☐ 12-3, ☐ 12-9**	
a. VERIFY RAS – actuated.	a. GO TO next applicable floating step.
b. VERIFY ECCS Pump(s) suction performance: 1) Operating ECCS Pump(s) flow – stable. 2) Operating ECCS Pump(s) discharge pressure – stable. 3) Operating ECCS Pump(s) motor amps – stable.	b. 1) OVERRIDE and STOP both Containment Spray Pumps. 2) OVERRIDE and CLOSE both Containment Spray Pump Discharge Valves. 3) IF HPSI Pump performance – unstable, THEN a) In Cold Leg Injection mode – THROTTLE SI flow to **minimum** requirements of Attachment 12, MINIMUM REQUIRED SI FLOWRATES DURING COLD LEG INJECTION. OR b) In Hot/Cold Leg Injection mode – THROTTLE HPSI flow to **minimum** requirements of Attachment 13, MINIMUM REQUIRED HPSI FLOWRATES DURING HOT / COLD LEG INJECTION 4) IF HPSI Pump performance – unstable, THEN STOP HPSI Pump. 5) NOTIFY SRO-in-charge. 6) GO TO SO23-12-9, FUNCTIONAL RECOVERY AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.

Comments / Reference: From SO23-12-11, Attachment 14	Revision # 6

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EOI SUPPORTING ATTACHMENTS

RAS OPERATION

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Filling the Containment sump improves the NPSH of the remaining HPSI Pump and filling to a level below 22'5" prevents flooding the ECU ductwork.

4 EVALUATE RWST Isolation:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. VERIFY Containment Sump level – greater than 22 feet.	a. 1) REQUEST Shift Manager/Operations Leader evaluate makeup to RWST using the source of water selected in FS-20, MONITOR RWST Level, step g. to maintain RWST level: – between 20% and 40%. 2) INITIATE makeup to the RWST as directed by the Shift Manager/Operations Leader. 3) WHEN Containment Sump level – greater than 22 feet, THEN CLOSE RWST Outlet Isolation valves: HV-9300 HV-9301. 4) GO TO step 5.
b. CLOSE RWST Outlet Isolation valves: HV-9300 HV-9301.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	025 AA2.07	
	Importance Rating		3.7

Loss of RHR System: Ability to determine the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation

Proposed Question: SRO 80

Given the following conditions:

- The Reactor Coolant System is in a Midloop Condition with RCS level lowering.
- Train A Shutdown Cooling Pump is oscillating ± 12 amps.

Which ONE (1) of the following identifies the desired action(s) for the conditions listed?

A. Vent the Train B Shutdown Cooling Heat Exchanger and start the Train B Shutdown Cooling Pump per SO23-3-2.6, Shutdown Cooling System Operation.

B. Vent the Train A Shutdown Cooling Heat Exchanger and throttle flow through Train B Shutdown Cooling Pump per SO23-3-2.6, Shutdown Cooling System Operation.

C. Commence emergency refill of the RCS by starting any HPSI Pump and open two Cold Leg Injection Valves per SO23-13-15, Loss of Shutdown Cooling.

D. Start any available Containment Spray Pump aligned as a Shutdown Cooling Pump per SO23-13-15, Loss of Shutdown Cooling.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Train A Shutdown Cooling Pump is cavitating and this action would be desired if the RCS were not in a Reduced Inventory Condition, however, venting the Train B SDC Heat Exchanger with the Train A SDC Pump running could worsen the condition.

B. Incorrect. Plausible because vortexing probably is occurring which is causing the Shutdown Cooling Pump to cavitate, however, in this condition emergency refill of the RCS is required.

C. Correct. Shutdown Cooling Pump cavitation is identified by an amperage fluctuation of ± 10 amps. With the Reactor Coolant System in a Reduced Inventory Condition emergency refill of the RCS must be performed and the desired flowpath is via a HPSI Pump and Hot or Cold Leg Injection Valves.

D. Incorrect. Plausible because the RNO action for emergency refill of the RCS requests that a Containment Spray Pump be started, however, the CS Pump must be aligned to the RWST in order to refill the RCS.

Technical Reference(s) SO23-13-15, Steps 4h & 4j
SO23-3-2.6, L&S 7.4 and 7.6
SO23-13-15, Attachment 10

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53933 / 55323
As the SRO, DIRECT operator response to a loss of shutdown cooling per SO23-13-15.
Per the Loss of Shutdown Cooling procedure, SO23-13-15, DESCRIBE: The basis for each step, caution, or note.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments / Reference: From SO23-13-15, Step 4h	Revision # 18

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LOSS OF SHUTDOWN COOLING

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 **RECOVER RCS Inventory:** (Continued)	
☐ d. VERIFY RCS level - less than 21 inches in Hot Leg OR - lowering. [5] [6]	d. ☐ GO TO Step 5.
☐ e. DETERMINE existing RCS perturbations from in-use Attachment for Reduced Inventory Condition RCS Perturbation Control. 1) STOP RCS perturbations.	
☐ f. INITIATE Attachment 11.	
☐ g. VERIFY plant status - Mode 5 OR - Mode 6 with Upper Refueling Cavity NOT filled	g. ☐ GO TO Attachment 5
☐ h. VERIFY running SDC Pump amperage - normal	h. ☐ IF running SDC Pump amperage fluctuating greater than ± 10 amps, THEN STOP the SDC Pump.

Comments / Reference: From SO23-13-15, Step 4j	Revision # 18

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LOSS OF SHUTDOWN COOLING

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 **RECOVER RCS Inventory:** (Continued)	
☐ i. ANNOUNCE "Commencing emergency refill of the RCS. All personnel stand clear of RCS openings."	
j. START operable or available HPSI Pump.	j. ☐ START the AVAILABLE CS Pump aligned from the RWST [9]
☐ 1) ENSURE selected flowpath does not bypass the RX Core through known leaks.	AND
2) ☐ ESTABLISH flow by throttling OPEN two Cold Leg Injection Valves,	☐ THROTTLE OPEN Discharge Valve to establish flow - less than 1000 gpm (FI-0338/FI-0348). • P-012: S2(3)1206MU012 • P-013: S2(3)1206MU014
OR	OR
☐ If location of inventory loss is unknown, or is identified as a Cold Leg Breach, then DIRECT FLOW thru one Hot Leg injection valve. [7][8]	☐ If CS Pump(s) not available, then START all available Charging Pump(s) from a borated source. (Tech. Spec. LCO B 3.1.10)
☐ k. If inventory available, Then CONSIDER dumping all available SITs to RCS.	

Comments / Reference: From SO23-3-2.6, L&S 7.4 and 7.6	Revision # 26

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7.0 RCS Reduced Inventory Condition

7.1 LIMIT: While the RCS is in a Reduced Inventory Condition, the maximum allowed RCS temperature will not exceed 148°F. (Ref. 2.2.6)

7.1.1 RCS temperature should be maintained at less than 140°F (preferably < 120°F) to maximize the Time-to-Boil margin.

7.2 To prevent vortexing and air entrapment in the SDCS piping, RCS level should be maintained ≥ 17 inches on 2(3)LI-1520N, RWLI and/or DLMS.

7.3 While the RCS is in Midloop Condition, excessive venting of SDCS may cause loss of SDC Pump suction due to vortexing in the RCS Hot Leg.

7.4 If SDC Pump cavitation develops (current swings of ±10 amps), then SDCS venting and any other activities affecting RCS level should be stopped, and recovery actions implemented per SO23-13-15, Loss of Shutdown Cooling.

7.5 If SDC flow is lost while in a Midloop Condition, then the only valid Reactor Core temperature indications are the operable CETs and HJTCs.

7.6 During SDC operation in a RIC, or when the RCS is depressurized, small amounts of gas trapped between HPSI cold leg flow indicating orifices and the upstream isolation valves can give rise to oscillating indications of HPSI flow. These indications are expected on the HPSI header associated with the SDC flow path and are characterized by:

- Oscillation period of a few seconds or less
- Oscillation magnitude of 100 gpm or less

When these conditions are present, the flow oscillations cannot be corrected by I&C and venting of the affected section of piping is generally not possible. No action besides monitoring is required.

Comments / Reference: From SO23-13-15, Attachment 10	Revision # 18

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RCS LEVEL CORRELATION CHART

REFERENCE	RWLI/DLMS WR	RWST LEVEL	PZR LEVEL	HJTC [1]	ABOVE FUEL	PLANT ELEV.
Refueling Level	+23.50'	95.1%	39.5%		434.5"	61'0"
PZR On Scale	+12.27'	59.5%	0%		297.5"	49'7"
RV Head Upper Pen. Overflow	+7.00'	43.6%		.	236.5"	44'6"
RV Head Middle	+3.375'	32.3%		# 1	193"	40'10.5"
Ref. Information	+1.833'	27.5%	RWLI-WR SIGHT GLASS	.	174.5"	39'.4"
Above Flange	+0.790'	24.3%		# 2	162"	38'3.5"
Vessel Flange	0.000'	21.8%	0.000'	.	152.5"	37'6"
Below Flange	-0.500'	20.2%	-0.500'	.	146.5"	37'0"
RV HJTC	-1.790'	16.2%	-1.790'	# 3	131.0"	35'8.5"
RCP Vent Overflow	-2.000'	15.6%	-2.000'	.	128.5"	35'6"
Ref. Information	-2.167'	15.1%	-2.167'	.	126.5"	35'4"
RV HJTC	-2.875'	12.9%	-2.875'	# 4	118.0"	34'7.5"
RIC and RCP Seal Overflow	-3.000'	12.5%	-3.000'	.	116.5"	34'6"
Above Hot Leg	-3.767'	10.1%	-3.767'	.	107.3"	33'8.8"
S/G Manway Ovfl.	-3.850'	9.8%	-3.850'	.	106.3"	33'7.8"

REFERENCE	RWLI/DLMS WR	RWLI/DLMS NR	WR SIGHT GLASS	HJTC [1]	ABOVE FUEL	PLANT ELEV.
MIDLOOP COND Top of Hot Leg	-4.875'	42" SIGHT GLASS	-4.875'	# 5	94"	32'7.5"
RCP Casing Flange	-4.958'	41"	-4.958'	.	93"	32'6.5"
Ref. Information	-5.292'	37"	-5.292'	.	89"	32'2.5"
NORMAL LEVEL	**-5.375'**	**36"**	**-5.375'**	.	**88"**	**32'1.5"**
Ref. Information	-5.542'	34"	-5.542'	.	86"	31'11.5"

MIDLOOP CONDITION CONTINUED ON NEXT PAGE

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	011 G 2.2.42	
Importance Rating		4.6

Large Break LOCA: Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications

Proposed Question: SRO 81

Given the following conditions on Unit 2 three (3) hours after a Large Break LOCA:

- Chemistry reports the following sampling results just prior to a Large Break Loss of Coolant Accident:
  - Safety Injection Tank T-007 boron concentration 2360 ppm.
  - Safety Injection Tank T-008 boron concentration 2540 ppm.
  - Safety Injection Tank T-009 boron concentration 2740 ppm.
  - Safety Injection Tank T-010 boron concentration 2380 ppm.
  - Refueling Water Storage Tank boron concentration 2875 ppm.
  - Reactor Coolant System pH is 7.2.

Based on the sample results, which ONE (1) of the following describes the impact of the Chemistry results and actions post-accident to remedy this condition?

A. Safety Injection Tank T-009 boron concentration is out of specification high. Containment Sump pH will result in accelerated corrosion of metals and production of hydrogen gas. Make preparations to purge hydrogen per SO23-1-4.2, Containment Purge and Recirculation Filtration System.

B. Combined Sump water boron concentration will be out of specification low causing an increased chance of return to criticality.
Monitor the Reactivity Critical Safety Function and initiate SO23-12-9, Functional Recovery, FR-1, Recovery - Reactivity Control actions.

C. Safety Injection Tanks T-007 and T-010 boron concentrations are out of specification low causing an increased chance of return to criticality.
Monitor the Reactivity Critical Safety Function and initiate SO23-12-9, Functional Recovery, FR-1, Recovery - Reactivity Control actions.

D. Refueling Water Storage Tank boron concentration is out of specification high. The amount of boron in the sump post-LOCA will cause early precipitation of boron on the core surfaces. Consider performing SO23-12-11, EOI Supporting Attachments, Attachment 11, Simultaneous Hot / Cold Leg Injection.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that this SIT boron was too high. High boron concentration will affect the sump pH. This is the required procedure if a Containment Purge is necessary.
B. Incorrect. Plausible because it could be thought that this SIT boron was too low if confused with the low limit for RWST boron of 2350 ppm. If boron was low the effects would be correct and the actions appropriate.
C. Incorrect. Plausible because it could be thought that these SITs boron were too low if confused with the low limit for RWST boron of 2350 ppm. If boron was low the effects would be correct and the actions appropriate.
D. Correct. The boron limit is 2800 ppm and could result in boron precipitation on the core earlier than expected on a large break LOCA. Early initiation of Simultaneous Hot / Cold Leg Injection would be a consideration.

Technical Reference(s): Technical Specification LCO 3.5.4 Bases; Technical Specification SR 3.5.4.3; Technical Specification SR 3.5.1.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55280 — EXPLAIN the different methods of recovery that are unique to the Functional Recovery Emergency Operating Instruction (EOI).

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 2, 5

Comments / Reference: From Technical Specification LCO 3.5.4 Bases	Amendment # 127
This LCO ensures that: a. The RWST contains sufficient borated water to support the ECCS during the injection phase; b. Sufficient water volume exists in the CES to support continued operation of the ESF pumps at the time of transfer to the recirculation mode of cooling; and c. The reactor remains subcritical following a LOCA. Insufficient water inventory in the RWST could result in insufficient cooling capacity of the ECCS when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of shutdown margin (SDM) or excessive boric acid precipitation in the core following a LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside containment.	
Comments / Reference: From Technical Specification LCO 3.5.4 Bases	**Amendment # 127**
The 2350 ppm limit for minimum boron concentration was established to ensure that, following a LOCA with a minimum level in the RWST, the reactor will remain subcritical in the cold condition following mixing of the RWST and RCS water volumes. Small break LOCAs assume that all control rods are inserted, except for the control element assembly (CEA) of highest worth, which is withdrawn from the core. Large break LOCAs assume that all CEAs remain withdrawn from the core. The most limiting case occurs at beginning of core life. The maximum boron limit of 2800 ppm in the RWST is based on boron precipitation in the core following a LOCA. With the reactor vessel at saturated conditions, the core dissipates heat by pool nucleate boiling. Because of this boiling phenomenon in the core, the boric acid concentration will increase in this region. If allowed to proceed in this manner, a point will be reached where boron precipitation will occur in the core. Post LOCA emergency procedures direct the operator to establish simultaneous hot and cold leg injection to prevent this condition by establishing a forced flow path through the core regardless of break location. These procedures are based on the minimum time in which precipitation could occur, assuming that maximum boron concentrations exist in the borated water sources used for injection following a LOCA. Boron concentrations in the RWST in excess of the limit could result in precipitation earlier than assumed in the analysis.	

Comments / Reference: From Technical Specification SR 3.5.4.3	Amendment # 127

SR 3.5.4.1	-------------------NOTE------------------- Only required to be performed when ambient air temperature is < 40°F or > 100°F. -- Verify RWST borated water temperature is ≥ 40°F and ≤ 100°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 362,800 gallons above the ECCS suction connection.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2350 ppm and ≤ 2800 ppm.	7 days

Comments / Reference: From Technical Specification SR 3.5.1.4	Amendment # 135

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each SIT isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each SIT is ≥ 1680 cubic feet and ≤ 1807 cubic feet.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each SIT is ≥ 615 psia and ≤ 655 psia.	12 hours
SR 3.5.1.4	Verify boron concentration in each SIT is ≥ 2200 ppm and ≤ 2800 ppm.	31 days <u>AND</u> ---NOTE---- Only required to be performed for affected SIT ---------- Once within 6 hours after each solution volume increase of ≥ 1% of tank volume that is not the result of addition from the refueling water storage tank

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	032 AA2.07	
	Importance Rating		3.4

Loss of Source Range NI: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Maximum allowable channel disagreement

Proposed Question: SRO 82

Given the following conditions:

- The Unit is in MODE 2 with a Reactor startup in progress.
- JI-001-2, Log Power Safety Channel has failed low and was placed in BYPASS.
- The remaining three Log Power Safety Channels indicate the following:
  - JI-001-1 reads 5 x 10^{-4}%.
  - JI-001-3 reads 4 x 10^{-5}%.
  - JI-001-4 reads 7 x 10^{-4}%.
- JI-001-4, Log Power Safety Channel has failed low and is INOPERABLE.

Which ONE (1) of the following identifies the required actions?

Place JI-001-4, Log Power Safety Channel in...

A. TRIP within one (1) hour. The Reactor Startup CANNOT continue until the channel is repaired.

B. BYPASS within one (1) hour. The Reactor Startup CANNOT continue until the cause of the greater than one-half (½) decade channel deviation between the remaining two channels is corrected.

C. TRIP or BYPASS within one (1) hour. The Reactor Startup CAN continue as long as the deviation between the two OPERABLE channels is less than one-half (½) decade when criticality is achieved.

D. TRIP or BYPASS within one (1) hour. The Reactor Startup CAN continue and criticality achieved with the remaining two channels deviation as is.

Proposed Answer: A

Explanation:

A. Correct. Because one channel is already failed and in BYPASS the second channel must be placed in TRIP per Technical Specifications. There are insufficient Log Safety Channels available to continue the startup.

B. Incorrect. Plausible because the failed channel action must occur within one hour, however, the 2nd failed channel must be placed in TRIP. A channel deviation of greater than ½ decade is allowed until the Reactor is critical; however, there are insufficient Log Safety Channels available to continue the startup.

C. Incorrect. Plausible because the guidance contained in SO23-3-3.2, Excore NI Calibration allows a one decade disagreement applies until criticality is achieved, however, the 2nd failed channel must be placed in TRIP.

D. Incorrect. Plausible if thought that only two channels were required during the startup and that the 2nd failed channel could be placed in TRIP or BYPASS.

Technical Reference(s): Technical Specification LCO 3.3.1.B; SO23-3-3.2, Attachment 4; SO23-3-1.1, Attachment 8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 — Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 ____
55.43 2, 5

Comments / Reference: From Technical Specification LCO 3.3.1	Amendment # 127

3.3.1 Reactor Protective System (RPS) Instrumentation — Operating

LCO 3.3.1 Four RPS trip and operating bypass removal channels for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

------------------------NOTES------------------------

1. Separate Condition entry is allowed for each RPS Function.
2. If a channel is placed in bypass, continued operation with the channel in the bypassed condition for the Completion Time specified by Required Action A.2 or C.2.2 shall be reviewed by the Onsite Review Committee.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one automatic RPS trip channel inoperable.	A.1 Place Channel in bypass or trip.	1 hour
	AND	
	A.2 Restore channel to OPERABLE status.	Prior to entering MODE 2 following next MODE 5 entry
B. One or more Functions with two automatic RPS trip channels inoperable.	B.1 -----NOTE------ LCO 3.0.4 is not applicable. ------------- Place one Functional Unit in bypass and the other in trip.	1 hour

Comments / Reference: From SO23-3-3.2, Attachment 4	Revision # 14

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2.0 PROCEDURE (Continued) PERF. BY INITIALS

[2.1.3.5] **If ≥ 5% Reactor Power**, then verify operable PPS Linear Power indicators agree within ± 7. (Value obtained by subtracting the lowest reading from the highest reading.)

[.6] **If < 5% Reactor Power**, then verify the following:

- All PPS Linear Power indications appear stable and reasonably close in value.
- Safety Channel green power lights illuminated.
- Test Panel white trouble lights extinguished.

SAT / UNSAT (Circle one) ________

2.1.4 Record Log Power Safety Channel readings. [1] ________

JI-0001-1 ____________ JI-0001A3 ____________

JI-0001-2 ____________ JI-0001A4 ____________

[.1] **Modes 1-4:** Verify operable Log Power Safety channel indicators agree within ½ decade of each other. (With the Reactor subcritical, the deviation between channels shall be within 1 decade due to electrical noise.)

[.2] **Mode 5:** Verify operable Log Power Safety channel indicators agree within 1 Decade of each other.

SAT / UNSAT (Circle one) ________

2.1.5 Record CPC Compensated LPD (Kw/ft) from CPC Point IDs 179. (Mark Section N/A if in Modes 3-5.) [1] ________

Channel A ____________ Channel C ____________

Channel B ____________ Channel D ____________

[.1] Verify all Compensated LPD indications are within 1.0 Kw/ft of each other.

SAT / UNSAT (Circle one) ________

Comments / Reference: From SO23-3-1.1, Attachment 8	Revision # 323

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ATTACHMENT 8

SO23-3-1.1
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REACTOR STARTUP SURVEILLANCE REQUIREMENTS

CONTINUOUS USE

OBJECTIVE

To ensure Technical Specification Surveillance Requirements are met in a timely fashion. Tech. Spec. LCO 3.1.6, LCO 3.4.2, LCS Figure 3.1.102-1, SR 3.3.1.7, SR 3.3.2.2, SR 3.3.4.4, and SR 3.4.2.1.

UNIT ________ DATE ______________________ TIME ________

PERF. BY INITIALS

1.0 PREREQUISITES

1.1 Verify this document is current by checking a controlled copy or by using the method described in SO123-VI-0.9. ________

2.0 PROCEDURE

2.1 Requirement - 3 of 4 Excore Logarithmic Power Channels shall be Operable: Modes 1-2, and Modes 3-5 with the RX Trip Breakers (RTCBs) closed and any CEA capable of being withdrawn. (Tech. Spec. SR 3.3.1.7 and SR 3.3.2.2)

2.1.1 Ensure 3 of 4 Excore Log Safety Channels have been tested and verified Operable within the required surveillance interval per Tech. Spec. SR 3.3.1.7 and SR 3.3.2.2.

Verified By: __________________ ____________ ________
I & C Foreman or GF Date Time

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	068 G 2.4.41	
	Importance Rating		4.6

Control Room Evacuation: Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications

Proposed Question: SRO 83

Given the following conditions from full power :

- At 1317 both Units 2 and 3 entered SO23-13-2, Shutdown from Outside the Control Room due to a fire with dense smoke inside the Control Room.
- At 1327 Unit 3 reported that Local Control was established.
- At 1329 the Fire Team Leader reported that the fire was out.
- At 1331 Unit 2 reported that Local Control was established.

Which ONE (1) of the following describes the required action associated with these conditions?

A. Notify the NRC within one hour that 10CFR50.54.X was invoked on both Units to establish Local Controls for Train A equipment.

B. Declare an UNUSUAL EVENT based on a fire threatening vital equipment lasting longer than 10 minutes.

C. Declare an ALERT based on Control Room evacuation and successfully establishing Local Control for both Units within 15 minutes.

D. Declare a SITE AREA EMERGENCY based on a fire that required Control Room evacuation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be thought that 50.54.X is invoked for disabling automatic features of safeguards equipment. 50.54.X would be invoked if Train B components were required to be used with Train A.
B. Incorrect. Plausible because the fire lasting longer than 15 minutes would result in an UNUSUAL EVENT.
C. Correct. Control Room evacuation and successfully establishing local control within 15 minutes is an ALERT.
D. Incorrect. Plausible if thought that two separate events required declaration of a SITE AREA EMERGENCY if it was determined that HOT SHUTDOWN capability was lost.

Technical Reference(s)	SO123-VIII-1, Attachment 2, Tab D2-4 SO123-VIII-1, Attachment 2, Tab D3-4 SO123-VIII-1, Attachment 2, Tab E1-1 SO23-13-2, Step 2 Caution (p198)	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: SO123-VIII-1, Attachment 2

Learning Objective: 56274 As the SRO, CLASSIFY emergency events requiring Emergency Plan implementation per SO123-VIII-1.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments / Reference: From SO123-VIII-1, Attachment 2, Tab D2-4	Revision # 27

NUCLEAR ORGANIZATION EPIP SO123-VIII-1
UNITS 1, 2 AND 3 REVISION 27 PAGE 35 OF 74

UNITS 2/3 AND SITE-WIDE EVENTS ATTACHMENT 2

LOSS OF SAFETY EQUIPMENT

ALERT

TAB D2

3. For Modes 1 - 4:

Unplanned loss of most or all Control Room annunciators with either:

(a) Alternate alarm indication from the Plant Monitoring System not available

OR

(b) Plant conditions associated with any of the systems listed in Attachment 5 are unstable and uncontrolled.

NOTE: For loss of annunciators with unstable plant conditions, and alternate alarm indication unavailable, see Event Code D3-3.

4. For Modes 1 - 6:

(a) The Control Room is evacuated

AND

(b) Control of shutdown systems is established locally or at the remote shutdown panel within 15 minutes.

NOTE: If control of shutdown systems is not established in 15 minutes, see Event Code D3-4.

Comments / Reference: From SO123-VIII-1, Attachment 2, Tab D3-4	Revision # 27

NUCLEAR ORGANIZATION EPIP SO123-VIII-1
UNITS 1, 2 AND 3 REVISION 27 PAGE 38 OF 74

UNITS 2/3 AND SITE-WIDE EVENTS ATTACHMENT 2

LOSS OF SAFETY EQUIPMENT
SITE AREA EMERGENCY TAB D3

4. For Modes 1 - 6:

(a) The Control Room is evacuated

AND

(b) Control of shutdown systems has not been established locally or at the remote shutdown panel within 15 minutes.

Comments / Reference: From SO123-VIII-1, Attachment 2, Tab E1-1	Revision # 27

NUCLEAR ORGANIZATION EPIP SO123-VIII-1
UNITS 1, 2 AND 3 REVISION 27 PAGE 42 OF 74

UNITS 2/3 AND SITE-WIDE EVENTS ATTACHMENT 2

DISASTER
UNUSUAL EVENT TAB E1

NOTE: Occurrence of any natural or manmade disaster should be immediately evaluated for impact on plant components or operation. Emergency classification of these occurrences should be made under applicable E1 event codes if the Emergency Coordinator judges the impact to be significant or to warrant emergency notification of offsite authorities, even though the explicit criteria of the event code may not be met.

1. For Modes 1 - 6: (Site-wide Event)

A fire which is not declared extinguished by the Fire Incident Commander within 15 minutes of Control Room notification or verification of a control room alarm at any of these locations:

- Inside the Protected Area and affecting or adjacent to areas and structures containing vital, safety related or safe-shutdown equipment
- Multipurpose Handling Facility (MPHF/T60)
- South Yard Facility (T10/T20)
- The Independent Spent Fuel Storage Installation (ISFSI)

NOTE: For Control Room evacuation, see Event Codes D2-4 and D3-4.

Comments / Reference: From SO23-13-2, Step 2 Caution	Revision # 11

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ABNORMAL OPERATING INSTRUCTION
REVISION 11
ATTACHMENT 24

SO23-13-2
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TRAIN B SYSTEMS RECOVERY

CONTINUOUS USE

UNIT ________

PERF. BY
INITIALS

1.0 Performance Guidelines:

1.1 Do not delay these actions for Security, or any other concerns unless a delay is necessary to maintain personnel safety.

1.2 Due to the seriousness of the emergency, prompt completion of these actions overrides all other Procedures, Documents, Work Plans, Technical Specifications, Technical Manuals, and/or Verbal Directions given by any person or group other than the Operations Shift Manager.

CAUTION

IF actions taken to recover Train A Safe Shutdown Systems per the analyzed methodology are not completely successful, THEN utilize any combination of Train A, B, C and D Systems as necessary to achieve and maintain Safe Shutdown conditions. If required, then such action would depart from license conditions. However, they are permitted in order to protect the public health and safety per 10 CFR 50.54(x). Consequently, NRC notification would be required within one hour per 10 CFR 50.72.

2.0 In the Train B 1E Switchgear Room:

2.1 Open Second Point of Control Cubicle A06-01. ________

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	067 AA2.03	
Importance Rating		3.5

Plant Fire on Site: Ability to determine and interpret the following as they apply to the Plant Fire on Site: Fire alarm

Proposed Question: SRO 84

Given the following conditions:

- Annunciator 61A15 - FIRE DETECTED has alarmed.
- The fire was in the Train C Y03 1E Inverter Room and has been extinguished.
- Efforts to maintain Y03 1E Inverter Room temperatures less than 90°F have not been successful.

Which ONE (1) of the following is required per SO23-1-5, Auxiliary Building Normal HVAC System Operation?

A. Open all 1E Inverter Room doors and initiate a Fire Impairment for the Technical Specification Fire Doors.

B. Align the Swing Battery Charger to the Y03 inverter and initiate a Fire Impairment for door blockage.

C. Align forced ventilation when Inverter Room temperatures exceed 120°F.

D. Open the Y03 1E Inverter Room door, align forced ventilation and initiate a Fire Impairment.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because these actions are correct, however, the only time all doors are opened is if temperatures exceed 100°F.
B. Incorrect. Plausible because a Fire Impairment would be issued anytime a door is blocked, however, aligning the Swing Battery Charger is not required.
C. Incorrect. Plausible because forced ventilation must be aligned, however, this action must be performed when room temperatures exceed 90°F.
D. Correct. With the affected room temperature greater than 90°F but less than 100°F, open the affected room door, align forced ventilation, and initiate a Fire Impairment.

Technical Reference(s) SO23-1-5, Attachment 9 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53413 — Per the Fire procedure, SO23-13-21, DESCRIBE: The basis for each step, caution, or note and the expected plant response for each step.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments / Reference: From SO23-1-5, Attachment 9	Revision # 20

NUCLEAR ORGANIZATION
UNITS 2 AND 3

OPERATING INSTRUCTION
REVISION 20
ATTACHMENT 9

SO23-1-5
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2.0 PROCEDURE (Continued) PERF. BY INITIALS

2.5.2 Actions to maintain 1E Inverter Room Temperature < 99°F (Alarm Setpoint): (LS-1.7)

.1 FULLY OPEN all the 1E Inverter Room Doors. ________

.2 Perform the following:

- Notify the Fire Department that the 1E Inverter Room Doors are propped Open. ________
- Initiate Fire Impairments for the affected Tech. Spec. Fire Doors. ________
- Log the name of the Fire Department person contacted. ________

.3 Within 3 hours, establish Forced Supply Ventilation, from a source of outside air to the hallway or into the 1E Inverter Rooms, sufficient to maintain temperatures below 99°F.

.4 If room temperature exceeds the room analytical limit of 104°F, then Initiate a Notification for an Operability Assessment.

2.5.3 Actions to maintain ESF Switchgear Room Temperature < 90°F(Alarm Setpoint): (LS-1.7)

.1 FULLY OPEN all the doors to each affected ESF SWGR Room.

.2 Perform the following:

- Notify the Fire Department that the ESF Switchgear Room Doors are propped Open. ________
- Initiate Fire Impairments for the affected Tech. Spec. Fire Doors. ________
- Log the name of the Fire Department person contacted. ________

.3 Within 3 hours, establish Forced Supply Ventilation, from a source of outside air to the hallway or into the ESF SWGR Rooms, sufficient to maintain temperatures below 90°F.

.4 If room temperature exceeds the room analytical limit of 95°F, then Initiate a Notification for an Operability Assessment.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	E09 G 2.2.42	
Importance Rating		4.6

Functional Recovery: Equipment Control: Ability to recognize system parameters that are entry level conditions for Technical Specifications

Proposed Question: SRO 85

Given the following conditions on Unit 2:

- A Station Blackout occurred two (2) hours ago.
- Efforts to recover power or cross-tie to Unit 3 have been unsuccessful.
- Entry into SO23-12-9, Functional Recovery, Attachment FR-2, Recovery - Vital Auxiliaries was made to perform power restoration.
- Auxiliary Feedwater Pump P140 tripped on startup and the overspeed trip linkage is broken and will NOT reset.
- Both Steam Generators are empty and entry to SO23-12-9, Functional Recovery, Attachment FR-5, Recovery - Heat Removal has been made due to NO OPERABLE Steam Generator and Reactor Coolant System temperatures RISING.
- Subcooling has been lost and Core Exit Thermocouples are approaching 700°F.

Which ONE (1) of the following describes the actions to take to establish heat removal?

Align Fire Water to the discharge of...

A. P504, Auxiliary Feedwater Pump and feed Steam Generator E089 using Fire Water at low flow for at least 30 minutes and then recover levels to at least 40% narrow range.

B. P504, Auxiliary Feedwater Pump and feed both Steam Generators using Fire Water. Invoke 10CFR50.54.X due to using Firewater to feed the Steam Generators and notify the NRC within 8 hours.

C. P141, Auxiliary Feedwater Pump and feed Steam Generator E088 using Fire Water. Enter the Severe Accident Management Guidelines (SAMG) due to Steam Generator dryout.

D. P141, Auxiliary Feedwater Pump and feed both Steam Generators by cross-tying Auxiliary Feedwater trains using Fire Water. Invoke 10CFR50.54.X due to isolating Auxiliary Feedwater Pumps and notify the NRC within one hour.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the connection was on P504. The Steam Generator recovery level listed is consistent with guidance in the Functional Recovery Procedure; however, the time is 15 minutes.

B. Incorrect. Plausible because invoking 50.54.X is required due to isolating the Auxiliary Feedwater Pumps, however, not for the reasons stated. Additionally, the Fire System connection is on P141 and NRC notification would be required within one hour.

C. Incorrect. Plausible because feeding both Steam generators is desired and the correct AFW Pump is being used, however, entry into the SAMG is not required.

D. Correct. With the Heat Removal Safety Function NOT met and all other design paths unavailable, the Functional Recovery Procedure directs aligning the Fire System which has a Diesel Driven Pump and feeding both Steam Generators. 10CFR50.54.X (Step 8a, Caution) is declared due to isolating the AFW Pumps because of overpressure concerns with the fire water connection (Step 8c, Caution).

Technical Reference(s): SO23-12-9, Step 8; Technical Specification LCO 3.7.5 Bases; SD-SO23-780, Figure 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55217 — Per the Functional Recovery procedure SO23-12-9 DESCRIBE: The basis for each step, caution or note.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
55.41 ____
55.43 1, 2, 5

Comments / Reference: From SO23-12-9, Step 8	Revision # 25

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 25
ATTACHMENT FR-5

SO23-12-9 ISS 2
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FUNCTIONAL RECOVERY

RECOVERY – HEAT REMOVAL

Success Path Actions: HR-1, S/G with no ECCS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

8 ESTABLISH Fire System Flow to available S/Gs:

NOTE

Isolation of P-141, AFW Pump, is required to establish boundary isolation for flange connection. Connecting the flange, located in locked box (55 key) inside of the AFW Pump room, to the discharge of P-141 AFW Pump will render the AFW system INOPERABLE. The provisions of 10 CFR 50.54(x) should be utilized after normal design actions have proven unsuccessful, or Safety Functions are challenged by being in danger of becoming NOT satisfied.

a. REQUEST Shift Manager:

1) APPROVE use of 10 CFR 50.54(x) for use of Fire Pump connection to AFW piping.

2) INITIATE NRC notification within one hour regarding actions per this step.

b. VERIFY one of the following Fire Pumps – available:

P-220, Diesel Firewater Pump
P-221, Firewater Pump
P-222, Firewater Pump.

RESPONSE NOT OBTAINED:

b. ESTABLISH alternate pump and water supply to the firewater system from one of the following sources:

- HFMUD
- SA1417MT351, Demineralized Water Storage Tank
- Onsite/Offsite Fire Trucks

AND GO TO step d.2).

Comments / Reference: From SO23-12-9, Step 8	Revision # 25

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 25
ATTACHMENT FR-5

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FUNCTIONAL RECOVERY

RECOVERY – HEAT REMOVAL

Success Path Actions: HR-1, S/G with no ECCS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8 ESTABLISH Fire System Flow to available S/Gs: (Continued)	

CAUTION

Flange is not rated for use with an AFW Pump in service. Isolation or removal of flange is required prior to starting an AFW Pump.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
c. ENSURE P-141, AFW discharge flowpath – isolated: CLOSE 1305MU127 CLOSE 1305MU131 CLOSE 1305MU154 CLOSE 1305MU635.	
d. Cross-connect Fire Header with AFW System:	d.
1) CONNECT one end of fire hose to Fire System supply: <u>For Unit 2</u> SA2301MU254, Fire Hydrant 8N, (located near Unit 2 Containment Emergency Access Hatch on 30 ft.). <u>For Unit 3</u> SA2301MU262, Fire Hydrant 8S, (located between Unit 3 Isophase Bus Cooling Unit and Tank Building on 30 ft.).	1) Connect Fire hose to any available pressurized fire system connection.
2) Connect other end of fire hose to – 6" connection on P-141, AFW Pump discharge piping.	

Comments / Reference: From SO23-12-9, Step 8	Revision # 25

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION SO23-12-9 ISS 2
REVISION 25 PAGE 153 OF 274
ATTACHMENT FR-5

FUNCTIONAL RECOVERY

RECOVERY – HEAT REMOVAL

Success Path Actions: HR-1, S/G with no ECCS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8 ESTABLISH Fire System Flow to available S/Gs: (Continued)	
e. ENSURE AFW Pump discharge and Discharge Bypass valves – closed: P-504: HV-4712, HV-4762 P-141: HV-4713, HV-4763	
f. START available Fire Pump.	
g. ESTABLISH fire water hose overpressure protection: 1) CLOSE AFW Pump Manual Discharge valves (sys 1305): P-504: MU533; P-140: MU122; P-141: MU127	
h. OPEN Fire System Supply valve selected in step 8d.1): 1) For Unit 2 SA2301MU254, Fire Hydrant 8N. For Unit 3 SA2301MU262, Fire Hydrant 8S.	h. Open an Alternate Fire Header valve selected in step 8d.1) RNO.
i. ALIGN flowpath to both S/Gs by OPENING the following valves: 1305MU131 1305MU154 1305MU634 1305MU635.	

Comments / Reference: From Technical Specification LCO 3.7.5 Bases	Amendment # 127

AFW System
B 3.7.5

BASES (continued)

BACKGROUND (continued)

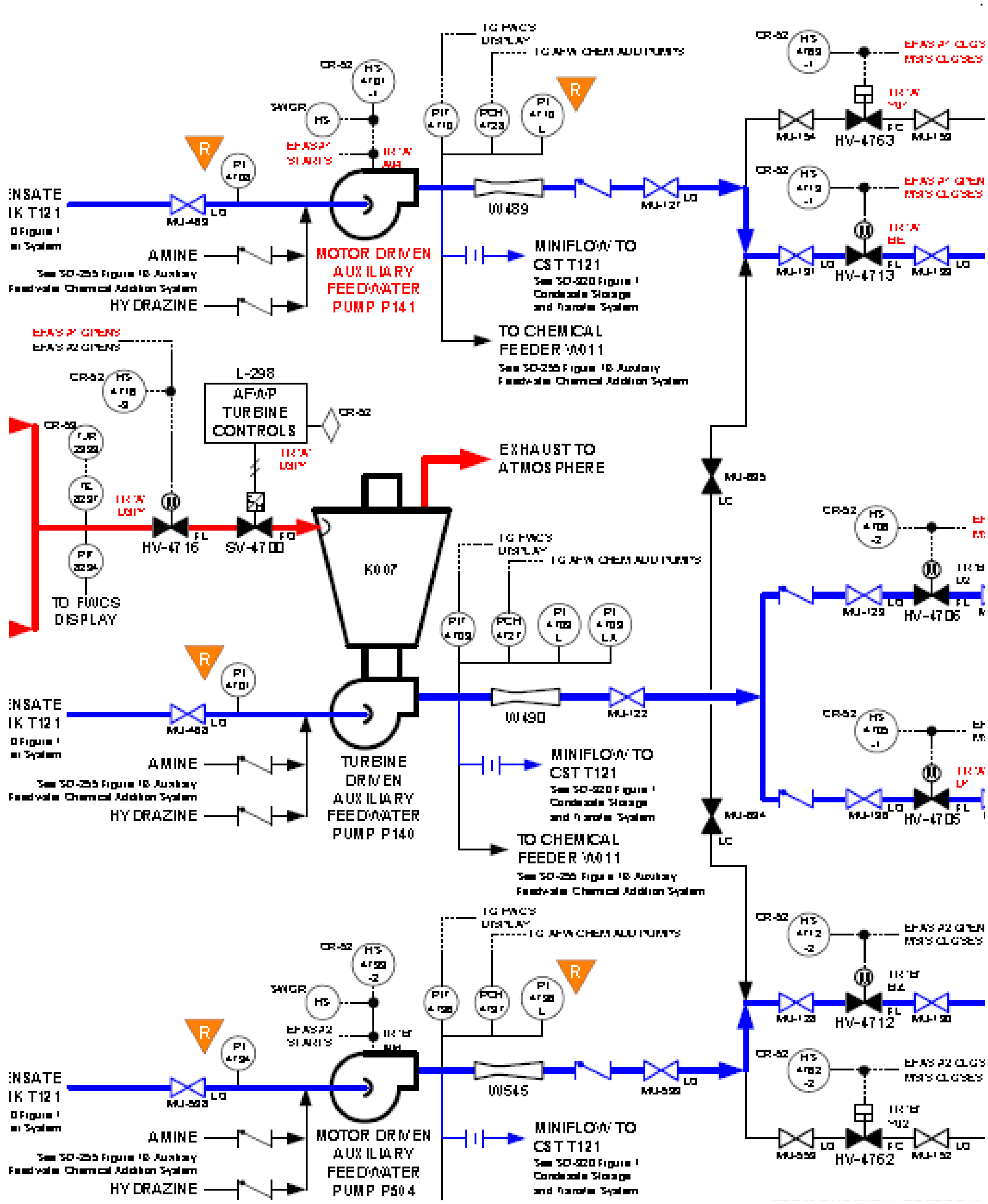
1. Motor-driven auxiliary feedwater pump discharge bypass control valves, HV-4762 and HV-4763, need only be capable of being closed, or be isolated by manual valves;

2. Steam turbine-driven auxiliary feedwater pump steam supply isolation valves, HV-8200 and HV-8201, and turbine stop valve, HV-4716, need only be capable of being opened, and

3. Manual crosstie valves 1305MU634 and 1305MU635 may be open in Mode 3 provided a minimum of 2 hours has elapsed since reactor shutdown.

The AFW System is discussed in the UFSAR, Section 10.4.9 (Ref. 1).

Comments / Reference: From SD-SO23-780, Figure 1	Revision # 10



Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	010 A2.01	
Importance Rating		3.6

Pressurizer Pressure Control System: Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures

Proposed Question: SRO 86

Given the following conditions with the Unit in MODE 1 at 80% power:

- A Steam Bypass Control Valve failed open 2 minutes ago and is now closed.
- Pressurizer pressure is 2020 psia and slowly increasing.
- PIC-100, Pressurizer Pressure Indicating Controller output is 0%.
- Annunciator 50A14 - PZR PRESS HI/LO is in alarm.
- Annunciator 50A04 - PZR PRESS DEVIATION HI/LO is NOT in alarm.
- All systems are in Automatic.

Which ONE (1) of the following:

1.) Identifies the impact of Pressurizer Pressure Control on Technical Specifications?

2.) What procedural action must be taken to address this situation?

A. 1.) Technical Specification LCO 3.4.1, DNB Pressure, Temperature, and Flow Limits must be entered and corrected.
2.) Refer to SO23-13-27, Pressurizer Pressure and Level Malfunctions, Step 3, Pressure Out of Band and ensure all Heaters are energized and both Pressurizer Spray Valves are closed.

Corrected font

B. 1.) Technical Specification LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits must be entered and Pressurizer Heaters restored.
2.) Refer to SO23-3-1.10, Pressurizer Pressure and Level, Section 6.1, Normal Pressurizer Pressure Control and control Pressurizer Heaters and Spray in manual.

C. 1.) Technical Specification LCO 3.4.3, RCS Pressure and Temperature (P/T) Limits must be entered and pressure restored.
2.) Trip the Reactor and enter SO23-12-1, Standard Post Trip Actions due to loss of Pressurizer Pressure control.

D. 1.) Technical Specification LCO 3.4.1, DNB Pressure, Temperature, and Flow Limits must be entered and corrected.
2.) Refer to SO23-3-1.10, Pressurizer Pressure and Level, Section 6.7, Manual Pressurizer Pressure Control and operate PIC-100, Pressurizer Pressure Controller in manual to control pressure.

Proposed Answer: A

Explanation:

A. Correct. Given that the reason for the pressure transient is known, entry into SO23-13-27 is appropriate given the response of the primary system. Swapping to the other Pressurizer Pressure Channel is not required because the deviation alarm is not annunciating. Technical Specification LCO 3.4.1 contains the correct information.
B. Incorrect. Plausible if thought that Pressurizer Heaters were out of service because pressure was responding slowly.
C. Incorrect. Plausible if thought that this Technical Specification LCO is not being met. Given the conditions listed, a Reactor trip is not warranted.
D. Incorrect. Plausible because the Technical Specification entry is correct, however, the wrong procedure and Section of SO23-3-1.10 is used.

Technical Reference(s):
- SO23-13-27, Step 3b RNO & 3a
- Technical Specification LCO 3.4.1
- SO23-3-1.10, Section 6.1 & 6.7
- Technical Specification LCO 3.4.3
- Technical Specification LCO 3.4.9
- SO23-15-50.A1, 50A04 & 50A14

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55213 / 56649

As the Control Room Supervisor and given plant conditions, DETERMINE appropriate steps to be taken for a Pressurizer Pressure or Level Malfunction in accordance with SO23-13-27.

Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 2, 5

Comments / Reference: From SO23-13-27, Step 3b RNO	Revision # 4

NUCLEAR ORGANIZATION UNITS 2 AND 3 | ABNORMAL OPERATING INSTRUCTION REVISION 4 | SO23-13-27 PAGE 11 OF 20

PRESSURIZER PRESSURE AND LEVEL MALFUNCTION

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3 Pressure out of band (Continued)	
☐ b. VERIFY Pressurizer Pressure is stable.	b. If Pressurizer pressure is trending **high**, then: ☐ 1) COMMENCE operating PIC-0100 in MANUAL per SO23-3-1.10, Section for Manual Pressurizer Pressure Control. ☐ 2) SECURE heaters, as necessary. If Pressurizer pressure is trending **low**, then: ☐ 1) START Pressurizer heaters, as necessary. ☐ 2) ENSURE both Pressurizer Spray Valves are closed. ☐ If unable to close affected Spray Valve in manual, then GO TO STEP 3e.

Comments / Reference: From 50A14SO23-13-27, Step 3a	Revision # 4

NUCLEAR ORGANIZATION
UNITS 2 AND 3

ABNORMAL OPERATING INSTRUCTION
REVISION 4

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PRESSURIZER PRESSURE AND LEVEL MALFUNCTION

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE | RESPONSE NOT OBTAINED

3 Pressure out of band

GUIDELINES

1) A Pressurizer Pressure signal failure affects the Modulate and Permissive circuits of SBCS in the following way:

- Channel X or Y high failure could delay the Master Controller response and bring in the permissives early
- Channel X or Y low failure will delay the response of both controllers

2) See Attachment 1 for the Pressurizer Pressure Control Block Diagram.

3) See Attachment 4 for Pressurizer Pressure Control Diagrams.

4) To diagnose controller alarms, refer to SO23-3-1.10, Attachment for Foxboro Alarm Response and Foxboro Controller Page Data.

5) Reactivity will be impacted by changes in Pressurizer Heater configuration and Pressurizer Spray control. The RCS Reactivity Pressure Coefficient is a positive coefficient and is about one tenth the absolute value of the Moderator Temperature Coefficient.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ a. VERIFY the selected Pressurizer Pressure channel is between 2225 and 2275 psia and stable.	☐ a. VERIFY the other pressure channel is available by observing PR-0100A or PR-0100B or CFMS page 325.
	☐ 1) POSITION HS-0100A, PZR Pressure Channel Select Switch, to the other channel.

Comments / Reference: From Technical Specification LCO 3.4.1	Amendment #149

3.4.1 RCS DNB (Pressure, Temperature, and Flow) Limits

LCO 3.4.1 RCS parameters for pressurizer pressure, cold leg temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ 2025 psia and ≤ 2275 psia;

b. RCS cold leg temperature (T_c):
   1. For THERMAL POWER less than or equal to 30% RTP, 522°F ≤ T_c ≤ 558°F,
   2. For THERMAL POWER greater than 30% RTP, 535°F ≤ T_c ≤ 558°F.

c. RCS total flow rate ≥ 396,000 gpm.

APPLICABILITY: MODE 1.

---------------------------NOTE---------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer pressure or RCS flow rate not within limits.	A.1 Restore parameter(s) to within limit.	2 hours

Comments / Reference: From SO23-3-1.10, Section 6.1	Revision # 21

NUCLEAR ORGANIZATION UNITS 2 AND 3 — OPERATING INSTRUCTION REVISION 21 — SO23-3-1.10 PAGE 5 OF 58

6.0 PROCEDURE

6.1 Normal Pressurizer Pressure Control

INFORMATION USE

6.1.1 The following operating conditions are the *preferred* targets for Pressurizer pressure parameters: These apply when both spray valves are in service and sprays are not being forced.

RCS Pressure (PIC-0100 Controller Pressure)	2250 +/- 5 psia
PIC-0100 controller output	20% to 60%
Spray Valves	Fully Closed
RCS pressure setpoint	≤ 2260 psia
Non-1E backup heaters (LS-1.3)	1 or 2 bank(s) energized

6.1.2 With the RCS at normal operating temperature, the normal PZR pressure control configuration is, as follows:

.1 PIC-0100, Pressurizer Pressure Controller, in AUTO with the setpoint fixed such that RCS pressure will be maintained at 2250 psia. (LS-1.5)

.2 HIC-0100 A & B, Spray Valve Controllers; both Controllers in AUTO. (LS-1.6)

.3 HS-0100A, PZR Pressure Channel Select Switch, selected to the channel closest to the highest Narrow Range pressure indicated at PCS PIDs CPC009A, B, C, D). The TS 3.4.1 limit of 2275 psi is based on Narrow Range pressure.

.4 HS-9170 and HS-9171, Proportional Heaters, are ON.

.5 When PZR and RCS boron concentration are within 10 ppm, then PZR Backup Heaters may be energized as required to maintain PZR Proportional Heater capacity at approximately 50%. (LS-1.5)

6.1.3 Pressurizer heater configuration may be changed per Section 6.4.

Comments / Reference: From SO23-3-1.10, Section 6.7	Revision # 21

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6.0 PROCEDURE (Continued)

6.7 Manual Pressurizer Pressure Control

CONTINUOUS USE

NOTES

1. Loss of Offsite Power does not preclude manual energization of Pressurizer 1E Backup Heaters even though tripping relay power is non-1E.
2. No interlock exists for prevention of manual energization of Pressurizer 1E Backup Heaters when RCS pressure is greater than 2340 psia.
3. Normal PZR Spray or Auxiliary Spray with ΔT > the limit specified in SO123-0-A4 will require a Design Cycle evaluation per Technical Specification Table 5.7-1 and completion of SO123-0-A4, Attachment for Pressurizer Spray Cycles-Units 2 and 3. (Ref. 2.1.4, 2.3.11)

6.7.1 If Pressurizer pressure deviates from program outside of the Operator's control, then GO TO SO23-13-27, Pressurizer Pressure and Level Malfunction.

6.7.2 **Manually control PIC-0100, Pressurizer Pressure Controller**

.1 ENSURE a Reactivity Brief has been conducted for this activity per SO123-0-A1, Section for Reactivity. ☐

.2 TRANSFER PIC-0100, Pressurizer Pressure Controller, to MANUAL. ☐

.3 ADJUST output as necessary to maintain setpoint. ☐

Comments / Reference: From Technical Specification LCO 3.4.3	Amendment #203

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 The combination of RCS pressure, RCS temperature and RCS heatup and cooldown rates shall be maintained within the limits as specified in the RCS PRESSURE-TEMPERATURE LIMITS REPORT (PTLR).

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours

Comments / Reference: From Technical Specification LCO 3.4.9	Amendment #161

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 57%; and

b. Two groups of pressurizer heaters OPERABLE with the capacity of each group ≥ 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit.	A.1 Be in MODE 3 with reactor trip breakers open. AND A.2 Be in MODE 4.	6 hours 12 hours
B. One required group of pressurizer heaters inoperable.	B.1 Restore required group of pressurizer heaters to OPERABLE status.	72 hours

Comments / Reference: From SO23-15-50.A1, 50A04	Revision # 8

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50A04 PZR PRESS DEVIATION HI/LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	WHITE	N/A	NONE

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT [1]	PCS ID	LINK # U2/U3
2(3)PSHL-0100 [2]	PZR Pressure Switch HI/LO	Opposite Channel ±5% Deviation from Selected Channel	2(3)PI-0101A 2(3)PI-0101B 2(3)PI-0102A 2(3)PI-0103	P100X P100Y P102A P102B P102C P102D	613/635

Comments / Reference: From SO23-15-50.A1, 50A14	Revision # 8

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50A14 PZR PRESS HI/LO

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-3	AMBER	NO	50A04, 50A24, 50A34

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT [1]	PCS ID	LINK # U2/U3
2(3)PSHL-0100AX [2]	Pressurizer Pressure Switch HI/LO Channel X	HI 2275 PSIA LO 2175 PSIA	2(3)PI-0101A 2(3)PI-0101B 2(3)PI-0102A 2(3)PI-0103	P100X P100Y P102A P102B P102C P102D	623/645 624/646
2(3)PSHL-0100AY	Pressurizer Pressure Switch HI/LO Channel Y	HI 2275 PSIA LO 2175 PSIA			

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	059 G 2.2.44	
Importance Rating		4.4

Main Feedwater: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions affect plant and system conditions

Proposed Question: SRO 87

Given the following conditions on both Steam Generators at full power:

- Feedwater flow is GREATER THAN steam flow.
- Steam Generator narrow range levels are RISING.
- Feedwater Control System Master Controller outputs are LOWERING.
- Feedwater Control Valves are CLOSING.
- K006, Main Feedwater Pump speed is RISING.
- K005, Main Feedwater Pump speed is LOWERING.

Which ONE (1) of the following actions is required?

A. Ensure both Steam Generator Feedwater Control Systems are operating in AUTO per SO23-9-6, Feedwater Control System Operations.

B. Place K006 EAP/MSC in MANUAL and attempt to lower output per SO23-13-24, Feedwater Control System Malfunctions.

C. Place both Steam Generator Feedwater Control Valves in MANUAL and lower output per SO23-9-6, Feedwater Control System Operations.

D. Place both Steam Generator Master Controllers in Preferred Manual and attempt to lower level per SO23-13-24, Feedwater Control System Malfunctions.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because as long as the system was in AUTO one would expect level to be controlled, however, a Feedwater malfunction exists and entry into the AOI is required.
B. Correct. Given the conditions listed, placing K006, Main Feedwater Pump in MANUAL and lowering output to lower speed is the desired action. **Refer to hard copy of SO23-13-24, Attachment 1; Flowchart in PDF version is not legible.**
C. Incorrect. Plausible because this would help to regain level control, however, the guidance would come from the Abnormal Operating Instruction.
D. Incorrect. Plausible because this is a desired action to lower level, however, the Master Controller is already performing this function.

Technical Reference(s) SO23-13-24, Attachment 1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56415 As an SRO, given specific indications of a Feedwater Control System malfunction, DESCRIBE the process in determining the required actions in accordance with SO23-13-24.

Question Source:	Bank #	128188	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments / Reference: From SO23-13-24, Attachment 1 (best available)	Revision # 3

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REVISION 3
ATTACHMENT 1

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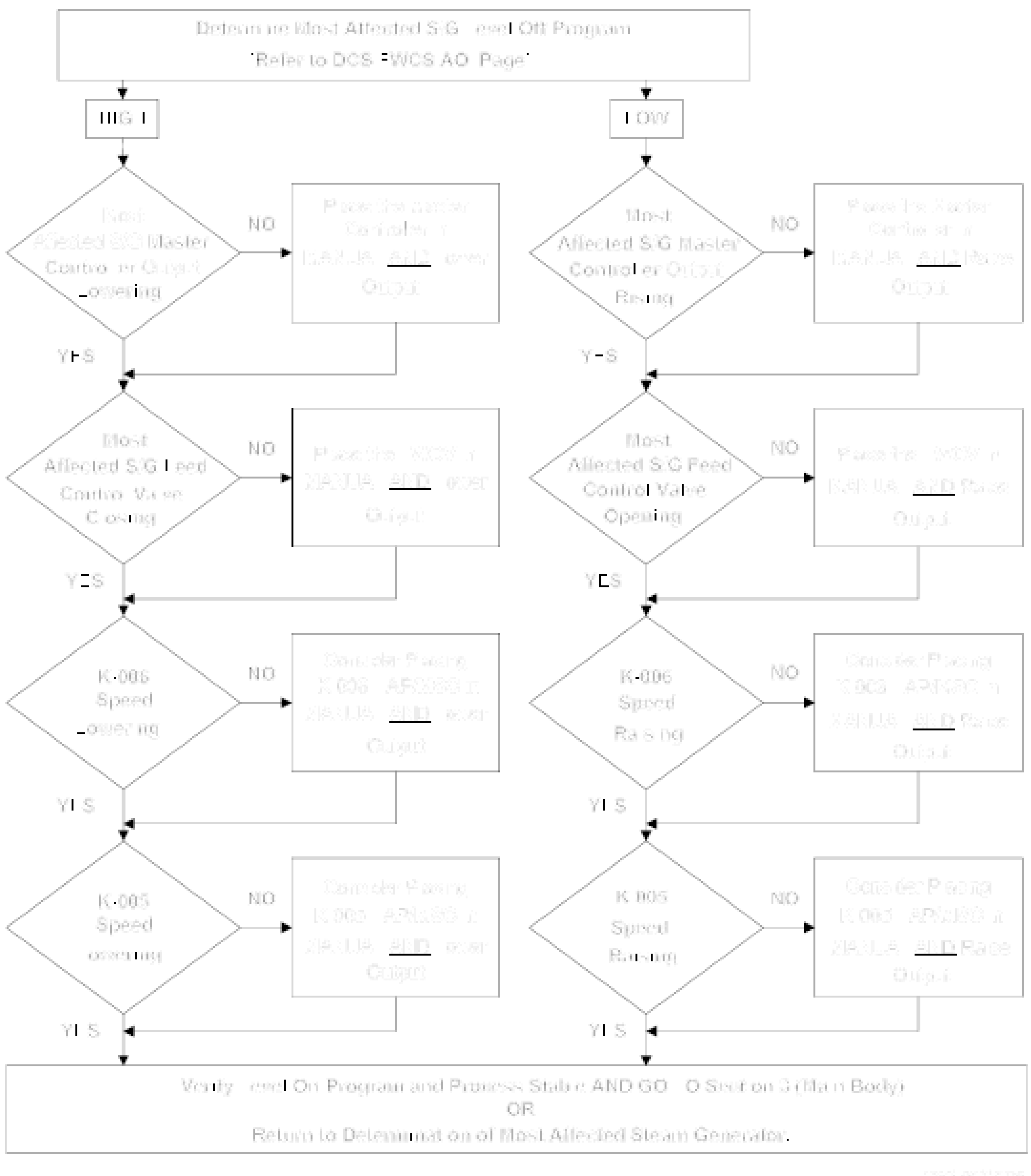
ADJUSTING STEAM GENERATOR LEVELS AFFECTS REACTIVITY

FEEDWATER CONTROL SYSTEM MALFUNCTION FLOWCHART



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	007 G 2.2.40	
	Importance Rating		4.7

Pressurizer Relief/Quench Tank System: Equipment Control: Ability to apply Technical Specifications for a system

Proposed Question: SRO 88

Given the following conditions:

- Unit 2 is operating at 100% power.
- The lift setpoint for PSV-0200, Pressurizer Safety Valve is determined to be out of tolerance (low) due to an error in the test calculation.
- PSV-0201, Pressurizer Safety Valve is OPERABLE.

Which ONE (1) of the following states the Technical Specification REQUIRED ACTION for this condition?

Restore within...

A. 15 minutes or be in MODE 3 in the next 6 hours.

B. one (1) hour or gag the safety valve and be in MODE 5 in the next 36 hours.

C. two (2) hours or be in MODE 4 within the next 24 hours.

D. four (4) hours or be in MODE 5 in the next 30 hours.

Proposed Answer: A

Explanation:

A. Correct. Per Technical Specification LCO 3.4.10, this is the REQUIRED ACTION and COMPLETION TIME. The valve must be restored within 15 minutes; however, MODE 3 entry is required in the next six hours.

B. Incorrect. Plausible because it could be thought that the ACTION time was 1 hour and that on a low setting, gagging the safety would be prudent and entry into MODE 5 would remove the heat energy for valve lift, however, the initial ACTION time is 15 minutes and MODE 3 entry is required in the next 6 hours per Tech Specs.

C. Incorrect. Plausible because it could be thought that two hours applies, however, MODE 3 entry is required in the next six hours per Tech Specs and the 12 hours to MODE 4 is the ACTION for two safeties INOPERABLE.

D. Incorrect. Plausible if thought that MODE 5 needed to be entered.

Technical Reference(s) Technical Specification LCO 3.4.10 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649
Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 ______
55.43 2

Comments / Reference: From Technical Specification LCO 3.4.10	Amendment # 156

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with as-found lift settings of 2500 psia, +3% or -2%. |

APPLICABILITY: MODES 1, 2, and 3.

----------------NOTE----------------

The lift settings are not required to be within LCO limits during MODE 3 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

Each pressurizer safety valve has an as-found tolerance of +3% or -2%. Following testing in accordance with TS 5.5.2.10, pressurizer safety valves shall be set within ±1% of the specified setpoint. |

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	013 A2.06	
	Importance Rating		4.0

Engineered Safety Features Actuation System: Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent ESFAS actuation

Proposed Question: SRO 89

Given the following conditions:

- The Unit is operating in MODE 1 when a Safety Injection Actuation Signal (SIAS) occurs.
- Pressurizer Pressure is 2245 psia.
- Containment Pressure is 0.3 psig.
- The cause of the SIAS was due to failure of a Matrix Channel.
- The Matrix Channel was declared INOPERABLE.

1.) Which ONE (1) of the following identifies the impact of the channel failure on Technical Specifications?

2.) What action must be taken to mitigate the situation?

A. 1.) Technical Specification LCO 3.3.5, ESFAS Instrumentation is applicable.
2.) Enter SO23-12-1, Standard Post Trip Actions and trip the Reactor and Turbine.

B. 1.) Technical Specification LCO 3.3.5, ESFAS Instrumentation is applicable.
2.) Enter SO23-3-2.7.2, Safety Injection System Removal/Return to Service Operation to reset the SIAS to restore Normal Containment Cooling.

C. 1.) Technical Specification LCO 3.3.6, ESFAS Logic and Manual Trip is applicable.
2.) Enter SO23-13-28, Rapid Power Reduction and place the Unit in MODE 3 within one hour.

D. 1.) Technical Specification LCO 3.3.6, ESFAS Logic and Manual Trip is applicable.
2.) Enter SO23-13-17, Recovery from Inadvertent Safety Injection/Containment Isolation or Containment Spray and override and stop all Charging Pumps.

Proposed Answer: D

Explanation:

A. Incorrect. The Tech Spec entry is incorrect. A Reactor trip and entry into SO23-12-1 per SO23-13-7, RCP Seal Failure would be considered because Controlled Bleed Off flow to the VCT is lost, however, the CBO relief would lift and the RCPs would be protected.
B. Incorrect. Plausible because the condition to reset SIAS to restore normal Containment Cooling is correct, however, the Technical Specification and procedure entry are both incorrect.
C. Incorrect. Plausible because the Tech Spec entry is correct, however, this would be the wrong procedure to enter because a Rapid Power Reduction is not required.
D. Correct. The SIAS must be recovered from in less than one hour to avoid Technical Specification LCO 3.0.3 entry. This is the correct Tech Spec and procedure entry. Letdown is isolated on an SIAS and the Charging Pumps must be overridden and stopped.

Technical Reference(s): Technical Specification LCO 3.3.6; SO23-13-17, Steps 3 & 5; SO23-13-6, Step 2; SO23-13-17, Steps 6k, 6l, & 6m

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53939 / 56649

As the SRO, DIRECT operator actions during recovery from an inadvertent safety injection / containment isolation per SO23-13-17.

Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Comments / Reference: From Technical Specification LCO 3.3.6	Amendment # 127

3.3 INSTRUMENTATION

3.3.6 Engineered Safety Features Actuation System (ESFAS) Logic and Manual Trip

LCO 3.3.6 Six channels of ESFAS Matrix Logic, four channels of ESFAS Initiation Logic, two channels of Actuation Logic, and two channels of Manual Trip shall be OPERABLE for each Function according to Table 3.3.6-1.

APPLICABILITY: According to Table 3.3.6-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ------NOTE------ This action also applies when three Matrix Logic channels are inoperable due to a common power source failure de-energizing three matrix power supplies. ------------- One or more Functions with one Matrix Logic channel inoperable.	A.1 Restore channel to OPERABLE status.	48 hours

Comments / Reference: From SO23-13-17, Step 3	Revision # 5

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RECOVERY FROM INADVERTENT SAFETY INJECTION/CONTAINMENT ISOLATION OR CONTAINMENT SPRAY

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3 **SIAS actuation actions:**	
a. VERIFY the following parameters met:	☐ a. TRIP the Reactor and Turbine
☐ 1) Pressurizer Pressure - ≥ 1740 psia	AND
	☐ GO TO S023-12-1
☐ 2) Containment Pressure - < 3.4 psig	AND
	☐ EXIT this procedure.
☐ b. ENSURE CVCS Letdown - ISOLATED	

CAUTIONS

1. Isolating Charging places the Unit in a Tech. Spec. LCO 3.0.3 action for inoperable Boration flow paths by rendering all Charging Pumps inoperable. SIAS shall be Reset within one hour; otherwise Tech. Spec. 3.0.3 shall be complied with. This is a management preapproved entry.

2. SIAS isolates Containment Normal Cooling, causing the CEDM shroud temperatures to rise rapidly. Loss of CEDM cooling for > one hour requires tripping the Reactor. CCAS is actuated to cool the CEDMs.

☐ c. OVERRIDE and STOP all Charging Pumps.

☐ d. REDUCE Turbine Load as necessary to MAINTAIN Turbine Power matched with Reactor Power.

☐ e. OVERRIDE and OPEN HV-5388, Instrument Air to Containment Isolation Valve.

f. OVERRIDE and OPEN RCP Bleed-off to VCT Isolation Valves:

☐ 1) HV-9217

☐ 2) HV-9218

☐ g. INITIATE CCAS.

☐ h. GO TO Step 6.

Comments / Reference: From SO23-13-6, Step 2	Revision # 5

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REVISION 5

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REACTOR COOLANT PUMP SEAL FAILURE

OPERATOR ACTIONS

2 **Immediate Diagnosis/actions:**

✓	AFFECTED PUMP CONDITIONS	ACTIONS
☞	3 Seal stages have failed.	☐ 1) Immediately TRIP the Reactor. ☐ 2) AFTER the CEAs have been inserted for 5 seconds, THEN TRIP the affected RCP(s). ☐ 3) GO TO SO23-12-1.
☞	Complete loss of CBO flow **AND** abnormal trends on multiple seal parameters.	

Comments / Reference: From SO23-13-17, Steps 6k, 6l, & 6m	Revision # 5

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RECOVERY FROM INADVERTENT SAFETY INJECTION/CONTAINMENT ISOLATION OR CONTAINMENT SPRAY

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

6 **Recover from inadvertent SIAS, as follows:** (Continued)

☐ g. OVERRIDE and CLOSE HV-9247, Emergency Boration Block Valve.

☐ h. INITIATE Attachment 2, Restoration of SIAS Actuated Breakers.

☐ i. NOTIFY opposite Unit SRO that S023-3-3.23, Attachment for A.C. Sources Verification, must be completed within 1 hour, due to the 1E 4kv Bus Tie Breakers being in Manual.

j. The Shift Manager SHALL:

☐ 1) NOTIFY the Manager, Plant Operations Units 2/3 or his designee.

☐ 2) DETERMINE the requirement for event classification per S0123-VIII-1.

☐ 3) DETERMINE reporting requirements of S0123-0-A7.

☐ k. VERIFY proper actuation per S023-3-2.22, Attachment for SIAS/CCAS Actuation Verification.

☐ l. PERFORM S023-3-2.22, Attachment for SIAS/CCAS and CIAS Reset.

☐ m. PERFORM S023-3-2.22, Attachment for SIAS/CCAS Restoration.

Comments / Reference: From SO23-13-17, Step 5	Revision # 5

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RECOVERY FROM INADVERTENT SAFETY INJECTION/CONTAINMENT ISOLATION OR CONTAINMENT SPRAY

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5 **Recover from Inadvertent CIAS, as follows:**

a. Restore the Non-Critical Loop and flow to the RCPs, as follows:

☐ 1) ENSURE a CCW/SWC loop operating.

☐ 2) DEPRESS HS-6397-1 to override the CIAS.

☐ 3) DEPRESS HS-6397-2 to override the CIAS.

☐ 4) OPEN HV-6212 and HV-6218, Noncritical Loop Supply and Return Valves (Loop A)

OR

☐ 5) OPEN HV-6213 and HV-6219, Noncritical Loop Supply and Return Valves (Loop B)

☐ 6) OPEN HV-6211, NCL Supply Containment Isolation Valve.

☐ 7) OPEN HV-6223, NCL Supply Containment Isolation Valve.

☐ 8) OPEN HV-6216, NCL Return Containment Isolation Valve.

☐ 9) OPEN HV-6236, NCL Return Containment Isolation Valve.

☐ b. VERIFY proper actuation per SO23-3-2.22, Attachment for CIAS Actuation Verification.

☐ c. PERFORM SO23-3-2.22, Attachment for SIAS/CCAS and CIAS Reset.

☐ d. PERFORM SO23-3-2.22, Attachment for CIAS Restoration.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	076 G 2.4.21	
	Importance Rating		4.6

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

Proposed Question: SRO 90

Given the following conditions on Unit 2:

- SO23-12-9, Functional Recovery was entered due to a Small Break Loss of Coolant Accident and Excess Steam Demand Event.
- Offsite power is unavailable and Train B Emergency Diesel Generator failed to start.
- Containment pressure is 17 psig.
- Reactor Coolant System pressure is 1600 psia and the Pressurizer is empty.
- Subcooled Margin is 2°F.
- All Train A Engineered Safety Feature actuations occurred as required.
- Reactor Vessel Level is 21% in the Plenum.

While performing Safety Function Status Checks you learn that the only available Salt Water Cooling Pump has just tripped.

1.) Which ONE (1) of the following identifies the highest Safety Functions that are NOT met?

2.) Which Functional Recovery Procedure will be entered FIRST?

A. 1.) RCS Inventory Control and RCS Pressure Control.
2.) Implement FR-3, Recovery - RCS Inventory Control.

B. 1.) RCS Pressure Control and Core Heat Removal.
2.) Implement FR-4, Recovery - RCS Pressure Control.

C. 1.) Core Heat Removal and RCS Heat Removal.
2.) Implement FR-5, Recovery - Heat Removal.

D. 1.) Containment Isolation and Containment Temperature and Pressure Control.
2.) Implement FR-6, Recovery - Containment Isolation.

Proposed Answer: A

Explanation:

A. Correct. These are highest Safety Functions affected and the highest priority Safety Function in jeopardy is RCS Inventory Control.
B. Incorrect. Plausible because it could be thought that the Core Heat Removal Safety Function was the highest priority Safety Function jeopardized due to the loss of the SI Pumps but the challenge doesn't exist if SG cooling is working and CETs are less than 700°F.
C. Incorrect. Plausible because it could be thought that both Heat Removal Safety Functions were jeopardized which would require entry into FR-5.
D. Incorrect. Plausible because Containment pressure is 17 psig with no indication that it is lowering, however, FR-6 would be implemented later if conditions did not improve.

Technical Reference(s) SO23-12-10, Attachment SF-3, Steps 3 to 8 Attached w/ Revision # See Comments / Reference
SO23-12-9, Attachment FR-3, Step 4

Proposed references to be provided during examination: None

Learning Objective: 55280 / 55217
EXPLAIN the different methods of recovery that are unique to the Functional Recovery Emergency Operating Instruction (EOI).
Per the Functional Recovery procedure SO23-12-9 DESCRIBE: The basis for each step, caution or note.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 3	Revision # 3

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ATTACHMENT SF-3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
3 RCS Inventory Control	
CONDITION 1	RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC.
a. RCS inventory: 1) PZR level – between 10% and 70% OR – greater than 70% for the purpose of compensating for void collapse. AND 2) Charging and/or Letdown or SI available to maintain PZR level.	• IF re-evaluation identifies another event, NOT Loss of Coolant Accident, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Loss of Coolant Accident OR b) More than one event,
b. Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311.	THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-3, RECOVERY – RCS INVENTORY CONTROL.
c. Reactor Vessel level – greater than or equal to 100% (Plenum): QSPDS page 622 CFMS page 312 Attachment SF-10.	

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 3	Revision # 3

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EMERGENCY OPERATING INSTRUCTION
REVISION 3
ATTACHMENT SF-3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
3 RCS Inventory Control (Continued)	
CONDITION 2 a. SI flow: 1) In Cold Leg Injection Mode – greater than or equal to minimum limits of Figure 1, MINIMUM REQUIRED SI FLOWRATES DURING COLD LEG INJECTION. OR 2) In Hot/Cold Leg Injection Mode – greater than or equal to minimum limits of Figure 2, MINIMUM REQUIRED HPSI FLOWRATES DURING HOT/COLD LEG INJECTION. b. Reactor Vessel level – greater than or equal to 41% (Plenum): QSPDS page 622 CFMS page 312 Attachment SF-10.	◘ RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT Loss of Coolant Accident, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Loss of Coolant Accident OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-3, RECOVERY – RCS INVENTORY CONTROL.

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 4	Revision # 3

NUCLEAR ORGANIZATION
UNITS 2 AND 3

EMERGENCY OPERATING INSTRUCTION
REVISION 3
ATTACHMENT SF-3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA

4 RCS Pressure Control

CONDITION 1

a. Core Exit Saturation Margin
– between 20°F and 160°F:

QSPDS page 611
CFMS page 311.

CONDITION 2

a. SI flow:

1) In Cold Leg Injection Mode
– greater than or equal to minimum limits of Figure 1, MINIMUM REQUIRED SI FLOWRATES DURING COLD LEG INJECTION.

OR

2) In Hot/Cold Leg Injection Mode
– greater than or equal to minimum limits of Figure 2, MINIMUM REQUIRED HPSI FLOWRATES DURING HOT/COLD LEG INJECTION.

ACCEPTANCE CRITERIA NOT MET

- RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC.

  - IF re-evaluation identifies another event, NOT Loss of Coolant Accident,

    THEN GO TO identified EOI.

  - IF re-evaluation identifies:

    a) Loss of Coolant Accident

    OR

    b) More than one event,

    THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY*

    AND

    INITIATE SO23-12-9, Attachment FR-4, RECOVERY – RCS PRESSURE CONTROL.

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 5	Revision # 3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET

5 Core Heat Removal

a. REP CET temperature
– less than 700°F:

QSPDS page 611
CFMS page 311.

- RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC.
  - IF re-evaluation identifies another event, NOT Loss of Coolant Accident,

    THEN GO TO identified EOI.
  - IF re-evaluation identifies:

    a) Loss of Coolant Accident

    OR

    b) More than one event,

    THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY*

    AND

    INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 6	Revision # 3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
6 RCS Heat Removal	
a. Any S/G: 1) Level – between 40% NR and 80% NR AND Feedwater – available. OR 2) Level – trending to between 40% NR and 80% NR AND Feedwater Flow confirmed – when level less than 40% NR. b. 1) Single-phase Operating Loop RCS flow: RCS T_C – stable or lowering. OR 2) Two-phase Operating Loop RCS flow: REP CET trend – stable or lowering.	o RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT Loss of Coolant Accident, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Loss of Coolant Accident OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 7	Revision # 3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET

7 Containment Isolation

NOTE

During extreme containment temperature transient conditions such as LOCA, the Containment Area High Range Radiation Monitors may indicate a dose rate as high as 100 R/Hr due to thermally induced currents on their signal cables.

a. Containment pressure
   – less than 3.4 PSIG.

   OR

   CIAS – actuated.

b. Containment Area Radiation Monitors
   – NOT alarming or trending to alarm

R7845	Access Hatch
R7848	General Area
R7820-1	Containment (High)
R7820-2	Containment (High).

   OR

   SIAS – actuated.

   OR

   CIAS – actuated.

c. Secondary Radiation Monitors
   – NOT alarming or trending to alarm.

R7870	Air Ejector, WRGM.
R7818	Air Ejector
R6759	E088 Blowdown
R7874A/R7875A	E088 Steamline
R6753	E089 Blowdown
R7874B/R7875B	E089 Steamline.

- RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC.

  - IF re-evaluation identifies another event, NOT Loss of Coolant Accident,

    THEN GO TO identified EOI.

  - IF re-evaluation identifies:

    a) Loss of Coolant Accident

    OR

    b) More than one event,

    THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY*

    AND

    INITIATE SO23-12-9, Attachment FR-6, RECOVERY – CONTAINMENT ISOLATION.

Comments / Reference: From SO23-12-10, Attachment SF-3, Step 8	Revision # 3

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ATTACHMENT SF-3

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SAFETY FUNCTION STATUS CHECK

LOSS OF COOLANT ACCIDENT

SAFETY FUNCTION ACCEPTANCE CRITERIA	ACCEPTANCE CRITERIA NOT MET
8 Containment Temperature and Pressure Control	
CONDITION 1 a. Containment average temperature – less than 205°F. b. Containment pressure – less than 14 PSIG. CONDITION 2 a. Containment Spray: 1) Containment Spray Train A flow – greater than 1600 GPM. 2) Containment Spray Train B flow – greater than 1600 GPM. 3) Containment pressure – less than 60 PSIG.	○ RE-EVALUATE event per, Attachment SF-1, RECOVERY DIAGNOSTIC. • IF re-evaluation identifies another event, NOT Loss of Coolant Accident, THEN GO TO identified EOI. • IF re-evaluation identifies: a) Loss of Coolant Accident OR b) More than one event, THEN GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-7, RECOVERY – CONTAINMENT TEMPERATURE AND PRESSURE CONTROL.

Comments / Reference: From SO23-12-9, Attachment FR-3, Step 4	Revision # 25

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ATTACHMENT FR-3

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FUNCTIONAL RECOVERY

RECOVERY – RCS INVENTORY CONTROL

RCS INVENTORY CONTROL RECOVERY ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 INITIATE Leak Isolation: (Continued)	
c. ENSURE Pressurizer Vent valves – closed: HV-0297A HV-0297B.	
d. ENSURE Reactor Vessel Head Vent valves – closed: HV-0296A HV-0296B.	
e. ENSURE Combined PZR/Reactor Vessel Vent valves – closed: HV-0298 HV-0299.	
f. VERIFY: 1) Outside Containment radiation alarms – NOT alarming or trending to alarm. 2) Outside Containment Sumps – NOT abnormally rising.	f. ENSURE SIAS – actuated AND REQUEST Shift Manager/Operations Leader: 1) EVALUATE possible LOCA outside Containment 2) INITIATE FS-20, MONITOR RWST Level. 3) EVALUATE CIAS actuation.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	045 G 2.1.20	
	Importance Rating		4.6

Main Turbine Generator System: Conduct of Operations: Ability to interpret and execute procedure steps

Proposed Question: SRO 91

Given the following conditions:

- Unit 3 is at 75% power and performing a shutdown from 100% power for Refueling.
- Unit 2 has been at 100% power for 95 days and is performing monthly In Service Testing on a standby Charging Pump.
- Annunciator 50A02 - COLSS ALARM actuates on Unit 2 and a steady slow power ramp to 100.06% is observed with a slight drop in T_{COLD}.

Which ONE (1) of the following describes the cause of this alarm and the actions required to mitigate the event?

A. Auxiliary Steam is aligned to Unit 2 and is starting to supply the Unit 3 Turbine Gland Sealing Steam.
Enter SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration and immediately borate to restore Shutdown Margin.

B. Auxiliary Steam is aligned to Unit 2 and is starting to supply the Unit 3 Turbine Gland Sealing Steam.
Enter SO23-5-1.7, Power Operations and immediately reduce power to less than or equal to 100%.

C. The standby Charging Pump had reduced boron concentration in its lines when it was started.
Enter SO23-13-11, Emergency Boration of the RCS / Inadvertent Dilution or Boration and immediately borate to restore Shutdown Margin.

D. The standby Charging Pump had elevated boron concentration in its lines when it was started.
Enter SO23-5-1.7, Power Operations and immediately reduce power to less than or equal to 100%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that there would be inadequate SHUTDOWN MARGIN with power at this level. If SHUTDOWN MARGIN were inadequate this would be the correct procedure to enter.

B. Correct. The Turbine Gland System is self-sealing above 80% power. Once power drops below 80%, Gland Sealing Steam will be supplied by Auxiliary Steam which will raise power on the supplying Unit. The action required is to reduce power to within 100% per SO23-5-1.7.

C. Incorrect. Plausible because it could be thought that there could be reduced boron in the Charging Pumps causing power to rise, however, an idle Charging Pump during an extended run is not likely to have reduced boron in the lines and a dilution would result in rising T_{COLD}.

D. Incorrect. Plausible because the procedural action is correct, however, an idle Charging Pump during an extended run is not likely to have elevated boron in the lines and a boration would result in a lowering T_{COLD}.

Technical Reference(s): SO23-5-1.7, L&S 5.21; SO23-5-1.7, Step 6.1.2; SO23-5-1.7, L&S 2.1; SO23-15-50.A1, 50A02 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55143 — During Power Operations, EXPLAIN the precaution, limitation or administrative requirement applicable to plant condition per SO23-5-1.7.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 1, 5

Comments / Reference: From SO23-5-1.7, L&S 5.21	Revision # 41

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SO23-5-1.7
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5.0 **SECONDARY PLANT** (continued)

5.17 The MSBSCAL Program provides the possibility of gaining additional electrical generation by using Main Steam Flow for the COLSS power calculation (BSCAL). The COLSS Calorimetric Selection Display Interface (COLSS CAL SELECT PAGE) has been designed to latch the most recent choice for the Default Secondary Calorimetric State. When MSBSCAL is selected (default), and power is > 92% for Unit 2 or > 95% for Unit 3, then MSBSCAL is automatically selected. (AR 050200596-01)

5.17.1 When the Advanced Measurement Analysis Group Crossflow System (AMAG) is Out of Service, then FWBSCAL may indicate approximately 1% higher than MSBSCAL. If MSBSCAL indicates >99% power, and COLSS is transferred to FWBSCAL, then indicated power may exceed 100%.

5.18 Rinsing the FFCPD to the Condenser causes perturbations in the Condensate and Feedwater Systems that can affect FWBSCAL and may cause the COLSS alarm to annunciate. This effect is not observed on MSBSCAL. (AR 030300700)

5.19 COLSS defaults to FWBSCAL when Unit 2 power is ≤92% or Unit 3 power is ≤ 95%. If reducing power at or below these values, then consideration should be given to manually transferring both COLSS Primary and Backup to FWBSCAL.

5.20 FT-3432A and FT-3433A are the same FT's that input to the miniflow controller. Assuring at least 10,500 gpm MFP Suction Flow will ensure the associated miniflow closes and remains closed (setpoint = 7000 gpm + 3500 gpm thru mini-flow = 10,500 gpm) when placed in MODULATE. When MFP miniflow valves are closed, a 30 MWe Unit power improvement is expected and condensate flow will drop by approximately 3500 gpm for each miniflow valve that is closed.

5.21 When Turbine load is reduced below 80%, the Turbine Gland seals begin to become "non-self sealing". This causes an increase in Aux Steam demand and a corresponding decrease in COLSS Power Margin for the Unit supplying Aux Steam. If the Unit supplying Aux Steam is at 100% power operation, the COLSS alarm should be anticipated and Unit load adjusted accordingly.

Comments / Reference: From SO23-5-1.7, Step 6.1.2	Revision # 41

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6.0 PROCEDURE

6.1 Guidelines for Steady State Power Operation

REFERENCE USE

6.1.1 **System Parameter Guidelines**

.1 REFER to Attachment 14 for parameter monitoring and control band guidelines.

6.1.2 **Power Level Guidelines** (LS-2.14)

.1 MAXIMIZE Unit generation within the limits of Attachment 5. (LS-5.10 and LS-5.28)

.2 MAINTAIN Reactor Power constant by diluting or borating as required per SO23-3-2.2, Section for Dilution Makeup Mode OR Section for Borating to the Charging Pump Suction. (LS-2.5)

.3 MATCH Turbine Generator load with Reactor Power changes per Section 6.3.

.4 MAINTAIN Power such that 50A02, COLSS Alarm, is not annunciated, except as allowed per ARP.

.5 If 100.6% Rated Thermal Power (BSCAL, CV9005AVG) was approached, then REQUEST the STA or Reactor Engineering to determine if 100.6% RTP was exceeded. (LS-2.2 and LS-5.17) |

.6 If Excore Linear Power, CPC ΔT Power (PID-177), or CPC Nuclear Power (PID 171) indication exceeds 101.5%, then PERFORM SO23-3-3.2, Excore Nuclear Instrumentation Calibration. (LS-2.1) |

Comments / Reference: From SO23-5-1.7, L&S 2.1	Revision # 41

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2.0 **POWER GUIDELINES**

2.1 LIMIT: SONGS is required to not exceed 100% Rated Thermal Power (RTP). The most accurate measure of power is secondary calorimetric (BSCAL, CV9005AVG). Power is monitored by COLSS (continuously) or by manual calculations (every 15 minutes). The excore nuclear instruments and core ΔT power are allowed to deviate from the secondary calorimetric by -1%/+5% and are verified once per shift per SO23-3-3.25. As long as BSCAL does not exceed 100% RTP averaged over an eight hour period it is acceptable for the NI/core ΔT powers to exceed 100%. BSCAL may fluctuate slightly above 100% without operator action. When it is determined by BSCAL that RTP has exceeded 100%, then action shall be taken to return power to ≤ 100%. (Ref. 2.4.2.10)

2.1.1 While it is acceptable for the NI/Core ΔT powers to exceed 100%, they should not exceed 102%.

Comments / Reference: From SO23-15-50.A1, 50A02	Revision # 8

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ALARM RESPONSE INSTRUCTION
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50A02 COLSS ALARM (Continued)

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 Plant power exceeds licensed power limit.	2.1 Reduce Reactor power per SO23-5-1.7, Section on Guidelines for Changing Turbine Load and Reactor Power. 2.1.1 IF the power increase is due to an inadvertent dilution, THEN initiate SO23-13-11, Emergency Boration of the RCS/Inadvertent Dilution or Boration.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	071 G 2.1.32	
	Importance Rating		4.0

Waste Gas Disposal System: Conduct of Operations: Ability to explain and apply system limits and precautions

Proposed Question: SRO 92

Given the following conditions:

- Unit 3 is performing a Waste Gas Decay Tank release using 2/3 RT-7808, Plant Vent Stack and Containment Purge Radiation Monitor.
- During the release, 3RT-7865, Containment Purge Radiation Monitor is declared INOPERABLE.

Per the Offsite Dose Calculation Manual (ODCM), which ONE (1) of the following is required?

A. Verify 2/3RT-7808 is OPERABLE.

B. Analyze two independent samples prior to continuing.

C. Estimate the process flow rate at least once per 12 hours.

D. Verify valve alignment, sample and release rate calculations.

Proposed Answer: A

Explanation:

A. Correct. Per ODCM, Table 4-3, a minimum of one Channel must be OPERABLE.

B. Incorrect. Plausible because ACTION 35a states that when the number of OPERABLE Channels is less than required, at least two independent samples of the tank contents should be analyzed.

C. Incorrect. Plausible because ACTION 36a states that when the number of OPERABLE Channels is less than required, the process flow rate should be estimated at least once per 12 hours.

D. Incorrect. Plausible because ACTION 35b states that when the number of OPERABLE Channels is less than required, the valve alignment and release rate calculations must be re-verified.

Technical Reference(s): SO123-ODCM, Table 4-3; SO123-ODCM, Table 4-3, ACTIONs 35a & 35b; SO123-ODCM, Table 4-3, ACTION 36a

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 53393 Given plant conditions concerning an effluent release, DETERMINE the applicable technical specification/administrative requirements or limitations.

Question Source:	Bank #	71065	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	1, 4	

Comments / Reference: From SO123-ODCM, Table 4-3 — Revision # 0

TABLE 4-3

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT***	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release - 2/3RT-7808, or 3RT-7865-1	1	*	35
b. Process Flow Rate Monitoring Device	1	*	36a

Comments / Reference: From SO123-ODCM, Table 4-3, ACTIONs 35a & 35b	Revision #

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

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

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

Otherwise, suspend releases of radioactive effluents via this pathway.

4-12

SO123-ODCM
Revision 0
02-27-07

Comments / Reference: From SO123-ODCM, Table 4-3, ACTIONs 36a	Revision # 0

TABLE 4-3 (Continued)

TABLE NOTATION

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided:

a. The process flow rate is estimated at least once per 12 hours during actual releases. In addition, a new flow estimate shall be made within 1 hour after a change that affects process flow has been completed. System design characteristics may be used to estimate process flow.

b. The particulate and iodine (P&I) sample flow rate is estimated or verified at least once per 12 hours during actual releases.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034 A3.02	
	Importance Rating		3.1

Fuel Handling Equipment System: Ability to monitor automatic operation of the Fuel Handling System, including: Load limits

Proposed Question: SRO 93

Given the following conditions during Refueling:

- A Spent Fuel Assembly was placed in the Fuel Transfer Carriage inside Containment.
- While transferring the Assembly between the Containment and Fuel Handling Building, the Fuel Transfer Carriage became lodged in the Transfer Canal.
- Transfer stopped when the 800 pound overload limit was reached.

Which ONE (1) of the following actions is required?

A. Evacuate Containment only.

B. Notify Health Physics to monitor radiation levels in the Fuel Transfer Tube Area.

C. Evacuate the Fuel Handling Building only.

D. Notify Health Physics to monitor radiation levels in the Radwaste Building.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that Containment must be evacuated because of the location of the Transfer Canal, however, this area is inaccessible to personnel during refueling.
B. Correct. Given the conditions listed, this the action to be taken for a stuck Fuel Assembly.
C. Incorrect. Plausible because there are individuals inside the Fuel Handling Building, however, it is the Fuel Transfer Tube Area (Penetration Room 111) that must be monitored.
D. Incorrect. Plausible if thought that the Radwaste Building was the location of the Transfer Canal.

Technical Reference(s) SO23-X-7, Precautions 4.43 and 4.51; SD-SO23-430, Pages 75 & 76 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54861 — As the SRO, DIRECT the response to Fuel Handling Accidents/Loss of Cavity or SFP Level Control per SO23-13-20.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	7	

Comments / Reference: From SO23-X-7, Precaution 4.43	Revision # 16

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4.41 Items such as tools, spacers, etc., may be stored in the spent fuel racks if their weight is less than 100 pounds. Items whose dimensions are similar to a fuel assembly and whose weight is less than 1500 pounds may also be stored in the racks.

EXAMPLES: trash baskets, dummy fuel assemblies, CEA carriers, etc.

Items which contain fuel, such as fuel rod storage baskets, are limited to a maximum drop height of 21.7 feet, by sizing these containers to the same height as a fuel assembly.

4.42 The rod storage baskets are currently approved for storage of fuel rods containing up to 4.8% initial U-235 enrichment (w/o).

4.42.1 Rod storage baskets shall be treated as if it were an assembly with enrichment and burnup of the rod in the basket with the most limiting combination of enrichment and burnup, in accordance with LCS 4.0.100.7.

4.43 Notify HP at the 70' Control point (86695) if planning to stop an irradiated fuel bundle or other highly radioactive items in the U2 or U3 transfer tube. Notify HP at the 70' Control Point (86695) if the transfer carriage, while carrying an irradiated fuel bundle or other highly radioactive items, stops in the U2 or U3 transfer tube and cannot be restarted within 3 minutes. Advise HP that dose rates in the area around the fuel transfer tube in penetration Room 111 of the affected unit could increase.

Comments / Reference: From SO23-X-7, Precaution 4.51	Revision # 16

NUCLEAR ORGANIZATION UNITS 2 AND 3 — NUCLEAR FUELS SERVICES PROCEDURE REVISION 16 — SO23-X-7 PAGE 24 OF 198

4.51 Only one fuel assembly containing a CEA may be placed in the upender at a time, pending reevaluation of the transfer system load rating (reference AR # 080100570 and order # 200003889). Placing two fuel assemblies in the upender, one containing a CEA, is acceptable.

Comments / Reference: From SD-SO23-430, Pages 75 & 76	Revision # 12

2.3.15 **Transfer Carriage (Manual):**

Transferring the carriage from one side of the transfer tube to the other in manual is accomplished by an operator on either console turning the "CARRIAGE TRANSFER" selector switch to the name of the side required. When the input is detected, the output for the carriage motor brake is turned on, the drive enable output is closed, and the drive speed reference is set for slow speed. When the carriage is fully in the transfer tube the speed reference is changed to the high speed setting, until the end of the tube is reached, where slow speed is again commanded. For carriage motion to be operated the following conditions are required:

.1 The carriage in position switch on the spent fuel pool side must be open if the carriage is commanded to that side.

.2 The carriage in position switch on the reactor side must be open if the carriage is commanded to that side.

.3 A normal overload of 800 LBS cannot exist or one of the load override pushbuttons must be held depressed while moving in the reverse direction of the overload.

.4 The Max. overload of 2000 LBS cannot be activated.

.5 An automatic sequence is not in operation.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		1
K/A #	G 2.1.5	
Importance Rating		3.9

Conduct of Operations: Ability to use procedures related to shift staffing, such as minimum crew complement, over time limitations, etc.

Proposed Question: SRO 94

Given the following:

- Unit 2 is in MODE 1.
- The shift is manned to the minimum composition per Appendix R.
- The shift has 4 hours remaining.
- The 21 Watch has become ill and must leave the site for emergency medical treatment.

Which ONE (1) of the following describes the requirements regarding the shift composition and required action in this situation?

A. The 21 Watch may leave the site immediately after turnover of responsibilities to another qualified person on shift. A replacement must arrive within 2 hours.

B. The 21 Watch may NOT leave the site until minimum manning has been maintained by calling in a qualified relief.

C. Responsibilities of the 21 Watch may be turned over to the 22 Watch for the remainder of the shift.

D. The CRS may assume the responsibilities of the 21 Watch. The Shift Manager may perform concurrent SM/CRS duties until shift relief with a qualified STA on site.

Proposed Answer: A

Explanation:

A. Correct. A medical emergency meets the requirements of an "unexpected absence" and the individual is allowed to leave. Immediate action must be taken to restore the shift complement to the minimum requirements.

B. Incorrect. Plausible because if there is an emergency they may leave, however, the position must be filled within 2 hours.

C. Incorrect. Plausible because the crew may choose to assume responsibilities as they need to, however, there is too much time left on shift (4 hours) to not have a replacement.

D. Incorrect. Plausible because the crew may choose to assume responsibilities as they need to, however, a maximum of 2 hours below minimum manning is allowed.

Technical Reference(s) SO123-0-A1, Attachment 2
Technical Specifications Section 5.2.2
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55164
Given plant conditions, DETERMINE if license conditions have been violated per 10CFR50.54.

Question Source:
Bank # 151972
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam SONGS 2006

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

10 CFR Part 55 Content:
55.41
55.43 2, 5

Comments / Reference: From SO123-0-A1, Attachment 2	Revision # 18

NUCLEAR ORGANIZATION
UNITS 1, 2 AND 3

OPERATIONS DIVISION PROCEDURE
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MINIMUM OPERATIONS SHIFT CREW COMPOSITION
(Tech Spec 5.2.2.a, 5.2.2.b, 5.2.2.c) (LCS 5.0.100.1; 5.0.100.1.1, Table 5.0.100-1)

IF Unit 2 is in...	**AND Unit 3 is in...**	**THEN the minimum Operations Shift crew composition is...** [1] [2]
MODE 1-4	MODE 1-4	1 SM (SRO) [5] [7] 2 CRSs (SRO) [7] 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [9] 1 FTA [8] [9] 1 Appendix R [8]
MODE 5-6	MODE 5-6	1 SM (SRO) [5] 1 CRS (SRO) 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [6] [9] 1 FTA [8] [9] 1 Appendix R [8]
MODE 1-4	MODE 5-6	1 SM (SRO) [5] [7] 2 CRSs (SRO) [7] 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [6] [9] 1 FTA [8] [9] 1 Appendix R [8]
MODE 5-6	MODE 1-4	1 SM (SRO) [5] [7] 2 CRSs (SRO) [7] 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [6] [9] 1 FTA [8] [9] 1 Appendix R [8]
MODE 1-4	CORE ALTS	1 SM (SRO) [5] [7] 3 CRSs (SRO) [3] [7] 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [9] 1 FTA [8] [9] 1 Appendix R [8]
CORE ALTS	MODE 1-4	1 SM (SRO) [5] [7] 3 CRSs (SRO) [3] [7] 5 CO/ACOs (RO) [4] 4 Primary Qualified NPEOs 1 STA [9] 1 FTA [8] [9] 1 Appendix R [8]

Comments / Reference: From SO123-0-A1, Attachment 2	Revision # 18

NUCLEAR ORGANIZATION
UNITS 1, 2 AND 3

OPERATIONS DIVISION PROCEDURE
REVISION 18
ATTACHMENT 2

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MINIMUM OPERATIONS SHIFT CREW COMPOSITION
(Continued)

Footnotes

[1] Except for the Shift Manager, the shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided IMMEDIATE ACTION is taken to restore the shift crew composition to within the minimum requirements. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewperson being late or absent. (Tech. Spec. 5.2.2.c, LCS 5.0.100.1.1)

[2] At least one ACTIVELY LICENSED Reactor Operator shall be in the Control Room when fuel is in the Reactor. (Tech. Spec. 5.2.2.b)

[3] Shift crew assignments during periods of CORE ALTERATIONS, shall include an additional LICENSED Senior Reactor Operator (SRO), who has no other concurrent duties during this operation, to directly supervise the CORE ALTERATION from the refueling deck inside Containment. (LCS 5.0.100.1.2)

[4] The minimum Tech. Spec. limit is 3 ROs. Normal RO staffing would include the 21, 31, 22, 32, ARO, and 41/51. (AR 070401390-4)

[5] Individual may fill position for both Units.

[6] To support shutdown safety assessment, the STA position shall be continuously manned in Modes 5 and 6 as directed by Management. (Ref. 2.3.4)

[7] While the Unit is in Mode 1, 2, 3 or 4, at least one ACTIVELY LICENSED Senior Reactor Operator shall be in the Control Room Area. (Tech. Spec. 5.2.2.b)

[8] The Fire Technical Advisor (FTA) position <u>must</u> be filled with a Licensed Reactor Operator who does not have concurrent Appendix "R" responsibilities. (AR 070401390-4)

[9] The WPS may be designated as the Fire Technical Advisor if not concurrently filling the STA position. (AR 070401390-4)

Comments / Reference: From Technical Specifications Section 5.2.2	Amendment # 207

5.2.2 UNIT STAFF

The unit staff organization shall include the following:

a. A non-Licensed Operator shall be assigned to each reactor containing fuel and an additional non-Licensed Operator shall be assigned for each unit when a reactor is operating in MODES 1, 2, 3, or 4.

With both units shutdown or defueled, a total of three non-Licensed operators are required for the two units.

b. At least one licensed Reactor Operator (RO) shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator (SRO) shall be in the Control Room Area.

c. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 5.2.2.a for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

d. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative controls shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., senior reactor operators, reactor operators, auxiliary operators, health physicists, and key maintenance personnel). The controls shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized in advance by the cognizant corporate officer, or designees, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the cognizant corporate officer, or designees, to ensure that excessive hours have not been assigned. Routine deviation from the above guidelines shall not be authorized.

f. The Manager, Plant Operations (at time of appointment), Shift Managers, and Control Room Supervisors shall hold a Senior Reactor Operator's license.

g. The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Manager in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G 2.1.23	
	Importance Rating		4.4

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation

Proposed Question: SRO 95

Given the following conditions:

- SO23-12-8, Station Blackout is currently in use.
- Subsequently, a Loss of Coolant Accident occurs.
- Qualified Safety Parameter Display System is OPERABLE.
- Pressurizer level has been below the indicating range for greater than 1 hour.
- Core Exit Thermocouple temperature is 713°F.
- Turbine Driven Auxiliary Feedwater Pump is in operation.

Based on the conditions listed, which ONE (1) of the following is the Control Room Supervisor's top priority?

A. Remain in SO23-12-8, Station Blackout, and attempt to restore power to at least one 1E 4160 Volt Bus.

A. Determine that core damage is in progress. Remain in SO23-12-8, Station Blackout and implement the Severe Accident Management Guidelines.

B. Reactor Vessel water level is below the top of the core. Transition to SO23-12-9, Functional Recovery and implement FR-3, Recovery - RCS Inventory Control.

B. The core has just reached saturation conditions and SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control should be implemented.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because restoration of power to a Safeguards Bus is a priority, however, Inadequate Core Cooling conditions exist and entry into SO23-12-9 is required.
B. Incorrect. Plausible because the Severe Accident Management Guidelines are available for use, however, conditions have not degraded to the point where the SAMGs would be implemented.
C. Correct. Given the conditions listed, entry into the Functional Recovery Procedure is required.
D. Incorrect. Plausible because feedwater is available, however, superheat conditions exist regardless of RCS pressure and entry into the Functional Recovery Procedure is required.

Technical Reference(s) SO23-12-9, SO23-12-9, Entry Conditions; SO23-14-9, FR-3 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55217 — Per the Functional Recovery procedure SO23-12-9 DESCRIBE: The basis for each step, caution or note.

Question Source: Bank # 127126
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam SONGS 2005A

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: From SO23-12-9, Entry Conditions	Revision # 25

FUNCTIONAL RECOVERY

PURPOSE

To provide a systematic approach to mitigate any event that actuates or requires a Reactor Trip, for which:

1 A diagnosis is NOT confirmed.

OR

2 A diagnosis of multiple malfunctions occurs.

OR

3 An optimal recovery instruction is NOT adequately recovering the plant.

The actions specified maintain each Safety Function.

ENTRY CONDITIONS

1 **ANY** of the following conditions present:

- An event diagnosis is NOT confirmed.
- Multiple events are diagnosed.
- SRO Operations Supervisor specifies Functional Recovery due to actions from an optimal recovery EOI.
- Optimal recovery EOI safety function acceptance criteria NOT satisfied.

AND

SO23-12-1, STANDARD POST TRIP ACTIONS, steps 1 through 10 have been completed (Modes 1 and 2)

OR

2 The event initiated from a lower Mode when Shutdown Cooling is **NOT** initially in service.

Comments / Reference: From SO23-14-9, FR-3 Bases	Revision # 25

FUNCTIONAL RECOVERY BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

4.0 BASES DESCRIPTION (Continued)

4.7 ATTACHMENT FR-3, RECOVERY - RCS INVENTORY CONTROL

The purpose of maintaining RCS Inventory Control, in conjunction with RCS Pressure Control, is to keep the core covered with an effective medium for the removal of decay heat. To do this, RCS inventory is maintained between the minimum volumes required to keep the core covered with an effective coolant medium and the maximum level desirable for operational purposes (i.e., to prevent solid plant operation with its associated pressure control problems). The Functional Recovery RCS Inventory Control attachment is divided into the following ***three*** main sections:

1. **Recovery Actions**

2. **Resource Assessment Charts**
   The two RACs provide graphical representation of plant equipment and information needed to fulfill the Inventory Control Safety Function via a given Success Path. The charts are used as aids in determining if a success path is available. They are:

   - RAC-IC-1: CVCS
   - RAC-IC-2: ECCS

3. **Success Path Actions**

   - Success Path Actions: IC-1, CVCS
   - Success Path Actions: IC-2, ECCS

4.7.1 Recovery Actions

.1 Step 1 Determine RCS Inventory Control Success Path available

The first step of the Recovery Actions provides direction for determining which Success Path should be used to recover the Inventory Control Safety Function. The Functional Recovery Safety Function Status Check or an optimal EOI normally directs entry into the recovery attachment. If a Success Path is already identified, then it is implemented. Unless a specific Success Path is identified for this Safety Function, the Resource Assessment Charts (RAC-IC-1 and RAC-IC-2) are used to determine the method(s) available to recover the Inventory Control Safety Function.

If no Success Paths are identified from the Functional Recovery SFSC, or the RACs, then SAMG initiation should be evaluated. The actual evaluation and initiation is done outside this EOI. Generally speaking, continued inability to establish control of this Safety Function, warrants SAMG initiation. In addition to this SAMG request, direction is provided to continue on with the remaining steps of the Recovery Actions.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		2
K/A #	G 2.2.6	
Importance Rating		3.6

Equipment Control: Knowledge of the process for making changes to procedures

Proposed Question: SRO 96

A Procedure Modification Permit which could potentially change the intent of an Operating Instruction is being prepared.

Which ONE (1) of the following activities is required for the Procedure Modification Permit?

A. Approval by the Shift Manager is required within 14 days of completion.

B. A 50.59 Safety Evaluation must be performed within 14 days of completion.

C. Approval by the Manager, Operations is required prior to implementation.

D. Approval by the Operations Procedure Group Supervisor must be performed prior to implementation.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because their approval may be required prior to implementation depending on the availability of the Ops Manager.
B. Incorrect. Plausible because a 50.59 is required, however, it is required based on the time the PMP will be used. See Step 3 of OP(123)28.
C. Correct. Per Step 4 of OP(123)28.
D. Incorrect. Plausible because they would review but only to determine if a REV or TCN is required.

Technical Reference(s): SO123-0-A3, Step 6.16; Form OP(123)28, Step 4 & Keypoint 13 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55169 — As an SRO, given a plant situation, DESCRIBE the administrative or technical specification requirements applicable to a situation not covered by Normal, Abnormal or Emergency Operating instructions and determine the required action.

Question Source: Bank # 128199

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam ____________

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

10 CFR Part 55 Content: 55.41 ____
55.43 3

Comments / Reference: From SO123-0-A3, Step 6.16	Revision # 8

NUCLEAR ORGANIZATION UNITS 1, 2 AND 3 | OPERATIONS DIVISION PROCEDURE REVISION 8 | SO123-0-A3 PAGE 25 OF 38

6.0 PROCEDURE (Continued)

6.16 Procedure Modification Permit (PMP), Form OP(123) 28

6.16.1 A PMP, Form OP(123) 28, should be used to resolve issues that will facilitate in-use or imminent use of a procedure for current plant conditions, when such conditions are different from those assumed by the original procedure, and are more extensive than CLARIFICATIONS.

6.16.2 When prompt Operator action is required to control plant abnormalities, and these actions result in modification to system alignments (not addressed by existing Operating instruction), then prepare a PMP using Form OP(123) 28, an Abnormal Evolution using Form OP(123) 29, or Status Control Form, Form OP(123) 1, per SO123-0-A4, as soon as practical.

6.16.3 Operation Division Forms are available in Portal or SAP GUI using the SCASE module and accessing Records >Record: Nuclear Controlled Forms. |

.1 In instances when the Portal or SAP GUI are unavailable, contact CDM for a copy of the form or to verify the form is the current version. |

6.16.4 PMPs shall be prepared, approved, and implemented using Form OP(123) 28, Keypoints-Procedure Modification Permit, as a guide.

.1 It is acceptable to use more than one permit form for a particular PMP. However, only one approval page is required and all pages shall be numbered and accounted for.

6.16.5 Initial Approval of the PMP shall be made by a member of PLANT MANAGEMENT STAFF-OPERATIONS and an SRO based upon an assessment of system and/or plant status and conditions. (LCS 5.0.103.1.3.b)

.1 After Initial Approval, then the PMP shall be attached to the affected STAND ALONE PROCEDURE/ATTACHMENT, and the listed procedure steps modified as described on the PMP.

6.16.6 If the preparer determines the PMP should be evaluated for incorporation into existing procedures, then a Notification should be initiated, and a copy of the PMP should be sent to the Operations Procedures Group Supervisor for evaluation as a TCN or revision.

6.16.7 After a PMP has received Initial Approval, then it shall not be duplicated for subsequent use unless specifically allowed by the PMP.

Comments / Reference: From OP(123)28, Step 4	Revision # 0

Reference: SO123-0-A3

PROCEDURE MODIFICATION PERMIT
CONTINUOUS USE

PERMIT #	UNIT	PROCEDURE NUMBER	REV/TCN	ATTACHMENT #	PMP PAGE
	☐ 2 ☐ 3 ☐ C				____ OF ____
SECTION/ATTACHMENT TITLE:					
NAME OF REQUESTER:					

		YES	NO
1.	Does this activity conflict with the Operating License, Technical Specifications, and/or LCS (Licensee Controlled Specifications)? If YES, then cancel activity, redraft PMP, or contact Licensing.	☐	☐
2.	Could implementation of this activity:	YES	NO
	a. Pose adverse ENVIRONMENTAL EFFECTS?	☐	☐
	b. Potentially impact the TOPICAL REPORT, SECURITY PLANS, EMERGENCY PLAN, FIRE PROTECTION PROGRAM (including UFHA), ODCM, IST, and/or ISI Program?	☐	☐
	c. Potentially impact TECHNICAL REQUIREMENTS?	☐	☐
	d. Potentially move fluid and/or gas in a radioactive system?	☐	☐
	If YES to a, b, c, and/or d, then obtain approval from the Cognizant Division Supervision, and attach justification if provided.		
3.	Check (✔) all items that apply to this PMP activity. ☐ Implements ALREADY APPROVED changes Enter identifiers and associated numbers:____________________ ☐ Involves MAINTENANCE ACTIVITIES ☐ Implements EDITORIAL CORRECTIONS (ECs) or MINOR PROCEDURE MODIFICATIONS ☐ Involves a procedure that is 50.59 DNA	YES	NA
	If there are parts of the PMP that are not entirely addressed by the items above, or if this supports a Maintenance Activity and the PMP will be in effect ≥90 days at power, then complete 10CFR50.59 screening prior to 90 days, and attach after this cover page. Enter AR Number: ____________ SCN Closed Date: ____________	☐	☐
4.	Does this PMP Change INTENT of the Operating Instruction by:	YES	NO
	a. Allowing performance of a new, unrelated function, or creates a new evolution or new flow path, not originally authorized?	☐	☐
	b. Requiring a new or additional 50.59 Evaluation?	☐	☐
	If YES to a and/or b, then OBTAIN approval from Manager, Operations prior to implementation. (TS D6.8.2 and LCS 5.0.103.12)		
5.	Does this activity involve DCS/ISFSI? If YES, then ensure review by a 72.48 Screener has been completed. Name: ____________________	YES ☐	NO ☐
6.	Does the proposed change affect Reactivity Management? If YES, or unknown, then complete Form OP(123) 91-1.	YES ☐	NO ☐

Comments / Reference: From OP(123)28, Keypoint 13	Revision # 0

KEYPOINTS - PROCEDURE MODIFICATION PERMIT (Continued)

13. If the permit changes the INTENT of the procedure, or requires a new or additional 10CFR50.59 Evaluation, or creates a new evolution or flow path, then the Manager, Operations shall approve the permit prior to implementation. (Keypoint 21)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G 2.2.25	
	Importance Rating		4.2

Equipment Control: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits

Proposed Question: SRO 97

Given the following BOC conditions:

- Unit 2 is operating at 100% power with a leaking Main Steam Safety valve on Steam Generator E088.
- There is one (1) Main Steam Safety Valve already gagged on Steam Generator E088.
- Preparations are being made to gag second leaking Main Steam Safety Valve on E088.

Which ONE (1) of the following describes the Technical Specification requirements that apply to gagging a second Main Steam Safety Valve on E088 and what is the basis for this requirement?

Reduce...

A. power which will allow the Safety Valves to remove the required amount of decay heat to prevent exceeding the Safety Limit of 1.03 for DNBR on a loss of the normal secondary heat removal path.

B. allowed power and High Power trip setpoints which ensures that the RCS Pressure Safety Limit of ≤ 2750 psia is NOT exceeded on a loss of the normal secondary heat removal path.

C. High Pressure trip setpoints which ensures that the peak centerline temperature of < 4960°F is NOT exceeded on a loss of the normal secondary heat removal path.

D. High Power trip setpoints which ensures that the Safety Limit of 1.03 for DNBR is NOT exceeded on a loss of the normal secondary heat removal path.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because it could be thought that limiting power would limit decay heat and subsequent RCS temperature post-event and protect against exceeding DNBR, however, the DNBR value is incorrect.

B. Correct. Limiting power and high power trip setpoints is required and in conjunction with the PZR Safety Valves protects against exceeding the RCS Pressure Safety Limit of ≤ 2750 psia.

C. Incorrect. Plausible because it could be thought that limiting power and High Pressure setpoints would limit decay heat and subsequent RCS temperature post-event and protect against exceeding Peak Centerline Temperature Safety Limit. The Peak Centerline Temperature listed reflects an EOC condition as PCT is reduced 58°F for every 10,000 MWD/MTU.

D. Incorrect. Plausible because it could be thought that limiting power would limit decay heat and subsequent RCS temperature post-event and protect against exceeding the DNBR safety limit, however, the DNBR value is incorrect.

Technical Reference(s): Technical Specification LCO 3.7.1; Technical Specification LCO 3.7.1, Table; Technical Specification LCO 3.7.1, Bases; Tech Spec Safety Limit 2.1.1 & 2.1.2, Bases

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 56649 — Given plant and equipment conditions, or Technical Specification/LCS surveillance results, DETERMINE system or equipment OPERABILITY and LCO(s) impacted along with all required actions and surveillances using SONGS procedures, Technical Specifications and Licensee Controlled Specifications (LCS).

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

Question History: Last NRC Exam ___

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

10 CFR Part 55 Content:
- 55.41 ___
- 55.43 2

Comments / Reference: From Technical Specification LCO 3.7.1	Amendment # 212

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two to seven required MSSVs per SG inoperable.	A.1 Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours
	<u>AND</u>	
	A.2 Reduce the Linear Power Level High trip setpoint in accordance with Table 3.7.1-1.	36 hours

Comments / Reference: From Technical Specification LCO 3.7.1, Table 3.7.1.1	Amendment # 212

Table 3.7.1-1 (page 1 of 1)
Maximum Allowable Power Level
versus Inoperable MSSVs

NUMBER OF INOPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)	MAXIMUM ALLOWABLE LINEAR POWER LEVEL HIGH TRIP (% RTP)
2	95	95
3	56	56
4	46	46
5 to 7	MODE 3	Not applicable

Comments / Reference: From Technical Specification LCO 3.7.1 Bases	Amendment # 127

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2; its purpose is to limit secondary system pressure to ≤ 110% of design pressure when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the MSSV relieving capacity, and thus RCS pressure, are those characterized as decreased heat removal events, and are presented in the UFSAR, Section 15.2 (Ref. 3). Of these, the full power loss of condenser vacuum (LOCV) event is the limiting AOO. An LOCV isolates the turbine and condenser, and terminates normal feedwater flow to the Steam Generators. Before delivery of auxiliary feedwater to the Steam Generators, RCS pressure reaches ≤ 2750 psig. This peak pressure is less than or equal to 110% of the design pressure of 2500 psia, but high enough to actuate the pressurizer safety valves. The maximum relieving rate of the MSSVs during the LOCV event (Ref. 3, Fig. 15.2-10), is within the rated capacity of the MSSVs.

The limiting accident for peak RCS pressure is the full power feedwater line break (FWLB), inside containment, with the failure of the backflow check valve in the feedwater line from the affected Steam Generator. Water from the affected Steam Generator is assumed to be lost through the break with minimal additional heat transfer from the RCS. With heat removal limited to the unaffected Steam Generator, the reduced heat transfer causes an increase in RCS temperature, and the resulting RCS fluid expansion causes an increase in pressure. The RCS pressure increases to ≤ 3000 psia (Ref. 3, Fig. 15.2-40), with the pressurizer safety valves providing relief capacity. The maximum relieving rate of the MSSVs during the Feedwater Line Break event (Ref. 3, Fig. 15.2-51), is within the rated capacity of the MSSVs.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

Comments / Reference: From Technical Specification LCO 3.7.1 Bases	Amendment # 127

LCO

This LCO requires all MSSVs to be OPERABLE in compliance with Reference 2, even though this is not a requirement of the DBA analysis. This is because operation with more than one MSSV inoperable per steam generator requires limitations on allowable THERMAL POWER (to meet Reference 2 requirements) and adjustment to Reactor Protection System trip setpoints. These limitations are according to those shown in Table 3.7.1-1, Required Action A.1, and Required Action A.2. An MSSV is considered inoperable if it fails to open upon demand.

The OPERABILITY of the MSSVs is defined as the ability to open in accordance with Lift Settings specified in Table 3.7.1-2, relieve Steam Generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the inservice testing program.

The Lift Settings specified in Table 3.7.1-2 correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety function to mitigate the consequences of accidents that could result in a challenge to the Reactor Coolant Pressure Boundary.

Comments / Reference: From Tech Spec Safety Limit 2.1.2 Bases	Amendment # 127

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence the valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Protective System (RPS) trip setpoints (LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation"), together with the settings of the MSSVs (LCO 3.7.1, "Main Steam Safety Valves (MSSVs)") and the pressurizer safety valves, provide pressure protection for normal operation and AOOs. In particular, the Pressurizer Pressure — High Trip setpoint is specifically set to provide protection against overpressurization (Ref. 5). Safety analyses for both the Pressure — High Trip and the RCS pressurizer safety valves are performed, using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Steam Bypass Control System;

b. Pressurizer Level Control System; or

c. Pressurizer Pressure Control System.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 2750 psia.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events.

Comments / Reference: From Tech Spec Safety Limit 2.1.1 Bases	Amendment # 207

SAFETY LIMITS

SL 2.1.1.1 and SL 2.1.1.2 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature remains below melting.

The minimum value of the DNBR during normal operation and design basis AOOs is limited to 1.31, based on a statistical combination of CE-1 CHF correlation and engineering factor uncertainties, and is established as an SL. Additional factors such as rod bow and spacer grid size and placement will determine the limiting safety system settings required to ensure that the SL is maintained.

A steady state peak linear heat rate of 21 KW/ft has been established as the Limiting Safety System Setting to prevent fuel centerline melting during normal steady state operation. Following design basis anticipated operational occurrences, the transient linear heat rate may exceed 21 KW/ft provided the fuel centerline melt temperature is not exceeded.

The design melting point of new fuel with no burnable poison is 5080°F. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the fuel. The 58°F per 10,000 MWD/MTU adjustment for burnup was accepted by the NRC in Topical Report CEN-386-P-A, Reference 3. Adjustments for burnable poisons are established based on NRC approved Topical Report CENPD-382-P-A, Reference 4.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		3
K/A #	G 2.3.4	
Importance Rating		3.7

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions

Proposed Question: SRO 98

Given the following conditions:

- An emergency event is in progress on Unit 2, and a SITE AREA EMERGENCY has been declared.
- The Emergency Coordinator duties have been turned over from the Station Emergency Director (SED) to the Corporate Emergency Director (CED).

In this situation, which ONE (1) of the following must authorize EMERGENCY RADIATION EXPOSURE exceeding 10CFR20 limits?

A. Operations Leader

B. Health Physics Leader

C. Station Emergency Director

D. Corporate Emergency Director

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Operations Leader has responsibilities associated with selecting a volunteer for emergency radiation exposure, however, this can only be authorized by the Station Emergency Director.
B. Incorrect. Plausible because an emergency radiation exposure is involved, however, this can only be approved by the Station Emergency Director.
C. Correct. Even though the Station Emergency Director has been relieved, they are still the individual responsible for this authorization.
D. Incorrect. Plausible because it could be thought that with turnover complete that responsibility now resides with the Corporate Emergency Director, however, that person would not be as familiar with plant conditions as the Station Emergency Director.

Technical Reference(s) SO123-VIII-10, Step 4.2.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55369	As an SRO, DETERMINE Emergency Coordinator actions required for notification, evacuation, and exposure control of onsite personnel, and for making notifications and Protective Action Recommendations to offsite agencies in accordance with SO123-VIII-10, Emergency Coordinator Duties.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 ______
55.43 4, 5

Comments / Reference: From SO123-VIII-10, Step 4.2.2	Revision # 26-1

3.0 PREREQUISITES

3.1 Emergency Planning is responsible for ensuring that the current copy of this document is in the emergency notebook for use during declared emergencies and drills.

3.2 Personnel are responsible for ensuring they use the current copy of this document when not in a declared emergency or drill by checking the electronic document management system or by use of one of the methods described in SO123-VI-0.9.

3.3 Verify level of use requirements on the first page of this document.

4.0 PRECAUTIONS

4.1 The EC should ensure the verbal notification to the Nuclear Regulatory Commission (NRC) is made within 20 minutes after declaration, and no later than one hour after declaration.

4.2 EC duties are normally performed by the Units 2/3 SM prior to turnover of the EC title to the SED, and ultimately to the Corporate Emergency Director (CED).

4.2.1 SM/EC may be relieved by other qualified ECs prior to TSC activation.

4.2.2 Prior to turnover of the EC title to the CED, only the EC (SM or SED) may authorize:

- Emergency Event Declaration/Classification
- Site Assembly and Site Evacuation
- Exceeding 10CFR20 Exposure Limits
- Notification to Offsite Agencies
- Offsite Protective Action Recommendations (PARs)

4.2.3 When the EC title is turned over to the CED the EC duties are split between the SED and the CED. Following turnover of the EC title to the CED,

.1 The SED retains the authority for:

- Emergency Event Declaration/Classification
- Site Assembly and Site Evacuation
- Exceeding 10 CFR 20 Exposure Limits

.2 The CED assumes the authority for:

- Notification to Offsite Agencies
- Offsite Protective Action Recommendations (PARs)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G 2.4.6	
	Importance Rating		4.7

Emergency Procedures/Plan: Knowledge of EOP mitigation strategies

Proposed Question: SRO 99

Given the following conditions:

- Unit 2 has tripped from 100% power due to a Loss of Offsite Power.
- A Pressurizer Safety Valve is partially stuck OPEN.
- Both Trains of SIAS have actuated and Safety Injection flow meets the Minimum Required SI Flowrates during Cold Leg Injection per SO23-12-11, EOI Supporting Attachments, Attachment 12.

Current conditions are as follows:

- Pressurizer pressure is 1025 psia.
- Core Exit Thermocouple temperature is 539°F.
- Pressurizer level is 80% and slowly rising.
- Containment temperature is 165°F.
- Reactor Coolant System subcooling is 10°F and lowering.

Which ONE (1) of the following describes the mitigation strategy in accordance with SO23-12-03, Loss of Coolant Accident?

A. Allow Pressurizer level to increase and cooldown the RCS per SO23-12-11 EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control.

B. Maintain RCS temperature constant and stabilize Pressurizer level per SO23-12-11 EOI Supporting Attachments, Floating Step 33, Monitor RCS Solid Operation.

C. Throttle HPSI flow to prevent a solid Pressurizer and cooldown the RCS per SO23-12-11 EOI Supporting Attachments, Floating Step 7, Verify SI Throttle/Stop Criteria.

D. Maintain RCS temperature constant while reducing Pressurizer level to less than 60% per SO23-12-11 EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed with subcooling < 20°F, allowing Pressurizer level to increase while cooling down is the desired action per Attachment 5, Core Exit Saturation Margin Control.

B. Incorrect. Plausible because solid Pressurizer conditions are imminent, however, cooldown must be performed due to lack of subcooling.

C. Incorrect. Plausible if thought that a solid Pressurizer was an undesirable condition, however, a lack of subcooling precludes SI Throttle/Stop.

D. Incorrect. Plausible because maintaining Pressurizer level less than 60% is guidance contained in Attachment 5 when Core Exit Saturation Margin is less than 20°F, however, a cooldown must be performed even if it results in a solid Pressurizer.

Technical Reference(s): SO23-12-3, Step 10; SO23-12-11, Attachment 5; SO23-12-11, Floating Steps 7 & 33 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 55279 / 54723

Per the EOI Attachments procedure, SO23-12-11, DESCRIBE: The basis for each step, caution or note.

As the SRO, DIRECT response to and recovery from a loss of coolant accident per SO23-12-3.

Question Source: Bank # 127481

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments / Reference: From SO23-12-3, Step 10	Revision # 20

NUCLEAR ORGANIZATION
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LOSS OF COOLANT ACCIDENT

OPERATOR ACTIONS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9 CONFIRM Leak Isolation:	
a. VERIFY rate of RCS inventory and pressure loss – less than available Charging Pump capacity.	a. GO TO step 15.
10 ESTABLISH RCS Inventory and Pressure Control:	
a. INITIATE indicated actions for available control methods of SO23-12-11, Attachment 5, CORE EXIT SATURATION MARGIN CONTROL.	
b. VERIFY RCS pressure – stable or rising AND – controlled.	b. GO TO step 15.
11 ESTABLISH RCS Heat Removal Control:	
a. VERIFY SBCS available: 1) Condenser Backpressure – less than SBCS Interlock Setpoint. AND 2) MSIVs – open.	a. OVERRIDE (as required) and OPERATE ADVs.
b. VERIFY MFW available: 1) MAINTAIN S/G levels – between 40% NR and 80% NR.	b. OPERATE AFW to establish at least one S/G level – between 40% NR and 80% NR.

Comments / Reference: From SO23-12-11, Attachment 5	Revision # 6

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ATTACHMENT 5

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EOI SUPPORTING ATTACHMENTS

CORE EXIT SATURATION MARGIN CONTROL

NOTE

During ESDE the value of PTS Subcooling (CFMS page 311) should be used in place of CESM.

CAUTION

Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS is the controlling attachment when: 1) the Natural Circulation cooldown strategy of minimizing Reactor Vessel Upper Head void formation is used, or 2) the EOIs are entered from a lower mode and Shutdown Cooling was NOT initially in service.

CORE EXIT SATURATION MARGIN (CESM)

CONTROL METHOD	**LOCA,SGTR,SBO,FR:** less than 20°F **ESDE,LOFW, LOOP/LOFC:** less than 80°F **LOWER MODE ENTRY:** less than 20°F	**OPTIMUM** **SBO:** between 20°F and 50°F **LOCA,SGTR,FR:** between 20°F and 160°F **ESDE,LOFW, LOOP/LOFC:** between 80°F and 160°F LOWER MODE ENTRY: greater than 20°F (No Upper Limit)	**SBO:** greater than 50°F **OTHER:** greater than 160°F LOWER MODE ENTRY: No Upper Limit
Feedwater Flowrate	RAISE, MAINTAIN S/G Levels – less than 80% NR.	STABILIZE S/G Level – between 40% and 80% NR.	LOWER, MAINTAIN S/G Levels – greater than 40% NR
S/G Steaming Rate	RAISE	MAINTAIN	LOWER
SI Flowrate	RAISE, ATTEMPT to maintain PZR level – less than 60%.	IF SI throttle/stop criteria (FS-7) – satisfied, THEN throttle flowrate to maintain PZR level – between 30% and 60% IF SIAS – reset, THEN ATTEMPT to place PLCS in AUTO.	IF SI throttle/stop criteria (FS-7) – satisfied, THEN lower flowrate and maintain PZR level – greater than 30%.
Charging Flowrate			
Letdown Flowrate	LOWER		RAISE
Normal Spray	LOWER, ATTEMPT to maintain PZR level – less than 60%.	MAINTAIN Saturation Margin as RCS temperatures are reduced.	RAISE, REQUEST SM/OL evaluate opening PZR Vents per *SO23-3-2.33, REACTOR COOLANT GAS VENT SYSTEM*.
Auxiliary Spray			
PZR Htrs.	If PZR level greater than 30%, ENSURE ON.		ENSURE OFF

Comments / Reference: From SO23-12-11, Floating Step 33	Revision # 6

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ATTACHMENT 2

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

FS-33 MONITOR RCS Solid Operation

Applicability: ❑ 12-3, ❑ 12-4, ❑ 12-5, ❑ 12-9

CAUTION

Water solid operation of the RCS should be avoided unless a minimum Core Exit Saturation Margin of 20°F can NOT be recovered by other means. If RCS water solid conditions are indicated, then changes in S/G feeding or steaming, SI flow, charging or letdown flow, sampling flow, CBO flow or other RCS drain paths should all be made slowly allowing time to monitor the effect of the action taken. Multiple or simultaneous changes should be avoided.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. VERIFY RCS is NOT water solid:	✿ IF RCS water solid conditions are indicated,
1) VERIFY liquid interface indicated:	THEN
a) PZR level – less than 100%	1) ENSURE RCS pressure within the limits of Attachment 29, POST-ACCIDENT PRESSURE / TEMPERATURE LIMITS.
OR	
b) Reactor Vessel level – less than 100% (Head) QSPDS page 622 CFMS page 311.	2) CONTROL RCS temperature.
	3) IF criteria of FS-7, VERIFY SI Throttle/Stop Criteria – satisfied, THEN CONTROL charging, letdown, and HPSI flow.
2) VERIFY exaggerated or severe pressure response associated with RCS inventory or temperature changes – NOT indicated.	4) NOTIFY Shift Manager/Operations Leader of possible RCS water solid condition.

Comments / Reference: From SO23-12-11, Floating Step 7	Revision # 6

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
FS-7 VERIFY SI Throttle/Stop Criteria	
Applicability: ALL	
a. VERIFY at least one S/G operating: 1) SBCS – available OR ADV – available. AND 2) Feedwater – available.	a. GO TO *SO23-12-9, FUNCTIONAL RECOVERY* AND INITIATE SO23-12-9, Attachment FR-5, RECOVERY – HEAT REMOVAL.
b. VERIFY PZR level – greater than 30% AND – NOT lowering. c. VERIFY Core Exit Saturation Margin – greater than or equal to 20°F: QSPDS page 611 CFMS page 311. d. VERIFY Reactor Vessel level – greater than or equal to 100% (Plenum): QSPDS page 622 CFMS page 312 Attachment 4.	o IF any criteria of steps b. through d. – NOT satisfied, THEN ▪ OPERATE Charging and SI systems as necessary to maintain Throttle/Stop criteria – satisfied. ▪ THROTTLE Loop Injection valves – as required. ▪ ENSURE auxiliaries to SI Pumps: a) Electrical power to pumps and valves. b) Proper system alignment. c) CCW flow. d) HVAC.

Comments / Reference: From SO23-12-3, Floating Step 7	Revision # 6

NUCLEAR ORGANIZATION
UNITS 2 AND 3

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EOI SUPPORTING ATTACHMENTS

FLOATING STEPS

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
FS-7 VERIFY SI Throttle/Stop Criteria (Continued)	
e. RCS Cooldown – NOT in progress.	e. MAINTAIN Boration – at least 40 GPM.
f. VERIFY SI Pumps – NOT operating per SO23-12-9, Attachment FR-1, RECOVERY – REACTIVITY CONTROL, to meet RC-3 Success Path.	f. GO to step h.
g. THROTTLE OR STOP SI Pumps as required – one train at a time.	
h. VERIFY Charging Pumps – NOT operating per SO23-12-9, Attachment FR-1, RECOVERY – REACTIVITY CONTROL, to meet RC-2 Success Path	h. GO to step k.
i. VERIFY PZR Level – less than 80%.	i. 1) INITIATE FS-31, ESTABLISH CVCS Letdown Flow. 2) INITIATE FS-33, MONITOR RCS Solid Operation.
j. STOP Charging Pumps as required one at a time.	
k. MAINTAIN criteria of steps a. through e. – satisfied.	
l. VERIFY Containment pressure – less than 3.4 PSIG.	l 1) ENSURE the following – actuated: SIAS CIAS CCAS CRIS. 2) GO TO next applicable floating step.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Group #		4
K/A #	G 2.4.9	
Importance Rating		4.2

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

Proposed Question: SRO 100

Given the following conditions:

- Unit 2 has tripped from 100% power.
- Reactor Trip Override is reset and Steam Generator levels are normal.
- Pressurizer level is 30% and slowly lowering.
- Pressurizer pressure is 2240 psia.
- Letdown is secured.
- Steam Bypass Control System (SBCS) is operating to maintain Steam Generator pressures at 1000 psia.
- Annunciator 50A01 - QUENCH TANK PRESS HI has come in and cleared.
- Annunciator 50A11 - QUENCH TANK LEVEL HI/LO is in alarm.
- Annunciator 50A21 - QUENCH TANK TEMP HI is in alarm.
- Annunciator 50A31 - PZR SAFETY VALVE OUTLET TEMP HI is in alarm.
- Containment pressure is 1.8 psia and rising.
- Containment temperature and humidity are rising.

Which ONE (1) of the following describes the mitigation strategy for the event in progress?

A. Commence controlled cooldown using the Steam Generators and Steam Bypass Control System per SO23-12-11, EOI Supporting Attachments, Attachment 3, Cooldown / Depressurization.

B. Initiate a manual Safety Injection to provide makeup water to the Reactor Coolant System and transition to SO23-12-9, Functional Recovery.

C. Commence cooldown at maximum achievable rate using Atmospheric Dump Valves per SO23-12-11, EOI Supporting Attachments, Attachment 5, Core Exit Saturation Margin Control.

D. Open Pressurizer Normal and Auxiliary Spray Valves to rapidly reduce pressure to less than 1400 psia per SO23-12-1, Standard Post Trip Actions.

Proposed Answer: A

Explanation:

A. Correct. Using information gleaned from Annunciator Responses, it can be determined that the rupture disc on the Quench Tank has blown. This allowed the high pressure condition on the Quench Tank [50A51 - See Note (1)] to clear while continuing a Small Break LOCA out the Pressurizer Safety Valves. In this condition a controlled cooldown is desired to maintain Pressurizer level and minimize voiding of the Reactor Coolant System.

B. Incorrect. Plausible if thought that a manual Safety Injection at this time was going to provide makeup and the starting of all Charging Pumps would do this via the SI, however, in this condition a control cooldown is required via the Steam Generators and SBCS. If a high radiation condition existed inside Containment then this would be an appropriate response, however, it is only addressed in the Functional Response Procedure.

C. Incorrect. Plausible because it could be thought that the faster the RCS was cooled down the quicker HPSI flow would be introduced, however, RCS voiding should be minimized and Pressurizer level control should be maintained.

D. Incorrect. Plausible because using the Pressurizer Normal and Auxiliary Spray Valves to reduce pressure is an approved method, however, voiding could occur and there is no guarantee that HPSI flow would be adequate for the break size.

Technical Reference(s) SO23-14-3, Section 3.0 / SO23-15-50-A.1, 50A01 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: 54723 — As the SRO, DIRECT response to and recovery from a loss of coolant accident per SO23-12-3.

Question Source: Bank # 126631
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History: Last NRC Exam SONGS 2003

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments / Reference: From SO23-14-3, Section 3.0	Revision # 8

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ATTACHMENT 1

LOSS OF COOLANT ACCIDENT BASES AND DEVIATIONS JUSTIFICATION

EOI STEP BASES

3.0 RECOVERY TECHNIQUE

Small Break LOCAs

RCS inventory is initially lost when the break flow rate exceeds the available Charging Pump capacity. For small breaks, RCS Inventory Control is regained via injection from the Charging Pumps, High Pressure Safety Injection (HPSI) Pumps, and control of RCS cooldown rate.

For small breaks, the RCS will initially depressurize to HPSI shutoff head. Pressure control is regained when the Safety Injection System (SIS) refills the RCS and PZR level is regained. Once pressure control is regained, subsequent small break post-LOCA operator actions which are associated with pressure control are: (1) decrease RCS pressure with Pressurizer (PZR) Spray, (2) control RCS level using HPSI and Charging Pumps, (3) heat removal via the S/Gs in order to establish SDC entry conditions, and (4) isolating or depressurizing the Safety Injection Tanks (SITs). For small break LOCAs, during the period of time the RCS is refilling, and pressure control has not yet been achieved, there may be voiding in the RCS. The voided areas may be located in the Reactor Vessel Head region, the RCS loops, or the S/G tubes.

There are two paths initially available for RCS Heat Removal: Heat transfer to the secondary side via the S/Gs and heat transfer via the fluid flowing out the break. For small break LOCAs, S/G heat removal is required since the heat removal due to break flow is inadequate.

If PZR pressure decreases to less than 1430 PSIA, then two RCPs must be tripped. The strategy of leaving two RCPs running will facilitate RCS depressurization when no charging pumps are available and insufficient RCS leakage exists to allow HPSI injection to restore RCS inventory control. This is a change to the previous approach that secured all four RCPs because analysis indicated that prolonged four RCP operations during a LOCA could increase the severity of the event.

Core heat removal is maintained by forced or natural circulation. The operator must maintain feedwater (main or auxiliary) to the S/Gs and control steam flow via the Steam Bypass Control System (SBCS) or Atmospheric Dump Valves (ADVs).

Once RCS pressure and temperature are reduced to SDC entry conditions RCS heat removal is provided by the Shutdown Cooling System.

Comments / Reference: From SO23-15-50-A.1, 50A01	Revision # 8

50A01 QUENCH TANK PRESS HI

APPLICABILITY	PRIORITY	REFLASH	ASSOCIATED WINDOWS
Modes 1-4	AMBER	N/A	50A31, 50A11

INITIATING DEVICE	NOUN NAME	SETPOINT	VALIDATION INSTRUMENT	PCS ID	LINK # U2/U3
2(3)PSH0116	Quench Tank Pressure Switch High	24 PSIG [1]	NONE	P116	609/631

1.0 **REQUIRED ACTIONS**:

1.1 Proceed to Section 2.0.

2.0 CORRECTIVE ACTIONS:

SPECIFIC CAUSES	SPECIFIC CORRECTIVE ACTIONS
2.1 2(3)PCV-5437, Quench Tank N2 Regulator, failed	2.1 Cycle 2(3)HV-9100, Quench Tank to Waste Gas Header Isolation Valve, as required, to lower Quench Tank pressure to the normal range of 3 to 20 psig. 2.1.1 If venting becomes difficult, then drain the Waste Gas Collection Header per SO23-8-14, Attachment for Draining the Waste Gas Header. 2.1.2 If 2(3)PCV-5437, Quench Tank N2 Regulator, is failed, and Containment is accessible, then perform the following: .1 Isolate 2(3)PCV-5437, Quench Tank N2 Regulator .2 Manually control Quench Tank pressure.

[1] Pressures exceeding 25 psig may distort the Quench Tank Rupture Disk.