Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003 K2.02	
Importance Rating	2.5	

Reactor Coolant Pump System: Knowledge of bus power supplies to the following: CCW pumps

Proposed Question: Common 1

Given the following condition:

- Unit 1 is operating at 100%.
- Routine equipment rotations are in progress.
- Current equipment in service is as follows:
  - Station Service Water Pumps 1-01 and 1-02.
  - Centrifugal Charging Pump 1-01.
  - Component Cooling Water Pump 1-02.
- An 86-1 lockout occurs on Safeguards Bus 1EA2.
- All automatic actions occur.

What is the status of the cooling water to the Reactor Coolant Pumps?

A. Seal Injection has been lost and Thermal Barrier cooling has been continuously maintained.

B. Seal Injection has been lost and Thermal Barrier cooling was momentarily lost and has been restored.

C. Seal Injection has been maintained and Thermal Barrier cooling has been continuously maintained.

D. Seal Injection has been maintained and Thermal Barrier cooling was momentarily lost and has been restored.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if believed that CCP 1-01 is powered from 1EA2 and CCWP 1-02 is powered from 1EA1.
B. Incorrect. Plausible if believed that CCP 1-01 is powered from 1EA2 and the CCW response is correct.
C. Incorrect. Plausible as the Seal Injection response is correct and if believed that CCWP 1-02 is powered from 1EA1.
D. Correct. CCP 1-01 is continuously powered from 1EA1 and thus Seal Injection is maintained throughout the event. CCWP 1-02 loses power with the lockout on 1EA2, however, the standby pump CCWP 1-01 will auto start and Thermal Barrier cooling will be automatically restored.

Technical Reference(s): LO21.SYS.CC1.LN, Page 14; LO21.SYS.CS1.LN, Pages 34 & 36; 1EA1 & 1EA2 Power Supply Lists — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Component Cooling Water 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 3, 7
55.43 ____

Comments / Reference: From LO21.SYS.CC1.LN, Page 14	Revision # 2/28/12

CCW PUMPS

The CCW pumps are located on the centerline of the Auxiliary Building, elevation 810'. They are 100% capacity, centrifugal, horizontal, double suction, single stage, motor-driven pumps with a nominal capacity of 14,700 gpm each at a head of 226 ft. The shafts have minimum leakage mechanical seals cooled by the discharge of the pump. The journal and thrust bearings are self-lubricated by oil rings.

The pumps are normally powered from uEA1 and uEA2. On a loss of power, they will be supplied from the train related emergency diesel generator. Control power for the pumps is from uED1-2 for Train A and uED2-2 for Train B.

Each pump is equipped with local, direct reading suction and discharge pressure gauges. There is also a pressure transmitter on the discharge of the pump that provides indication in the control room and inputs to the plant computer.

The pumps are controlled from the control room on u-CB-03. After control is transferred, the pumps may be controlled at the Hot Shutdown Panel (HSP). (The HSP is also referred to as the Remote Shutdown Panel (RSP.) When control is transferred to the HSP, an alarm is generated in the control room and all automatic starts are bypassed.

The CCW pump will receive an automatic start signal from:

- Safety Injection Sequencer
- Blackout Sequencer
- Low discharge pressure on the running train of CCW
- An AUTO start of the associated train SSW pump on low pressure in the alternate SSW train.

On the start of a CCW pump, the train related SSW pump and the train associated safety chiller receive a start signal. The room cooler for the CCW pump will also start.

Comments / Reference: From 1EA1 Power Supply List	Revision # N/A

1EA1

E1-0004 sh -

CUB 1	1EA1-2, ALT FEEDER BKR, E1-0031 shs 03 & 04
CUB 2	1EA1-3, ALTERNATE POWER DG INPUT FOR OUTAGES, E1-0031 sh 63
CUB 3	CP1-CCAPCC-01, CCW PUMP 1-01, E1-0031 shs 25, 25A, & 26
CUB 4	CPX-CHCICE-01, VENT CHILLER 1-01, E1-0031 shs 57 & 58
CUB 5	CP1-AFAPMD-01, MDAFW PUMP 1-01, E1-0031 shs 37 & 38
CUB 6	CP1-CTAPCS-03, CNTMT SPRAY PUMP 1-03, E1-0031 shs 31 & 32
CUB 7	CP1-MEDGEE-01, DIESEL GEN 1-01, E1-0031 shs 21, 21A, & 22
CUB 8	CP1-CTAPCS-01, CNTMT SPRAY PUMP 1-01, E1-0031 shs 29 & 30
CUB 9	TBX-RHAPRH-01, RHR PUMP 1-01, E1-0031 shs 49 & 50
CUB 10	TBX-SIAPSI-01, SI PUMP 1-01, E1-0031 shs 45 & 46
CUB 11	TBX-CSAPCH-01, CENTRIF CHARGING PUMP 1-01, E1-0031 sh 53 & 54

Comments / Reference: From LO21.SYS.CS1.LN, Page 34	Revision # 04/28/11

CENTRIFUGAL CHARGING PUMPS

Centrifugal Charging Pumps u-01 and u-02 take separate suctions off of the common charging pump suction piping. These pumps are powered by 600 horsepower, 1800 rpm motors supplied from 6.9 kV engineered safeguards buses uEA1 and uEA2. Buses uED1 and uED2 provide 125 VDC power to the breaker control circuits of Centrifugal Charging Pumps u-01 and u-02, respectively. Each pump is rated at 150 gpm at a differential pressure of approximately 2590 psid across the pump. Each pump is rated for a maximum rated flow of 550 gpm at approximately 625 psid across the pump.

Comments / Reference: From LO21.SYS.CS1.LN, Page 36	Revision # 04/28/11

The following signals will automatically start CCP u-01 when aligned for control from the control room:

- Train A Safety Injection signal in conjunction with a Train A Safety Injection Sequencer start signal
- Train A Blackout Sequencer start signal

The following signals will automatically trip CCP u-01:

- Undervoltage on uEA1
- 86M lockout on the motor, which can be initiated by motor phase to phase or phase to ground overcurrent

Comments / Reference: From 1EA2 Power Supply List	Revision # N/A

1EA2

E1-0004 sh -

CUB 1 CP1-EPTRET-04, T1EB4, E1-0005 sh A, E1-0031 sh 19

CUB 2 1EA2-1, PWR FEED FROM XST2, E1-0001

CUB 3 CP1-EPTRET-02, T1EB2, E1-0005 sh A, E1-0031 sh 17

CUB 4 BT-1EA2, 1EA2 INNER BUS TIE BRKR, E1-0031 sh 11

CUB 5 1EA2 BUS INSTRMENTATION CUBICLE

CUB 6 CP1-SWAPSW-02, SSWP 1-02, E1-0031 sh 43

CUB 7 CPX-CHCICE-04, VENT CHILLER X-04, E1-0031 sh 59

CUB 8 TBX-RHAPRH-02, RHRP 1-02, E1-0031 sh 51

CUB 9 TBX-SIAPSI-02, SIP 1-02, E1-0031 sh 47

CUB 10 CP1-CTAPCS-02, CNTMT SPRAY PUMP 1-02, E1-0031 sh 33

CUB 11 CP1-CTAPCS-04, CNTMT SPRAY PUMP 1-04, E1-0031 sh 35

CUB 12 TBX-CSAPCH-02, CCP 1-02, E1-0031 sh 55

CUB 13 CP1-AFAPMD-02, AFWP 1-02, E1-0031 sh 39

CUB 14 CP1-CCAPCC-02, CCWP 1-02, E1-0031 sh 27

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	004 K2.06	
Importance Rating	2.6	

Chemical and Volume Control System: Knowledge of bus power supplies to the following: Control instrumentation

Proposed Question: Common 2

What is the normal power supply to the solenoid valves for the Reactor Makeup System Flow Control Valves?

A. Safety related 125 VDC.

B. Non-safety related 125 VDC.

C. Safety related 118 VAC.

D. Non-safety related 118 VAC.

Proposed Answer: A

Explanation:

A. Correct. These solenoid operated valves are all powered from 125 VDC Bus uED1-1.
B. Incorrect. Plausible because these valves are powered from DC, however, it is a safety-related power supply.
C. Incorrect. Plausible because the power supply is from a safety-related bus, however, it is supplied from DC power not AC.
D. Incorrect. Plausible if thought that these valves were associated with an AC power supply, however, they are DC powered solenoid operated valves.

Technical Reference(s) LO21.SYS.CS2.LN, Pages 10, 12, 17, & 18 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Chemical and Volume Control System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank # X
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 LO21.SYS.CS2.LN, Page 10	Revision # 04/28/11

RMUW to CVCS Boric Acid Blender Flow Control Valve

Flow control valve u-FCV-0111A is a fail closed, 2", air-operated globe valve with a valve positioner. The function of this valve is to regulate the flow of reactor-grade dilution water corresponding to the desired reactivity addition rate and to regulate the flow of dilution water to achieve the desired blended makeup boric acid concentration. The valve accomplishes these functions in the following manner.

Instrument air operates the valve through two, in-series, two position, three port solenoid valves, solenoid valve 1 (SV1) and solenoid valve 2 (SV2). Both solenoids are powered from uED1-1 (Figure 10).

Comments / Reference: From LO21.SYS.CS2.LN, Page 12	Revision # 04/28/11

RCS Makeup to VCT Isolation Valve

u-FCV-0111B is a fail closed, 2", air-operated globe valve. The function of this valve is to direct the flow of reactor-grade dilution water to the inlet piping of the VCT. A single solenoid must energize to align instrument air pressure to open the valve. Power to the solenoid is from uED1-1.

u-FCV-0111B is controlled from a three position (CLOSE, AUTO, OPEN) maintained handswitch, 1/u-FCV-0111B, located on CB-06. In CLOSE, the solenoid is de-energized which vents the diaphragm allowing the valve to fail closed. In AUTO, the solenoid is permitted to energize based on input from the Reactor Makeup System controls. In OPEN, the solenoid is energized and air pressure is aligned to fully open the valve.

These valves are located on the 832' elev. of the Safeguards Bldg. in the VCT valve room.

Comments / Reference: From LO21.SYS.CS2.LN, Page 17	Revision # 04/28/11

Boric Acid to Boric Acid Blender Flow Control Valve

Flow control valve u-FCV-0110A is a fail open, 2", air-operated, globe valve with a valve positioner. The function of this valve is to adjust the flow of boric acid to control the desired rate of reactivity addition or the resultant boric acid blended flow concentration.

Instrument air operates the valve through two in-series, two position, three port solenoid valves, solenoid valve 1 (SV1) and solenoid valve 2 (SV2). Both solenoids are powered from uED1-1 (Figure 10).

Comments / Reference: From LO21.SYS.CS2.LN, Page 18	Revision # 04/28/11

RCS Makeup to Charging Pump Flow Control Valve

u-FCV-0110B is a fail closed, 2", air-operated, globe valve. The function of this valve is to direct the flow of concentrated boric acid, reactor-grade dilution water, or blended flow to the outlet piping of the VCT. A single solenoid must energize to align instrument air pressure to open the valve. Power to the solenoid is from uED1-1.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005 A2.03	
	Importance Rating	2.9	

Residual Heat Removal System: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR pump/motor malfunction

Proposed Question: Common 3

Given the following conditions:

- Unit 2 is in a Mid-Loop condition when the following alarms are received:
  - 2-ALB-4B, Window 2.4 – RHRP 1/2 OVRLOAD TRIP.
  - 2-ALB-4B, Window 4.4 – RHRP 1/2 TO CL INJ FLO LO.
- The running Train A Residual Heat Removal (RHR) Pump has tripped.
- The standby Train B RHR Pump will NOT start.
- The Reactor Vessel Head is removed.

Which of the following actions should be performed per ABN-104, Residual Heat Removal System Malfunction?

Initially attempt to...

A. ...align the Refueling Water Storage Tank for gravity feed.

B. ... initiate Hot Leg injection with a Safety Injection Pump.

C. ...initiate Cold Leg injection with a Centrifugal Charging Pump.

D. ...align the Volume Control Tank for gravity feed.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because RWST gravity feed is a subsequent action if Hot Leg and Cold Leg injection fail.
B. Correct. Given the conditions listed, this is the correct action per ABN-104.
C. Incorrect. Plausible because Cold Leg injection is a subsequent action if Hot Leg injection fails.
D. Incorrect. Plausible because VCT gravity feed is a subsequent action if Hot Leg and Cold Leg injection fail.

Technical Reference(s) ABN-104, Section 8.1
ABN-104, Steps 8.3.9, 8.3.10, & 8.3.11

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Residual Heat Removal System including interrelations with other systems to include interlocks and control loops.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:	55.41	7, 10
	55.43	

Comments / Reference: From ABN-104, Section 8.1	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 39 OF 102

8.0 MODE 5 OR 6 COMPLETE LOSS OF DECAY HEAT REMOVAL CAPABILITY - RCS NOT FILLED

8.1 Symptoms

a. Annunciator Alarms

- "RHRP 1/2 OVRLOAD TRIP" (4B-2.4)
- "RHRP 1/2 RWST & HL SUCT XTIED" (4B-2.5)
- "RHRP 1/2 TO CL INJ FLO LO" (4B-4.4)
- "RCS MARGIN TO SAT LO" (5C-4.5)

b. Plant Indications

- Complete loss of RHR capability (both trains unavailable)
- Complete loss of Component Cooling Water
- RCS level below the required level for RHR flow needed for decay heat removal.

8.2 Automatic Actions

- None

Comments / Reference: From ABN-104, Step 8.3.9	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 41 OF 102

8.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Without an adequate hot leg vent path, the core exit will pressurize when saturation conditions in the RCS are reached. If a cold leg opening exists, or if a cold leg nozzle dam fails (if installed) rapid core uncovery may occur (within 10 minutes under the most limiting conditions).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ [C] 7 Verify Hot Leg Vent Path.	IF any cold leg opening exists OR any RCS temporary seal installed, THEN ensure adequate Hot Leg Vent Path per IPO-010A/B.
☐ 8 OPEN BOTH pressurizer PORV block valves AND OPEN BOTH pressurizer PORVs	

CAUTION: Racking in the SIP Breaker will place the unit into the Action Statement of TS 3.4.12.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[C] 9 Initiate Hot Leg Injection.	IF Hot Leg Injection can NOT be initiated, THEN GO TO Step 10.
☐ a. Locally rack in Affected Unit Safety Injection Pump breaker(s). • 1EA1/10/BKR, SAFETY INJECTION PUMP 1-01 MOTOR BREAKER (SFGD 810 Rm 1-083) • 1EA2/9/BKR, SAFETY INJECTION PUMP 1-02 MOTOR BREAKER (SFGD 852 Rm 1-103) • 2EA1/8/BKR, SAFETY INJECTION PUMP 2-01 MOTOR BREAKER (SFGD 810 Rm 2-083) • 2EA2/9/BKR, SAFETY INJECTION PUMP 2-02 MOTOR BREAKER (SFGD 852 Rm 2-103)	
☐ b. Verify available Safety Injection Train(s) - ALIGNED PER IPO-010A/B, ATTACHMENT 1.	b. Align safety injection per IPO-010A/B, Attachment 1 EXCEPT do NOT rack out SIP breakers.

Comments / Reference: From ABN-104, Step 8.3.9	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 42 OF 102

8.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: The use of SIP hot leg injection can cause an increase in RCS pressure and subsequent failure of temporary seals, if the temporary seal maximum allowable pressure is exceeded.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9 ☐ c. Verify NO opening exists in any RCS loop.	c. Ensure SIP hot leg injection valve that supplies loop with opening - CLOSED • 1/u-8802A, SI TO HL 2 & 3 INJ ISOL VLV • 1/u-8802B, SI TO HL 1 & 4 INJ ISOL VLV

NOTE: The preferred SIP to start is for the train that supplies loop with NO opening. Crossconnecting SIP discharge may be necessary if the preferred SIP is NOT available.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ d. Start the selected SI Pump.	
☐ e. Open the associated hot leg injection valve: • 1/u-8802A, SI TO HL 2 & 3 INJ ISOL VLV • 1/u-8802B, SI TO HL 1 & 4 INJ ISOL VLV	
☐ f. Verify Hot Leg Injection Flow. • u-FI-918, SIP 1 DISCH FLO • u-FI-922, SIP 2 DISCH FLO	f. Perform following 1. Check valve alignment to ensure flow path to Hot Legs exist. 2. IF Hot Leg Injection cannot be established, THEN GO TO Step 10 to Initiate Cold Leg Injection.

Comments / Reference: From ABN-104, Step 8.3.10	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 43 OF 102

8.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Racking in the CCP breaker may place the unit in violation of TS 3.5.3.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10 Initiate Cold Leg Injection.	IF Cold Leg Injection OR RCS Makeup can NOT be started, THEN GO TO Step 11.
☐ a. Locally rack in Affected Unit Centrifugal Charging Pump breaker(s): • 1EA1/11/BKR, 1APCH1, CENTRIFUGAL CHARGING PUMP 1-01 MOTOR BREAKER (SFGD 810 Rm 1-083) • 1EA2/12/BKR, 1APCH2, CENTRIFUGAL CHARGING PUMP 1-02 MOTOR BREAKER (SFGD 852 Rm 1-103) • 2EA1/7/BKR, 2APCH1, CENTRIFUGAL CHARGING PUMP 2-01 MOTOR BKR (SFGD 810 Rm 2-083) • 2EA2/6/BKR, 2APCH2, CENTRIFUGAL CHARGING PUMP 2-02 MOTOR BKR (SFGD 852 Rm 2-103)	a. IF the positive displacement pump is available, THEN rack in affected unit PDP breaker. • 1EB1/2B/BKR, 1APPD, POSITIVE DISPLACEMENT PUMP 1-01 MOTOR BREAKER (SFGD 810 Rm 1-083) • 2EB1/2B/BKR, 2APPD, POSITIVE DISPLACEMENT CHARGING PUMP 2-01 MOTOR BREAKER 2EB1/2B (SFGD 810 Rm 2-083)
[C] ☐ b. Verify the RWST suction isolation valves - OPEN: • 1/u-LCV-112D, RWST TO CHRG PMP SUCT VLV • 1/u-LCV-112E, RWST TO CHRG PMP SUCT VLV	b. Open valves as necessary.
[C] ☐ c. Verify VCT suction isolation valves - CLOSED: • 1/u-LCV-112B, VCT TO CHRG PMP SUCT VLV • 1/u-LCV-112C, VCT TO CHRG PMP SUCT VLV	c. Close valves as necessary.

Comments / Reference: From ABN-104, Step 8.3.11	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-104
RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 44 OF 102

8.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10 [C] ☐ d. Verify u-ZL-8220 AND U-ZL-8221, CHRG PMP SUCT HI POINT VENT VLV - CLOSED.	
[C] ☐ e. Ensure 1/u-8202A AND 1/u-8202B, VENT VLV - CLOSED.	
☐ f. Align ONE of the following charging flow paths, as necessary to bypass any known open RCS loop penetrations: • Loop 4 Cold Leg Charging • Loop 1 Cold Leg Charging • Centrifugal Charging Pump High Head Injection - Loops 1 - 4 Cold Leg	
☐ g. Start the selected charging pump.	
☐ h. Adjust charging flow per Attachment 2.	
☐ i. GO TO Step 12.	

CAUTION: If temporary seals are installed, passive injection may not be available due to RCS pressurization above available VCT pressure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ [C] 11 Align the RWST OR VCT to allow gravity feed per Attachment 9.	IF passive injection can NOT be established, THEN align the SI accumulator(s) for injection per Attachment 10.

Comments / Reference: From CPNPP Exam Bank	Revision # 10/06/97

Given the following conditions:

- During Mid-Loop operations on unit 2, the following alarms are received:
  - 2-ALB-4B, Window 2.4 – RHRP 1/2 OVRLOAD TRIP.
  - 2-ALB-4B, Window 4.4 – RHRP 1/2 TO CL INJ FLO LO.
- The Reactor Operator determines the running RHR Pump (Train A) has tripped and is unable to start the Train B RHR Pump.

Which of the following actions should be performed per ABN-104, Residual Heat Removal System Malfunction?

A. Isolate the RHR Hot Leg suctions, and initiate Hot Leg Injection with an SIP.

B. Ensure a Hot Leg vent path, and initiate Hot Leg injection with an SIP.

C. Open PORV's, initiate Cold Leg injection with an SIP.

D. Open PORV's, initiate Hot Leg injection with a CCP.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	005 K6.03	
Importance Rating	2.5	

Residual Heat Removal System: Knowledge of the effect that a loss or malfunction of the following will have on the RHRS: RHR heat exchanger

Proposed Question: Common 4

During a Design Basis Accident, which of the following would prevent the Residual Heat Removal Heat Exchanger 1-01 from performing its design function?

A. A loss of air to 1-HCV-0606, RHR HX 1-01 FLO CTRL VLV.

B. Closing 1-HS-4572, RHR HX 1 CCW RET VLV.

C. Closing 1-HCV-0128, U1 RHR LTDN FLO CTRL VLV.

D. A loss of air to 1-FCV-618, RHR HX 1-01 FLO CTRL VLV.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that the RHR Heat Exchanger Flow Control Valve will close, however, this valve fails open on a loss of air.
B. Correct. Isolating Component Cooling Water flow from the RHR Heat Exchanger prevents it from performing its design function.
C. Incorrect. Plausible because closing this valve would alter RHR flow, however, it would not impact the function of the RHR Heat Exchanger.
D. Incorrect. Plausible if thought that the RHR Heat Exchanger Bypass Flow Control Valve will open, however, this valve fails closed on a loss of air.

Technical Reference(s): ABN-301, Attachments 1 & 3; LO21.SYS.RH1.LN, Page 12 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Residual Heat Removal System including interrelations with other systems to include interlocks and control loops.

Question Source:

Bank #	X	
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 4, 7
55.43 ______

Comments / Reference: From ABN-301, Attachments 2	Revision # 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-301
INSTRUMENT AIR SYSTEM MALFUNCTION	REVISION NO. 12	PAGE 79 OF 122

ATTACHMENT 2
PAGE 27 OF 47

AIR OPERATED EQUIPMENT FAILURE POSITIONS

BLDG	EQUIPMENT		FAILURE POSITION
Rm u-067, TRN A ECCS VLV RM			
SFGD 790	u-HCV-0606	RHR HX u-01 FLO CTRL VLV	F.O.
SFGD 790	u-FCV-0618	RHR HX u-01 BYP FLO CTRL VLV	F.C.

Comments / Reference: From LO21.SYS.RH1.LN, Page 22	Revision # 08/25/04

RESIDUAL HEAT REMOVAL HEAT EXCHANGERS

The RHR Heat Exchangers are shell and U-Tube type heat exchangers. RHR System flow passes through the tubes while Component Cooling Water flows around the outside of the tubes. Each heat exchanger is designed to provide one-half of the capacity as necessary to meet the design cooldown requirements. After the cooldown is complete, one heat exchanger is sufficient to maintain the RCS temperature at 140°F. Each heat exchanger is arranged vertically and shares a room with the Containment Spray Heat Exchanger on the 790' level of the Safeguards Bldg.

Component Cooling Water to the heat exchanger is normally isolated. CCW is restored to the heat exchanger when the RHR System is placed into service. CCW is automatically restored to the heat exchanger in the event a Safety Injection or Containment Spray signal is actuated.

Comments / Reference: From LO21.SYS.RH1.LN, Page 29	Revision # 08/25/04

RHR SYSTEM LETDOWN CONTROL VALVE (u-HCV-0128)

HCV-0128 is an air operated valve located downstream of u-TCV-0381B in the Letdown Heat Exchanger Valve Room. The RHR Letdown Control Valve:

- allows purification of the Reactor Coolant System during shutdown conditions when the temperature of the coolant is maintained by residual heat removal.
- allows for additional letdown flow capacity during heatup of the Reactor Coolant System.
- allows additional letdown flow when establishing a bubble in the Pressurizer.
- is open during solid plant operation to allow u-PCV-0131 to control Reactor Coolant System pressure.

u-HCV-0128 is closed and locally isolated during normal power operations.

A Hagan hand controller, u-HC-0128, mounted on Main Control Board CB-06 is adjusted to position u-HCV-0128. The valve receives a full open signal when the hand controller is set to 100% and a full close signal when the controller is set to 0%. There are no automatic open or close signals to this valve and it will fail closed on loss of air or control power.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006 A1.16	
	Importance Rating	4.1	

Emergency Core Cooling System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: RCS temperature, including superheat, saturation and subcooled

Proposed Question: Common 5

Given the following conditions:

- Unit 1 is responding to a Small Break Loss of Coolant Accident per EOP-1.0A, Loss of Reactor or Secondary Coolant.
- Reactor Coolant System (RCS) subcooling is 19°F and is very slowly becoming less subcooled.
- RCS pressure is 1435 psig and very slowly lowering.
- Core Exit Thermocouple temperature is 573°F and stable.
- All Reactor Coolant Pumps are stopped.
- Pressurizer Relief Tank level and pressure are slowly rising.
- Engineering Safeguards Features Actuation signals are RESET.
- Residual Heat Removal Pumps are in STANDBY.
- All Reactor Vessel Level Indication System lights are LIT.
- All Steam Generator levels are between 50% and 60% with Auxiliary Feedwater flow secured.
- No Steam Generators have indication of either a fault or a rupture.
- Core Cooling Critical Safety Function Status Tree is YELLOW.

Which of the following actions may restore the Core Cooling Critical Safety Function Status tree to GREEN status?

A. Manual closure of a PORV Block Valve.

B. Termination of Safety Injection flow.

C. Placing Residual Heat Removal in service for cooldown.

D. Starting a Reactor Coolant Pump to initiate forced flow.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, closing the PORV Block Valve should restore subcooling and allow exiting FRC-0.3A, Response to Saturated Core Cooling.

B. Incorrect. Plausible because minimal Safety Injection Pump flow exists at the listed pressure, however, conditions are not met to terminate Safety Injection.

C. Incorrect. Plausible because placing Residual Heat Removal (RHR) in service would assist in restoring subcooling if RCS pressure were lower, however, at these plant conditions placing RHR in service would not have a beneficial effect.

D. Incorrect. Plausible if thought this would improve subcooling, however, the conditions for tripping the RCPs has been met in EOP-1.0A

Technical Reference(s): FRC-0.3A, Steps 2, 3, & 4; FRC-0.3A, CSFST; EOP-1.0A, Attachment 1.A — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the symptoms or Entry Conditions for FRC-0.3, Response to Saturated Core Cooling.

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 FRC-0.3A, Step 4	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.3A
RESPONSE TO SATURATED CORE COOLING	REVISION NO. 8	PAGE 4 OF 11

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Check RCS Vent Paths:	
	a. Power to PRZR PORV block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. PRZR PORVs - CLOSED	b. Manually close PRZR PORV(s). IF any valve can NOT be closed, THEN manually close its block valve.
	c. Block valves - AT LEAST ONE OPEN	c. Manually open block valve unless it was closed to isolate an open PRZR PORV.
	d. Reactor vessel head vents - CLOSED	d. Manually close reactor vessel head vent(s).
	e. PRZR vents - CLOSED	e. Manually close PRZR vent(s).
5	Return To Procedure And Step In Effect.	

Comments / Reference: From FRC-0.3A, Steps 2 & 3	Revision # 8

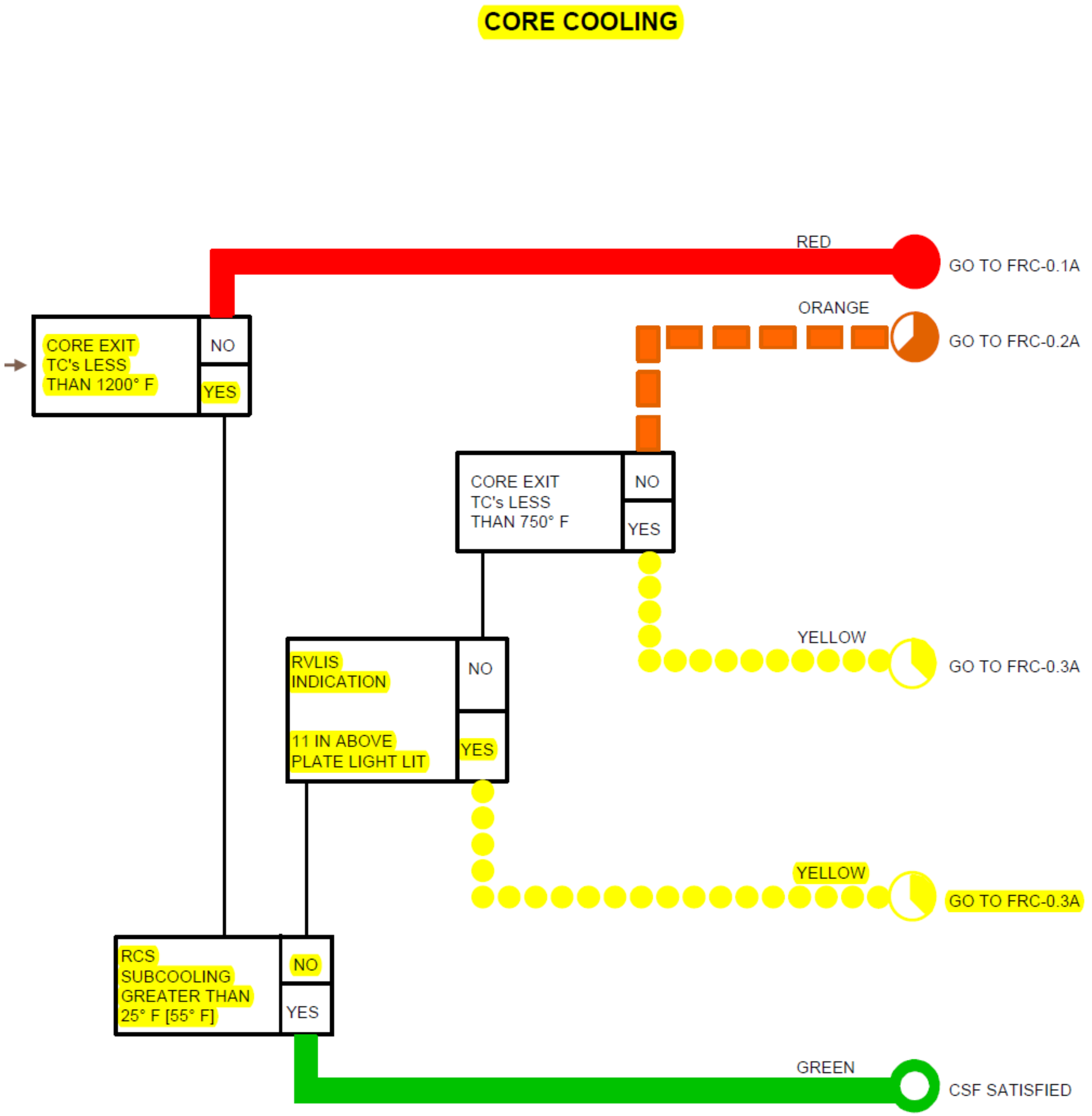
CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRC-0.3A
RESPONSE TO SATURATED CORE COOLING	REVISION NO. 8	PAGE 3 OF 11

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: If ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED, is in effect, this procedure should not be performed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1	Check RWST Level - GREATER THAN LO-LO LEVEL	Go to EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION.
2	Check RHR System Status:	
	a. RHR System - HAS BEEN PLACED IN SERVICE FOR COOLDOWN	a. Go to Step 3.
	b. Go to ABN-104, RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION.	
3	Verify ECCS Flow:	
	a. CCP safety injection flow indicator - CHECK FOR FLOW	a. Start pumps and align valves as necessary.
	b. SI pump flow indicators - CHECK FOR FLOW	b. Start pumps and align valves as necessary.
	c. RCS pressure - LESS THAN 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT)	c. Go to Step 4.
	d. RHR pump flow indicators - CHECK FOR FLOW	d. Start pumps and align valves as necessary.

Comments / Reference: From FRC-0.3A, CSFST	Revision # 8



Comments / Reference: From EOP-1.0A, Attachment 1.A	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 18 OF 44

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT

1. RCP TRIP CRITERIA

Trip all RCPs if BOTH conditions listed below occur:

a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)

b. CCP or SI pump - AT LEAST ONE RUNNING

2. SI REINITIATION CRITERIA

Manually start ECCS pumps as necessary if EITHER condition listed below occurs:

- RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	006 K5.06	
Importance Rating	3.5	

Emergency Core Cooling System: Knowledge of the operational implications of the following concepts as they apply to the ECCS: Relationship between ECCS flow and RCS pressure

Proposed Question: Common 6

The following plant conditions exist following an automatic Safety Injection:

- Containment pressure is 0.7 psig.
- 118 Volt Protection Bus 1PC2 is de-energized.
- Reactor Coolant System (RCS) T_{AVE} is 568°F.
- RCS pressure is 1810 psig.
- All Steam Generator pressures are approximately 1150 psig.

Which of the following indicates the approximate Safety Injection Pump discharge flows?

	Safety Injection Pump 1-01	Safety Injection Pump 1-02
A.	450 gpm	450 gpm
B.	0 gpm	450 gpm
C.	450 gpm	0 gpm
D.	0 gpm	0 gpm

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that when RCS pressure is 1150 psig the SIP flow would be ~ 450 gpm.

B. Incorrect. Plausible because loss of 1PC2 leads to a loss of Train B Safety Injection Sequencer and it may be thought that it leads to a loss of Train A SI sequencer.

C. Incorrect. Plausible because loss of 1PC2 leads to a loss of Train B Safety Injection Sequencer and it may be thought that it leads to a loss of Train A SI sequencer.

D. Correct. At the given RCS pressure, there will be no SIP discharge flow.

Technical Reference(s) EOP-0.0A, Attachment 2, Step 8 Attached w/ Revision # See Comments / Reference

LO21.SYS.ES2.LN, Page 44

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basic design and flow path of the Emergency Core Cooling System.
DEMONSTRATE an understanding of the components of the Emergency Core Cooling System including interrelations with other systems to include interlocks and control loops.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 10, 14
55.43 ______

Comments / Reference: From EOP-0.0A, Attachment 2, Step 8	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 24 OF 115

ATTACHMENT 2
PAGE 3 OF 9

SAFETY INJECTION ACTUATION ALIGNMENT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5	Verify CCW Pumps - RUNNING	☐ Manually start pump(s).
☐ 6	Verify RHR Pumps - RUNNING	☐ Manually start pump(s).
7	Verify Proper CVCS Alignment:	
	☐ a. Verify CCPs - RUNNING	☐ a. Manually start pump(s).
	b. Verify Letdown Relief Valve isolation:	
	☐ 1) Letdown orifice isolation valves - CLOSED	☐ 1) Manually close valve(s).
	2) Letdown isolation valves - CLOSED	☐ 2) Manually close valve(s).
	☐ • 1/1-LCV-459	
	☐ • 1/1-LCV-460	
8	Verify ECCS flow:	
	☐ a. CCP SI flow indicator - CHECK FOR FLOW	☐ a. Manually align valves as appropriate.
	☐ b. RCS pressure - LESS THAN 1700 PSIG(1800 PSIG FOR ADVERSE CONTAINMENT)	☐ b. Go to Step 9 of this attachment.
	☐ c. SIP discharge flow indicators - CHECK FOR FLOW	☐ c. Manually align valves as appropriate.

Comments / Reference: From LO21.SYS.ES2.LN, Page 44	Revision # 06/09/11

SLAVE RELAYS

The slave relays are powered from uPC1 for train "A" and uPC2 for train "B".

The slave relays can be energized either by their associated master relay or else by the Safeguards Test Cabinet test switch. The Safeguards Test Cabinet will be covered later.

The slave relay continuity can be checked, when the MODE SELECTOR Switch is in TEST, and it's associated Master Relay is tested. When the Master Relay is energized, its contacts close to put 15 VDC on the slave relay coil. This is not enough power to energize the Slave relay, but when the Relay test switch is depressed and energizes the small relays on the bottom of the test panel (K 650-653), they open the normal low resistance path to the slave relays and force the 15 VDC through the Continuity Lamps. If the master relay contacts are closed and the Slave relay coil is okay, the continuity lamp will light.

If the lamp remains on after the Relay test pushbutton is released, then the K 650-653 relays have not reclosed their contacts in the low resistance path and the slave relay(s) associated with that test relay is (are) inoperable. If the low resistance path does not realign, when the slave relay is called upon to actuate, the light bulb path to the coil either may provide too much resistance to energize the slave relay coil. Additionally, the light bulb may burnout when subjected to full current of 118 VAC when the MODE SELECTOR switch is in OPERATE (which would also prevent the slave relay from energizing).

Comments / Reference: From CPNPP Exam Bank	Revision # 02/02/03

The following plant conditions exist following an automatic Safety Injection:

- CNTMT pressure 0.7 psig.
- RCS T_{ave} - 568°F.
- RCS pressure - 1810 psig.
- S/G pressure - 1150 psig.

Which of the following describes the reason that both Safety Injection pump discharge flow indications are ZERO (0) gpm?

A. Power to the indications are load shed by SI Sequencer.

B. **NO** auto start signal exists to the SI pumps.

C. The SI pump auto start is blocked by an automatic lockout.

D. NO flow expected at this pressure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 K3.01	
	Importance Rating	3.3	

Pressurizer Relief/Quench Tank System: Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

Proposed Question: Common 7

Given the following conditions:

- A Pressurizer (PRZR) Power Operated Relief Valve (PORV) opened at 2335 psig and will NOT close.
- The associated PRZR PORV Block Valve failed to close manually and the Pressurizer Relief Tank rupture disk has blown.
- PRZR PORV Outlet (Tailpipe) temperature is indicating 260°F.

Which of the following is the expected Containment pressure for the conditions listed?

A. ~5 psig.

B. ~20 psig.

C. ~35 psig.

D. ~50 psig.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that the isenthalpic expansion continues straight across until it intersects with the Saturation Curve. This point corresponds to 5 psig.
B. Correct. With a nominal opening pressure of 2335 psig, the isenthalpic expansion occurs at approximately ~1110 BTUs/lbm. Following this on the curve to the point where 260°F intersects the Saturation Curve corresponds to a Containment pressure of approximately 20 psig.
C. Incorrect. Plausible if psia is not converted to psig.
D. Incorrect. Plausible if psia conversion to psig is reversed.

Technical Reference(s) Steam Tables — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Steam Tables

Learning Objective: **DISCUSS** the operator actions, including all cautions, notes, RNOs and bases associated with EOP-1.0, Loss of Reactor or Secondary Coolant.

Question Source: Bank # X
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, 14
55.43 ______

Comments / Reference: From Steam Tables	Revision # N/A
Selecting a nominal lift pressure of 2335 psig implies an intersection with the Mollier Diagram saturation curve at approximately ~1115 BTUs/lbm. Following an isenthalpic expansion to the point where 260°F intersects the Saturation Curve and read the corresponding Containment pressure.	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	008 A4.01	
Importance Rating	3.3	

Component Cooling Water System: Ability to manually operate and/or monitor in the control room: CCW indications and controls

Proposed Question: Common 8

Given the following conditions:

- Unit 1 is operating at 100% power.
- Train A Component Cooling Water System is in service.
- Annunciator 1-ALB-3B, Window 2.2 – CCW SRG TK TRN A/B EMPTY is in alarm.
- 1-LR-4500, TRN A SRG TK LVL is reading 53% and lowering.

Which of the following is the response of the Component Cooling Water (CCW) System?

A. Train A CCW Pump trips; Train B CCW Pump AUTO starts if RED flagged on the CCW Pump handswitch.

B. Train A CCW Pump trips; Train B CCW Pump remains in standby if RED flagged on the CCW Pump handswitch.

C. Train A Safeguards Loop Isolation Valves close; Train B CCW Pump AUTO starts if GREEN flagged on the CCW Pump handswitch.

D. Train A Safeguards Loop Isolation Valves close; Train B CCW Pump remains in standby if GREEN flagged on the CCW Pump handswitch.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it is a misconception that an empty CCW Surge Tank would trip Train A CCW Pump causing an AUTO start of the Train B CCW Pump. An empty CCW Surge Tank does not trip Train A CCW Pump, therefore, Train B CCW Pump does not auto start.

B. Incorrect. Plausible because it is a misconception that an empty CCW Surge Tank would trip Train A CCW Pump causing an AUTO start of the Train B CCW Pump. An empty CCW Surge Tank does not trip Train A CCW Pump, therefore, Train B CCW Pump remains in standby.

C. Incorrect. Plausible because it is a misconception that closure of Train A Safeguards Loop Isolation valves would lead to an AUTO start of Train B CCW Pump.

D. Correct. Train A Safeguards Loop Isolation Valves will close with CCW Surge Tank level at < 57% and the Train B Component Cooling Water Pump remains in standby.

Technical Reference(s) ALM-0032A, 1-ALB-3B, Window 2.2
ABN-502, Steps 2.1, 2.2, & 3.2
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the Main Control Board/Plant Computer controls, alarms and indications associated with the Component Cooling Water System.

Question Source: Bank # X
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 4, 7
55.43

Comments / Reference: From ALM-0032A, 1-ALB-3B, Window 2.2	Revision # 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0032A
ALARM PROCEDURE 1-ALB-3B	REVISION NO. 7	PAGE 53 OF 169

ANNUNCIATOR NOM./NO.: **CCW SRG TK TRN A/B EMPTY** 2.2

PROBABLE CAUSE:

Failure of manual and automatic makeup
Gross leakage from CCW System

AUTOMATIC ACTIONS:

1-HV-4512, U1 SFGD LOOP TRN A CCW RET VLV, AND 1-HV-4514, U1 SFGD LOOP TRN A CCW SPLY VLV, close
OR
1-HV-4513, U1 SFGD LOOP TRN B CCW RET VLV, AND 1-HV-4515, U1 SFGD LOOP TRN B CCW SPLY VLV, closes

OPERATOR ACTIONS:

1. Determine affected surge tank:
   - 1-LR-4500, TRN A SRG TK LVL
   - 1-LR-4501, TRN B SRG TK LVL

   A. If surge tank level is < 57%, ensure affected safeguard loop is isolated.

      Train A
      - 1-HS-4512, SFGD LOOP CCW RET VLV, closed
      - 1-HS-4514, SFGD LOOP CCW SPLY VLV, closed

      Train B
      - 1-HS-4513, SFGD LOOP CCW RET VLV, closed
      - 1-HS-4515, SFGD LOOP CCW SPLY VLV, closed

2. Ensure both CCW pumps are in service.
   - 1-HS-4518A, CCWP 1
   - 1-HS-4519A, CCWP 2

   A. IF standby CCW pump is NOT supplying non-safeguard loop, THEN refer to ABN-502 for Loss of CCW to the Non-safeguards Loop.

Comments / Reference: From ABN-502, Steps 2.1 & 2.2	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 3 OF 75

2.0 CCW PUMP TRIP

2.1 Symptoms

a. Annunciator Alarms

- CCWP 1/2 OVRLOAD/TRIP (3B-2.3)
- CCW TRN B SFGD LOOP PRESS LO (3B-3.3)
- CCW HX 1/2 OUT & RECIRC FLO LO (3B-3.5)
- CCW TRN A SFGD LOOP PRESS LO (3B-4.3)
- CCW HX 1/2 SPLY FLO LO (3B-4.5)

b. Plant Indications

- Temperature increasing on the components supplied by affected CCW train.

2.2 Automatic Actions

- The standby CCW Pump will start on low CCW pressure in the opposite train.
- The train associated SSW Pump will start on a CCW Pump start.
- The train associated safety chiller will start on a CCW Pump start.

Comments / Reference: From ABN-502, Step 3.2	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 9 OF 75

3.2 Automatic Actions (Continued)

b. The safeguard loop isolation valves close on a train related CCW Surge Tank Empty Signal of approximately 57% (33%).

- Train A

  u-HS-4512, SFGD LOOP CCW RET VLV
  u-HS-4514, SFGD LOOP CCW SPLY VLV

- Train B

  u-HS-4513, SFGD LOOP CCW RET VLV
  u-HS-4515, SFGD LOOP CCW SPLY VLV

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010 K6.01	
	Importance Rating	2.7	

<u>Pressurizer Pressure Control System</u>: Knowledge of the effect that a loss or malfunction of the following will have on the PPCS: Pressure detection systems

Proposed Question: Common 9

Given the following conditions:

- Unit 1 is operating at 50% power.
- PS-455F, PRZR PRESS CTRL CHAN SELECT, is in the '455/456' position.
- PT-456, PRZR PRESS CHAN II, fails high.

Assuming NO operator action, which of the following is the expected plant response?

A. A high pressure Reactor Trip occurs.

B. A low pressure Reactor Trip and Safety Injection occur.

C. Unit remains at power, with pressure being controlled by PORV-456 at approximately 2185 psig.

D. Unit remains at power, with pressure being controlled by PORV-456 at approximately 2355 psig.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because pressure would increase with no operator action if PT-455 were to fail low and, if PORV 456 were to also fail, a high pressure Reactor Trip would occur, but with PT-456 failing high, pressure stabilizes around 2185 psig.

B. Incorrect. Plausible because a low pressure Reactor Trip and Safety Injection would occur with no operator action if PT-455 were to fail high, but with PT-456 failing high pressure stabilizes around 2185 psig.

C. Correct. With PT-456 failing high, PORV 456 opens and actual pressure begins lowering. The PRZR Spray Valves remain closed and the PRZR Heaters will energize as pressure sensed by PT-455 lowers, but the open PORV causes pressure to continue to lower. When PT-457 senses pressure below 2185 psig, the open interlock for PORV 456 is lost and the PORV closes. As pressure increases above 2185 psig due to the heaters, the open interlock for the PORV is restored and pressure will cycle around 2185 psig.

D. Incorrect. Plausible because pressure would increase and cycle around 2335 psig with no operator action if PT-455 were to fail low, but with PT-456 failing high pressure stabilizes around 2185 psig.

Technical Reference(s) ABN-705, Step 2.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Pressurizer Pressure and Level Control System and **PREDICT** the system response.

Question Source: Bank # X
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 ABN-705, Step 2.2	Revision # 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 12	PAGE 4 OF 26

2.2 Automatic Actions

NOTE: Control responses will only occur if failure occurs in a channel selected for control.

a. Control response for a pressurizer pressure channel failure HIGH.

1) PORV will open until pressure is reduced to 2185 psig, then the other channel will close the PORV.

- 1/u-PCV-455A, PRZR PORV
- 1/u-PCV-456, PRZR PORV

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012 K1.02	
	Importance Rating	3.4	

<u>Reactor Protection System</u>: Knowledge of the physical connections and/or cause-effect relationships between the RPS and the following systems: 125V DC system

Proposed Question: Common 10

Given the following conditions:

- Unit 2 is operating at 100% power.
- 125 VDC Bus 2ED1 de-energizes due to a fault on the bus.

Which of the following describes the effect of a loss of the DC bus?

A. Reactor will trip due to a shunt trip of Train A Reactor Trip Breaker.

B. Reactor will trip due to an undervoltage trip of Train A Reactor Trip Breaker.

C. A shunt trip signal will NOT be capable of opening Train A Reactor Trip Breaker.

D. An undervoltage trip signal will NOT be capable of opening Train A Reactor Trip Breaker.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because most actuations are de-energized to actuate, but the shunt trip requires that 125 VDC be applied to the shunt trip coil to cause the breaker to open.
B. Incorrect. Plausible because the UV trip receives power from a DC power supply, but the power supply is 48 VDC within SSPS.
C. Correct. 125 VDC Bus 2ED1 supplies power to the shunt trip coil for Train A Trip Breaker. UV coils and shunt trip relays are supplied from 48 VDC from SSPS. The shunt trip coil is normally de-energized and without power available, a shunt trip of Train A Trip Breaker is not possible.
D. Incorrect. Plausible because the UV trip receives power from a DC power supply, but the power supply is 48 VDC within SSPS and is de-energized to actuate.

Technical Reference(s) LO21.SYS.ES2.LN, Page 66 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Solid State Protection System including interrelations with other systems to include interlocks and control loops

Question Source: Bank # X
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 6
55.43 ______

Comments / Reference: From LO21.SYS.ES2.LN, Page 66	Revision # 09/02/04

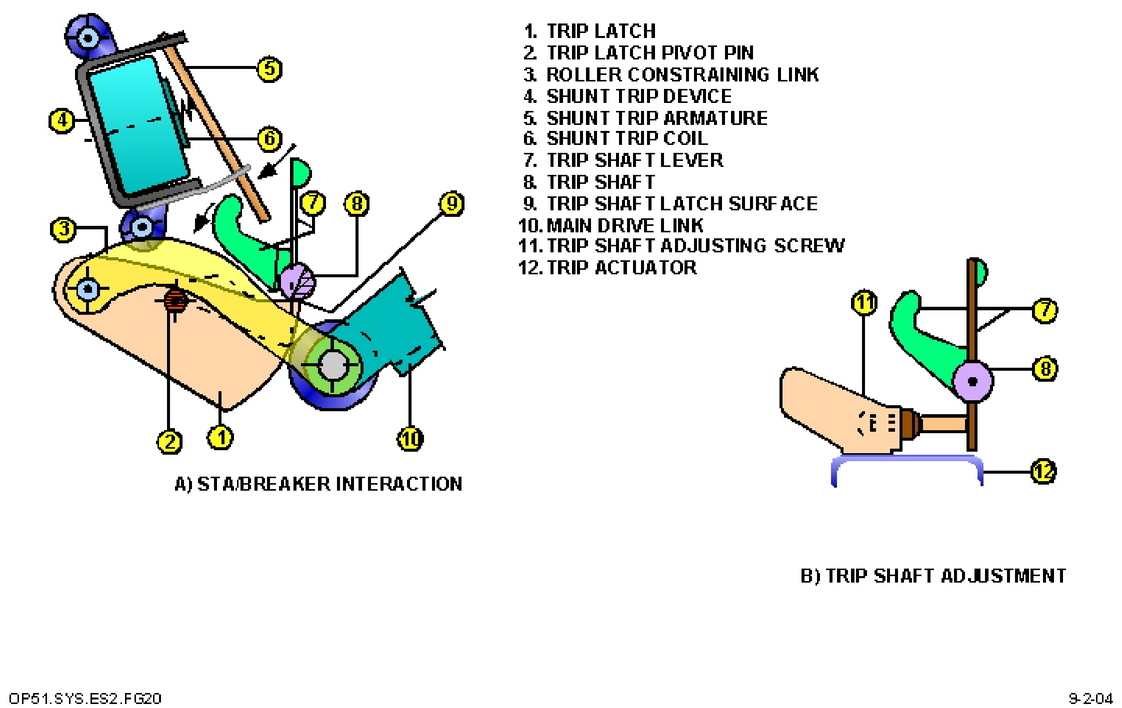
THE SHUNT TRIP COIL

The Shunt Trip Coil on the Reactor Trip and Bypass breakers is actuated by any of the following:

- Either Manual Reactor Trip switch
- Either Manual SI Actuation switch
- Auto Shunt Trip Relay STA(B) - only on the Reactor Trip Breakers
- Both Bypass Breakers connected and closed- only on the Bypass Breakers

The Shunt Trip Coil is normally de-energized. When it is actuated by applying 125 VDC to its coil, the coil attracts an armature which pushes the trip lever on the breaker trip shaft, causing the breaker to trip (**Figure 20**). This trip device is a mechanically less complicated and more forceful mechanism than the undervoltage trip coil. The power to the shunt trip coils comes from uED1 (2)-2.

SHUNT TRIP COIL (ATTACHMENT)



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	013 K5.01	
Importance Rating	2.8	

Engineered Safety Features Actuation System: Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Definitions of safety train and ESF channel

Proposed Question: Common 11

Given the following conditions:

- All Unit 1 systems are in a normal MODE 1 alignment.
- Containment pressure is rising.
- Annunciator 1-ALB-6C, Window 4.7 – CNTMT PRESS HI SI ACT, is FLASHING red.
- Annunciator 1-PCIP, Window 1.8 – SI ACT, is ON and SOLID.

Which of the following describes the status of the Engineered Safety Feature Actuation System?

At least [1] Containment Pressure Intermediate Range Channel(s) has(have) actuated on HI-1 and [2] Train(s) of Emergency Core Cooling System equipment is(are) operating.

	[1]	[2]
A.	one	one
B.	one	two
C.	two	one
D.	two	two

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that to actuate a High Containment Pressure SI requires only 1 of 3 pressure channels to be less than the setpoint. Alarm PCIP Window 1.8 – SI ACT ON and SOLID correctly indicates that 2 of 2 ECCS equipment trains are operating.
B. Incorrect. Plausible if thought that to actuate a High Containment Pressure SI requires only 1 of 3 pressure channels to be less than the setpoint. Alarm PCIP Window 1.8 – SI ACT ON and SOLID correctly indicates that 2 of 2 ECCS equipment trains are operating.
C. Incorrect. High Containment Pressure SI requires 2 of 3 channels to exceed setpoint. A single Train actuation of SI will place PCIP Window 1.8 in ON and flashing.
D. Correct. High Containment Pressure SI requires 2 of 3 channels to exceed setpoint. A dual Train actuation of SI will place PCIP Window 1.8 in ON and SOLID.

Technical Reference(s): ALM-0063A, 1-ALB-6C, Window 4.7; ALM-0065A, PCIP Window 1.8; EOP-0.0A, Step 4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Solid State Protection System.

COMPREHEND the normal, abnormal and emergency operation of the Safety Injection and Blackout Sequencers.

Question Source:

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

Question History: Last NRC Exam

Question Cognitive Level:

Memory or Fundamental Knowledge	
Comprehension or Analysis	X

10 CFR Part 55 Content:

55.41 5, 7, 8

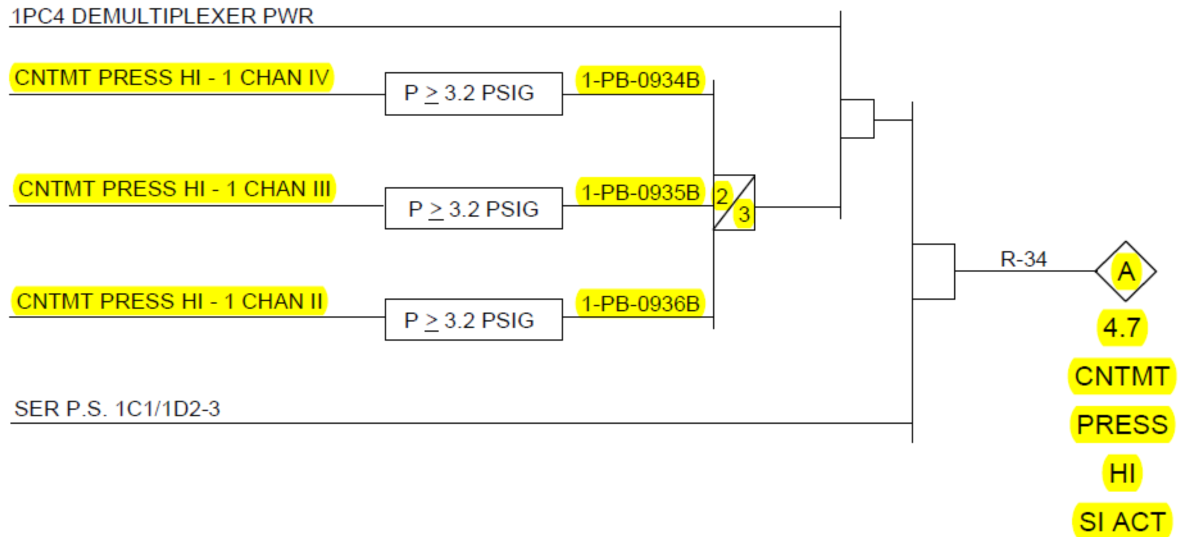
55.43

Comments / Reference: From ALM-0063A, 1-ALB-6C, Window 4.7	Revision # 5

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0063A
ALARM PROCEDURE 1-ALB-6C	REVISION NO. 5	PAGE 66 OF 69

ANNUNCIATOR NO.: 4.7

LOGIC:

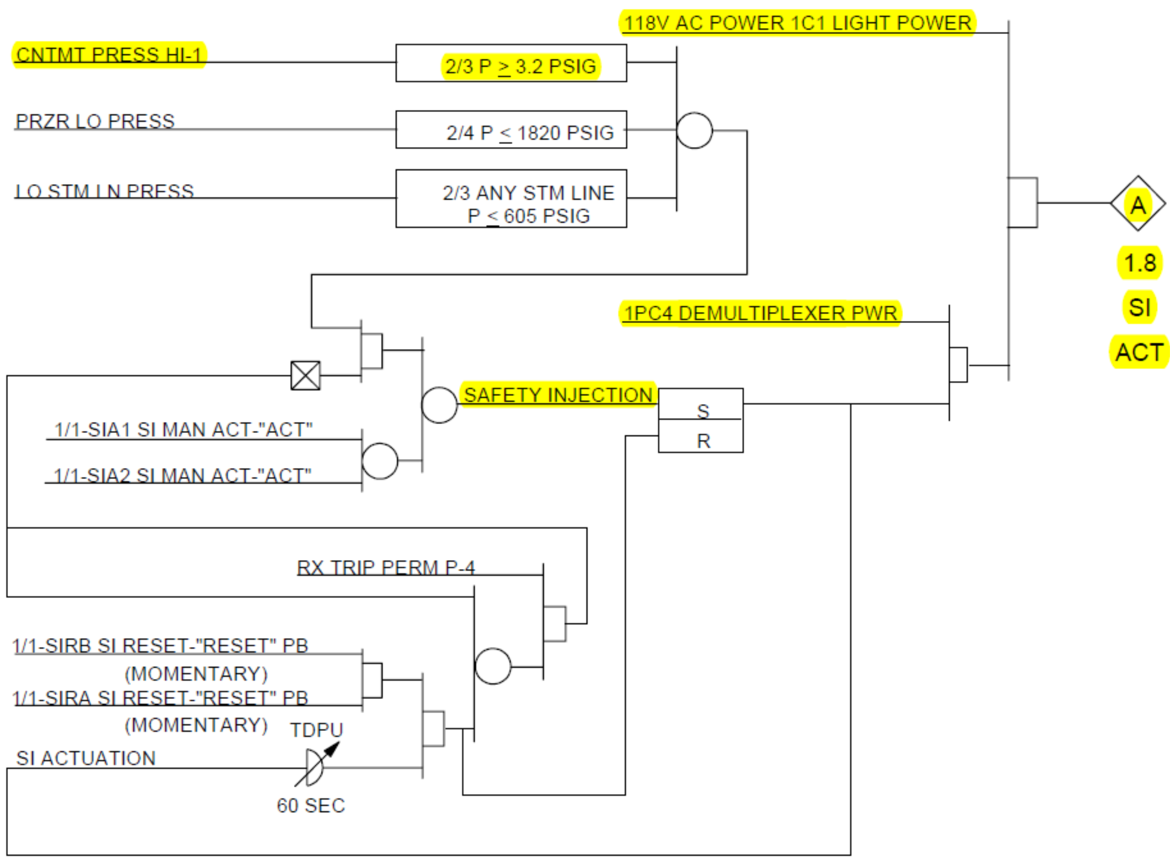


Comments / Reference: From ALM-0065A, PCIP Window 1.8	Revision # 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 20 OF 73

ANNUNCIATOR NO.: 1.8

LOGIC:



Comments / Reference: From EOP-0.0A, Step 4	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 4 OF 115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4	Check SI Status:	
	a. Check If SI Is Actuated: • SI actuation as indicated on the First Out Annunciator 1-ALB-6C • SI Actuated blue status light - ON	a. Check if SI is required: • Steam Line Pressure less than 610 psig. • Pressurizer Pressure less than 1820 psig. • Containment Pressure greater than 3.0 psig. <u>IF</u> SI is required, <u>THEN</u> manually actuate SI from either handswitch. <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> go to EOS-0.1A, REACTOR TRIP RESPONSE, Step 1.
	b. Verify Both Trains SI Actuated: • SI Actuated blue status light - ON <u>NOT</u> FLASHING	b. Manually Actuate SI.

Comments / Reference: From CPNPP Exam Bank	Revision # 03/24/11

Given the following conditions:

- All Unit 1 systems are in a normal MODE 1 alignment.
- A steam leak occurs on Steam Generator 1-02 outside Containment.
- Annunciator 1-ALB-6C, Window 3.7 – MSL PRESS LO SI ACT, is FLASHING red.
- Annunciator PCIP Window 1.8 – SI ACT, is ON and FLASHING.

Which of the following describes the status of the Engineered Safety Feature Actuation System?

At least _____ Main Steam Line Pressure Low Channel(s) has(have) actuated and _____ train(s) of Emergency Core Cooling System equipment is(are) operating.

A. one one

B. one two

C. <u>two</u> <u>one</u>

D. two two

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	022 A1.02	
Importance Rating	3.6	

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

Proposed Question: Common 12

Given the following conditions:

- Unit 1 Containment pressure is 1.2 psig and slowly lowering.
- A Containment Vent is in progress per SOP-801A, Containment Ventilation System.

Which of the following is the MINIMUM pressure limit that must be maintained to ensure the Containment internal design pressure is not exceeded during an inadvertent Containment Spray actuation?

A. -0.3 psig

B. 0.0 psig

C. 1.2 psig

D. 1.3 psig

Proposed Answer: A

Explanation:

A. Correct. Per Technical Specification LCO 3.6.4.
B. Incorrect. Plausible because SOP-801A instructs the operator to reduce Containment pressure to atmospheric when venting Containment.
C. Incorrect. Plausible because the alarm setpoint for Containment Narrow Range pressure high is 1.226 psig.
D. Incorrect. Plausible because the Technical Specification maximum limit is 1.3 psig.

Technical Reference(s): Technical Specification LCO 3.6.4; SOP-801A, Step 4.1; ALM-0031A, 1-ALB-3A, Window 4.6 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Containment Ventilation 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	9, 10	
	55.43		

Comments / Reference: From Technical Specification LCO 3.6.4	Amendment # 158

Containment Pressure
3.6.4

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ - 0.3 psig and ≤ + 1.3 psig.

APPLICABILITY: MODES 1, 2, 3, and 4

Comments / Reference: From SOP-801A, Step 4.1	Revision # 14

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. SOP-801A
CONTAINMENT VENTILATION SYSTEM	REVISION NO. 14 CONTINUOUS USE	PAGE 7 OF 56

4.0 LIMITATIONS AND NOTES

4.1 Limitations

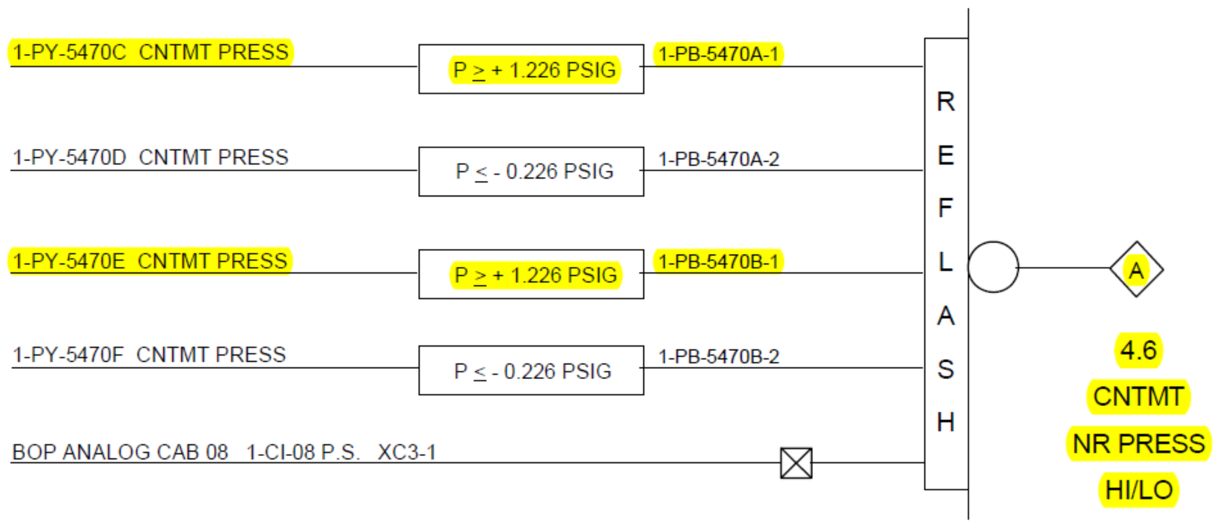
- Primary containment internal indicated pressure shall be maintained between -0.3 AND 1.3 psig as per TS 3.6.4.
- Primary containment average air temperature shall NOT exceed 120°F per TS 3.6.5.

Comments / Reference: From ALM-0031A, 1-ALB-3A, Window 4.6	Revision # 8

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0031A
ALARM PROCEDURE 1-ALB-3A	REVISION NO. 8	PAGE 102 OF 113

ANNUNCIATOR NO.: 4.6

LOGIC:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022 A4.01	
	Importance Rating	3.6	

Containment Cooling System: Ability to manually operate and/or monitor in the control room: CCS fans

Proposed Question: Common 13

Assuming Containment Recirculation Cooler Fan 1-01 was operating, which of the following identifies the expected handswitch indication lights for HS-5405A, CNTMT FN CLR FN 1, two minutes following a Safety Injection?

	GREEN FAN	AMBER MISMATCH	RED FAN
A.	OFF	OFF	ON
B.	ON	ON	ON
C.	ON	ON	OFF
D.	OFF	OFF	OFF

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because these are the indications that would be available if the fan were still operating which would occur if the signal were a Blackout instead of a Safety Injection (SI), but the fan will load shed on the SI.
B. Incorrect. Plausible if thought that the shunt trip of the breaker caused all lights to illuminate.
C. Correct. The fan is load shed via a shunt trip of the breaker upon receipt of an SI signal. With the handswitch in the AUTO AFTER START position, this will cause the amber light to energize. The green light is on because the breaker is open, which is also why the red light is off.
D. Incorrect. Plausible if thought that the shunt trip of the breaker caused all lights to extinguish.

Technical Reference(s) EOP-0.0A, Attachment 8; ALM-0031A, 1-ALB-3A, Window 1.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Containment Ventilation System.

Question Source:	Bank #	X	
	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, 9	
	55.43		

Comments / Reference: From EOP-0.0A, Attachment 8	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 52 OF 115

ATTACHMENT 8
PAGE 3 OF 10

LOAD SHEDDING
1-MLB-9

MLB		LOAD DESCRIPTION	CONTROL LOCATION
3.3		LOAD SHEDDING COMPLETE XEB3-2 (AUX 852 East Side in Passageway)	
☐	•	PRIMARY PLANT EXHAUST FAN X-15 MOTOR BREAKER	XEB3-2/4G/BKR
☐	•	PRIMARY PLANT EXHAUST FAN X-17 MOTOR BREAKER	XEB3-2/4M/BKR
☐	•	PRIMARY PLANT EXHAUST FAN X-19 MOTOR BREAKER	XEB3-2/5G/BKR
☐	•	480/120 VAC TRANSFORMER (SPACE HEATER) XEB3-2/2F/TR FEEDER BREAKER	XEB3-2/2F/BKR
1.4		LOAD SHEDDING COMPLETE 1EB1 & 1EB3 (SFGDs 810 Train A Swgr)	
		1EB1	
☐	•	PDP	1/1-APPD (CB-06)
☐	•	PRZR CTRL HTR GROUP C	1/1-PCPR (CB-05)
☐	•	CNTMT FN CLR FN 1	1-HS-5405A (CB-03)

Comments / Reference: From ALM-0031A, 1-ALB-3A, Window 1.2	Revision # 8

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0031A
ALARM PROCEDURE 1-ALB-3A	REVISION NO. 8	PAGE 15 OF 113

ANNUNCIATOR NOM./NO.: **CNTMT FN CLR FN 1 ΔP LO** **1.2**

PROBABLE CAUSE:

Containment Air Cooling AND Recirc Fan 1 failure
Operating only one fan during an outage. (TE-97-000171)

AUTOMATIC ACTIONS: None

NOTE:
- Containment fan cooler fans start on a BOS.
- This is an expected alarm during outages if only one fan cooler is operating AND /OR Equipment Hatch is removed.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	026 A2.07	
Importance Rating	3.6	

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: Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding or sump level below cutoff (interlock) limit

Proposed Question: Common 14

Given the following conditions:

- Unit 1 experienced a Large Break Loss of Coolant Accident (LOCA) about 20 minutes ago.
- Train A Emergency Core Cooling System (ECCS) Pumps are running in Cold Leg Recirculation Mode per EOS-1.3 A, Transfer to Cold Leg Recirculation.
- Transfer of Containment Spray Pumps to recirculation is complete with the following indications:
  - Containment Spray Pumps 1-01 and 1-03 are running with:
    - 3750 gpm and stable discharge flow on each pump.
    - 265 psig and stable discharge pressure on each pump.
  - Containment Spray Pumps 1-02 and 1-04 are running with:
    - 3800 to 2800 gpm fluctuating discharge flow on each pump.
    - 270 to 160 psig fluctuating discharge pressure on each pump.

Which of the following lists the action required per EOS-1.3 A, Transfer to Cold Leg Recirculation?

A. Close 1-HV-4777, CS HX 1-02 OUT VLV.

B. Open 1-HV-4759, RWST TO CS PMP 1-02 & 1-04 SUCT VLV.

C. Place Containment Spray Pumps 1-02 & 1-04 handswitches in PULLOUT.

D. Close 1-HV-4783, CNTMT SMP TO CS PMP 1-02 & 1-04 SUCT ISOL VLV.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 1-HV-4777, CS HX 1-02 OUT VLV is closed if Containment Spray Pumps have been stopped due to low RWST level, however, the valve would be reopened when the Containment Spray Pumps were started.

B. Incorrect. Plausible if thought that opening 1-HV-4759, RWST TO CS PMP 1-02 & 1-04 SUCT VLV would assist with eliminating cavitation in the Containment Spray Pumps, however, under no condition in EOS-1.3A is this valve opened.

C. Correct. Per the EOS-1.3A, Step 3 CAUTION.

D. Incorrect. Plausible if thought that closing 1-HV-4783, CNTMT SMP TO CS PMP 1-02 & 1-04 SUCT ISOL VLV would assist with eliminating cavitation in the Containment Spray Pumps, however, under no condition in EOS-1.3A is this valve closed.

Technical Reference(s): EOS-1.3A, Step 3 CAUTION; EOS-1.3A, Step 3 & Step 4 RNO — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Containment Spray system including interrelations with other systems to include interlocks and control loops.

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, 14
- 55.43

Comments / Reference: From EOS-1.3A, Step 3 CAUTION	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 4 OF 53

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: Any ECCS pump taking suction from RWST should be stopped at RWST EMPTY. Any Containment Spray pump taking suction from RWST should be stopped when RWST level reaches 0%.

CAUTION: Any ECCS or Containment Spray pump that loses suction or shows indication of cavitation should be stopped. The CCP and SI pump should be stopped before stopping the RHR pump.

Comments / Reference: From EOS-1.3A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 9 OF 53

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

3) IF containment spray pumps have been stopped due to RWST level, THEN perform the following:

A) Ensure CS HX 1 AND 2 OUT VLVs closed and handswitches in PULL-OUT:

- 1-HS-4776
- 1-HS-4777

B) Perform Steps C) and D) simultaneously.

C) Open CS HX 1 AND 2 OUT VLVs:

- 1-HS-4776
- 1-HS-4777

D) WHEN CS HX OUT VLVs begin to open, THEN start CS PUMPS:

- 1-HS-4764 AND 1-HS-4765
- 1-HS-4766 AND 1-HS-4767

Comments / Reference: From EOS-1.3A, Step 4 RNO	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 7 OF 53

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1H] * 4	Align Containment Spray System For Recirculation:	
	a. Check RWST level - LESS THAN 6%	a. Perform the following: 1) IF Containment Spray ACTUATED, THEN continue with Step 5. WHEN RWST level less than 6%, THEN perform Step 4b. 2) IF Containment Spray NOT ACTUATED, THEN perform the following to align Containment Spray System: A) Stop containment spray pumps and place in standby. B) Open CNTMT SMP TO CSP 1 & 3 AND 2 & 4 SUCT ISOL VLVs: • 1-HS-4782 • 1-HS-4783 C) Close RWST TO CSP 1 & 3 AND 2 & 4 SUCT VLVS: • 1-HS-4758 • 1-HS-4759

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039 A1.05	
	Importance Rating	3.2	

Main and Reheat Steam System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: RCS Tave

Proposed Question: Common 15

Given the following conditions on Unit 1:

- A Reactor and Turbine Trip have just occurred from 100% power.
- Loop 1 T_{AVE} Channel has failed HIGH.

Which of the following describes the effect on the Steam Dump System due to this failure and any operator action that is required?

A. Steam Dumps will remain OPEN and reduce T_{AVE} to below the No-Load T_{AVE} of 557ºF. Close the Steam Dump Valves by placing one of the two Steam Dump Interlock Select Switches to OFF.

B. Steam Dumps will remain OPEN and reduce T_{AVE} to below the No-Load T_{AVE} of 557ºF. Take MANUAL control of the Plant Trip Controller and close the Steam Dump Valves.

C. Steam Dump System will automatically shift to the Steam Pressure Mode and control at set pressure. Ensure the set pressure maintains T_{AVE} at the No-Load value of 557ºF.

D. The Steam Dump System will automatically shift from the Load Rejection Controller to the Plant Trip Controller and Steam Dump Valves will maintain No-Load T_{AVE}. Select the faulty T_{AVE} signal on the Tave Channel DEFEAT Switch.

Proposed Answer: A

Explanation:

A. Correct. The Steam Dump Valves will control based on the differential temperature between the Average T_{AVE} and No-Load T_{AVE}. The Average T_{AVE} will be higher based on the channel failure and keep the valves open past where they should be closed. EOP-0.0A, Reactor Trip or Safety Injection requires the operator to stop dumping steam if temperature goes below 557ºF and this is accomplished by selecting a Steam Dump Interlock Switch to OFF.

B. Incorrect. Plausible because the Steam Dump Valves will lower temperature below the program, however, the Plant Trip Controller has no manual controls for operators to use.

C. Incorrect. Plausible because it could be thought that the system would swap to Pressure Control because this is the mode normally used at low power or during plant startup and the T_{AVE} signal would not be controlling, however, the system does not automatically shift to Pressure Control mode.

D. Incorrect. Plausible because the automatic shift to the Plant Trip Controller does occur and it could be thought that this control would be unaffected by the T_{AVE} failure, however, the T_{AVE} failure affects both the Load Rejection and Plant Trip Controllers and the Steam Dump Valves would still remain open below No-Load T_{AVE}.

Technical Reference(s) ABN-704, Steps 2.2.a, 2.3.2 and 2.3.3 / EOP-0.0A, Step 9 RNO a — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Steam Dump System.

Question Source: Bank # X
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 4, 5, 7, 10
55.43

Comments / Reference: From ABN-704, Step 2.2.a	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-704
Tc/N-16 INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 4 OF 14

2.1 b. 4) One OTN16 setpoint higher or lower than the other three.

- u-TI-411B, RC LOOP 1 OT N16 SETPOINT CHAN I
- u-TI-421B, RC LOOP 2 OT N16 SETPOINT CHAN II
- u-TI-431B, RC LOOP 3 OT N16 SETPOINT CHAN III
- u-TI-441B, RC LOOP 4 OT N16 SETPOINT CHAN IV

5) One OPN16 setpoint higher or lower than the other three.

- u-NI-411A, RC LOOP 1 OP N16 SETPOINT CHAN I
- u-NI-421A, RC LOOP 2 OP N16 SETPOINT CHAN II
- u-NI-431A, RC LOOP 3 OP N16 SETPOINT CHAN III
- u-NI-441A, RC LOOP 4 OP N16 SETPOINT CHAN IV

2.2 Automatic Actions

a. Any failure that results in an increased Loop Tave will cause Average Tave to be higher with the following actions:

- Rapid control rod insertion due to Tave-Tref mismatch if in AUTO.
- Steam dumps will open if armed with a C-7 (loss of load).
- Pressurizer reference level increase (to a maximum of 60%) with charging flow increase when in auto.

Comments / Reference: From ABN-704, 2.3.2 and 2.3.3	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-704
Tc/N-16 INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 5 OF 14

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:

- If the failed channel was reading lower than the substituted channel, then AVE Tave will increase when the failed channel is defeated due to another channel being substituted for the failed signal to maintain accurate averaging.
- Rod Control should remain in MANUAL until all channels are operable. This does not preclude placing rods in AUTO during rapidly changing transient conditions such as runbacks, etc. as long as rod control is returned to MANUAL when the plant is stabilized.

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1	Place Control Rods in - MANUAL	
☐ 2	Select the failed channel on: u-TS-412T, Tave CHAN DEFEAT	
☐ 3	Verify Steam Dump System: • NOT actuated • NOT armed.	IF Steam Dump operation is not required, THEN place one of the two interlock select switches in OFF: • 43/u-SDA, STM DMP INTLK SELECT • 43/u-SDB, STM DMP INTLK SELECT

Comments / Reference: From EOP-0.0A, Step 9 RNO a	Revision #

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 8 OF 115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 9	Check RCS Temperature - • RCS AVERAGE TEMPERATURE STABLE AT OR TRENDING TO 557°F	IF temperature less than 557°F and decreasing, THEN perform the following: a. Stop dumping steam.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	059 K4.16	
Importance Rating	3.1	

Main Feedwater System: Knowledge of the MFW design feature(s) and/or interlock(s) that provide for the following: Automatic trips for MFW pumps

Proposed Question: Common 16

Which of the following conditions or situations will result in an automatic trip of a Main Feedwater Pump?

A. Safety Injection Signal.

B. High Main Feedwater Pump vibration.

C. Low Condenser vacuum of 21" HG.

D. LO-LO T_{AVE} with P-4, Reactor Trip Signal.

Proposed Answer: A

Explanation:

A. Correct. This is the only signal listed that will result in trip of the Main Feedwater Pump.
B. Incorrect. Plausible because the Main Feedwater Pump is equipped with a vibration monitor, however, high vibration does not generate a Main Feedwater Pump trip.
C. Incorrect. Plausible because the Main Feedwater Pump will trip on low vacuum, however, the setpoint is 17.5" HG.
D. Incorrect. Plausible because this permissive will trip the Main Turbine and initiate automatic Feedwater Isolation, however, it does not generate a Main Feedwater Pump trip.

Technical Reference(s):
SOP-302A, Step 4.2.C
ALM-0065A, 1-PCIP, Window 1.5
LO21.SYS.MF1.LN, Page 55

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Main Feedwater System and **PREDICT** the system response.

Question Source:
Bank # X
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 SOP-302A, Step 4.2.C	Revision # 17

CPSES SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-302A
FEEDWATER SYSTEM	REVISION NO. 17	PAGE 10 OF 207

4.2 Notes

A. FWP trip above 60% power will initiate a turbine runback to 60% power at 35%/min.

B. Trip of both FWPs will initiate an auto start of MDAFWPS, isolate condenser makeup and reject, and isolate Steam Generator Blowdown and Sampling.

C. The FWPs will automatically trip on the following conditions:

- SI signal
- Low Vacuum - 2/3 coincidence with 2 sec T.D. - 17.5 in Hg
- Low Lube Oil Pressure - Pump end 7 psig (2/3 with 2 sec T.D.)
  Turb end 4 psig (2/3 with 2 sec T.D.)
- Overspeed - 5663 to 5777 rpm
- Hi-Hi Steam Generator Level (P-14) - 84% NR level
- Thrust bearing wear - 2/3 coincidence
- Lo Suction Pressure - 2/3 coincidence - Staggered Pump Trip:
  190 psig - Pump A (30 sec time delay)
  190 psig - Pump B (45 sec time delay)
- Lo-Lo Suction Pressure - 2/3 coincidence - both pumps trip at ≈170 psig (4 sec time delay)
- Hydraulic trip header - 2/3 coincidence

Comments / Reference: From ALM-0065A, 1-PCIP, Window 1.5	Revision # 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 15 OF 73

ANNUNCIATOR NOM./NO.: **RX TRIP PERM P-4** **1.5**

PROBABLE CAUSE:

Automatic or manual reactor trip

> NOTE: This window is normally illuminated in Mode 3-6 unless rod drop time testing is in progress.

AUTOMATIC ACTIONS:

Turbine trip
Permits manual reset of safety injection actuation
Blocks automatic safety injection actuation following manual reset of safety injection
Provides automatic feedwater isolation when coincident with 2 of 4 low $T_{AVE} < 564°F$
Train A arms the condenser steam dump valves
Train B automatically shifts the steam dump controller from loss of load to plant trip mode

OPERATOR ACTIONS:

1. Go to EOP-0.0A.

Comments / Reference: From LO21.SYS.MF1.LN, Page 55	Revision # 05/31/07

FEED PUMP VIBRATION MONITOR

The Unit 1 main feed pumps use the Bentley-Nevada proximity probe type vibration monitors. VD1 and VD2 sense vibration amplitude in the X direction while VD3 and VD4 sense amplitude in the Y direction. The X and Y signals are auctioneered high for display. These where chosen because they can interface directly with the MK-V digital control system and are displayed on the <I> and plant computer. Unit 2 has the original Bentley-Nevada vibration detectors and the MK-V. The Unit 2 MK-V will input to the plant computer just like Unit 1 but does not have the axial vibration data input. Unit 1 uses all live data input.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061 K5.03	
	Importance Rating	2.6	

Auxiliary/Emergency Feedwater System: Knowledge of the operational implications of the following concepts as they apply to the AFW: Pump head effects when control valve is shut

Proposed Question: Common 17

Given the following conditions:

- Unit 1 has experienced a Main Steam Line Break inside Containment from Steam Generator 1-02.
- Containment pressure is 12 psig and slowly increasing.
- Following isolation of Steam Generator 1-02, Steam Generator 1-01 is identified as ruptured.
- Steam Generator 1-01 narrow range level is 40% and rising.

Once Steam Generator level reaches [1] %, Auxiliary Feedwater flow should be stopped to Steam Generator 1-01. When the Flow Control Valve is closed Motor Driven Auxiliary Feedwater Pump 1-01 discharge pressure will [2].

	[1]	[2]
A.	43	increase
B.	43	decrease
C.	50	increase
D.	50	decrease

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the pump head will increase when the Flow Control Valve is closed, however, with Adverse Containment parameters this action should not be taken until the level in the ruptured Steam Generator is a minimum of 50%.

B. Incorrect. Plausible if thought that overall flow would increase when the Flow Control Valve is closed as occurs with other pumps (i.e., the miniflow valve opens when the normal discharge path is closed). However, with the Auxiliary Feedwater Pumps the miniflow is open throughout the evolution. Additionally, with Adverse Containment parameters this action should not be taken until the level in the ruptured Steam Generator is a minimum of 50%.

C. Correct. Given the conditions listed, once Steam Generator level rises above 50%, the Flow Control Valve should be closed restricting the pump discharge to only miniflow, this restriction of the discharge flow will result in an increase in discharge pressure.

D. Incorrect. Plausible because once Steam Generator level rises above 50%, the Flow Control Valve should be closed restricting the pump discharge to only miniflow. If thought that overall flow would increase when the Flow Control Valve is closed as occurs with other pumps (i.e., the miniflow valve opens when the normal discharge path is closed), and discharge pressure could decrease. However, with the Auxiliary Feedwater Pumps the miniflow is open throughout the evolution.

Technical Reference(s) EOP-3.0A, Step 7 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from EOP-3.0, 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 ___
Comprehension or Analysis X

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

Comments / Reference: From EOP-3.0A, Step 7	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 10 OF 103

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 7	Check Intact SG Levels:	
	a. Narrow range level - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)	a. Maintain total AFW flow greater than 460 gpm until narrow range level greater than 43% (50% FOR ADVERSE CONTAINMENT) in at least one intact SG.
	b. Control AFW flow to maintain narrow range level between 43% and 60%.	b. IF narrow range level in any intact SG continues to increase in an uncontrolled manner, THEN stop RCS cooldown and return to Step 1.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	062 K4.02	
Importance Rating	2.5	

AC Electrical Distribution System: Knowledge of the AC electrical distribution design feature(s) and/or interlock(s) which provide for the following: Circuit breaker automatic trips

Proposed Question: Common 18

Which of the following describes a design feature of the Second Level Undervoltage Relays?

A. Starts the Emergency Diesel Generators and initiates the Blackout Sequencer.

B. Provides transfer to an energized Offsite Power source upon a loss of bus voltage.

C. Protects 1E motors from a low voltage/high current condition by opening the Preferred and Alternate Feeder Breakers.

D. Ensures the opening of the Alternate Feeder Breaker within one second after the Preferred Feeder Breaker closes.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because the Emergency Diesel Generators will start due to bus undervoltage, however, the Blackout Sequencer is initiated from another source.
B. Incorrect. Plausible if thought that was the function of these relays, however, the 2nd Level Undervoltage Relays are used on the 1E Safeguards Buses.
C. Correct. Per ABN-602, automatic actions.
D. Incorrect. Plausible because this action can occur, however, it is not associated with these relays.

Technical Reference(s): ABN-602, Step 2.2 & Attachment 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the 6.9 KV and 480 V Electrical Distribution System and **PREDICT** the system response.

Question Source: Bank # X
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 ABN-602, Step 2.2	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 4 OF 107

2.2 Automatic Actions

- Second level undervoltage at 6192 V (6.9 KV bus) or 442.4 V (480 V bus) will trip 6.9 KV safeguard bus preferred and alternate feeds. Attachment 3 lists specific undervoltage response.
- Diesel generators auto-start and energize affected safeguard bus, loading via blackout sequencer. Attachment 2 lists specific sequence.
- Transformer XST2 or XST2A faults isolate 1EA1 and 1EA2 preferred source, causing slow transfer.
- Transformer 1ST fault momentarily de-energizes XST2 OR XST2A, causing a slow transfer.
- Transformer XST1 fault isolates 2EA1 and 2EA2 preferred source, causing slow transfer.
- Both affected 6.9 KV safeguard bus offsite feeder breakers open.
- Associated diesel generator starts
- Equipment powered from affected bus de-energizes.

Comments / Reference: From ABN-602, Attachment 3	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 81 OF 107

ATTACHMENT 3
PAGE 1 OF 1

SECOND LEVEL UNDERVOLTAGE OPERATIONS

1. 6.9 KV Second Level Undervoltage Without SI

   - 6192 volts for approximately 46 seconds will initiate slow bus transfer by tripping preferred breaker to affected 6.9 KV 1E bus.

   - IF 6.9 KV 1E bus voltage is NOT restored above approximately 6240 volts within approximately 2 seconds by alternate source, THEN alternate breaker will also trip.

2. 6.9 KV Second Level Undervoltage With SI

   - 6192 volts for approximately 7.5 seconds will initiate slow bus transfer by tripping preferred breaker to affected 6.9 KV 1E bus.

   - IF 6.9 KV 1E bus voltage is NOT restored above approximately 6240 volts within approximately 2 seconds by alternate source, THEN alternate breaker will also trip.

3. 480 V Second Level Undervoltage Without SI

   - 442.4 volts for approximately 54 seconds will initiate a slow bus transfer by tripping preferred breaker to affected 6.9 KV 1E bus.

   - IF 480 V 1E bus voltage is NOT restored above approximately 444 volts within approximately 2 seconds by alternate source, THEN alternate breaker will also trip.

4. 480 V Second Level Undervoltage With SI

   - 442.4 volts for approximately 8 seconds will initiate slow bus transfer by tripping preferred breaker to affected 6.9 KV 1E bus.

   - IF 480 V 1E bus voltage is NOT restored above approximately 444 volts within approximately 2 seconds by alternate source, THEN alternate breaker will also trip.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063 A3.01	
Importance Rating	2.7	

DC Electrical Distribution System: Ability to monitor automatic operation of the DC electrical distribution, including: Meters, annunciators, dials, recorders and indicating lights

Proposed Question: Common 19

Given the following conditions of Unit 1 DC Safeguards Bus voltage:

- DC Bus 1ED1 is at 135 volts with 1 amp CHARGE.
- DC Bus 1ED2 is at 134 volts with 2 amp CHARGE.
- DC Bus 1ED3 is at 135 volts with 1 amp CHARGE.
- DC Bus 1ED4 is at 134 volts with 2 amp CHARGE.

After a plant transient the following is observed:

- DC Bus 1ED1 is at 134 volts with 2 amp CHARGE.
- DC Bus 1ED2 is at 123 volts with 188 amp DISCHARGE.
- DC Bus 1ED3 is at 134 volts with 1 amp CHARGE.
- DC Bus 1ED4 is at 124 volts with 190 amp DISCHARGE.

Which of the following events has caused the change in DC Safeguards Bus voltage?

A loss of...

A. Motor Control Center XEB1-1.

B. Safeguards Bus 1EA1.

C. Motor Control Center XEB2-1.

D. Safeguards Bus 1EA2.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this is a Train A Motor Control Center, however, it powers common loads as opposed to the Battery Chargers
B. Incorrect. Plausible because two Motor Control Centers have become de-energized, however, it is the Train B Safeguards Bus that is affected as opposed to Train A.
C. Incorrect. Plausible because this is a Train B Motor Control Center, however, it powers common loads as opposed to the Battery Chargers.
D. Correct. A loss of Safeguards Bus 1EA2 will de-energize 480 V Motor Control Centers 1EB2-1 and 1EB4-1 which power their respective Battery Chargers. As a result, the Battery will become the only source of power and begin to discharge as shown.

Technical Reference(s) SOP-604A, Attachments 5 and 6 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the DC Electrical Distribution System.

Question Source: Bank # X
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, 8
55.43

Comments / Reference: From SOP-604A, Attachment 5	Revision # 12

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-604A
480 VAC SWITCHGEAR and MCCs	REVISION NO. 12 CONTINUOUS USE	PAGE 79 OF 134

ATTACHMENT 5
PAGE 3 OF 10

GUIDELINES FOR DEENERGIZING 1E MCC LOADS

	EQUIPMENT NO.	NOMENCLATURE	NOTE
4.0	Motor Control Center 1EB2-1 (TRAIN B)		
4.1	1EB2-1/1FR/BKR,	125 VDC BATTERY CHARGER BC1D2 SUPPLY BREAKER (SWBD 1D2)	3
4.2	1EB2-1/10D/BKR-1,	480/120-208 VAC LTG TRANSFORMER (1EB2-1/S2) XF-S2 FEEDER BREAKER 1	4
	1EB2-1/10D/BKR-2,	480/120-208 VAC LTG TRANSFORMER (1EB2-1/S2) XF-S2 FEEDER BREAKER 2	4
4.3	1EB2-1/10F/BKR,	125 VDC BATTERY CHARGER BC1ED4-1 SUPPLY BREAKER	3
4.4	1EB2-1/11F/BKR,	125 VDC BATTERY CHARGER BC1ED2-1 SUPPLY BREAKER (SWBD 1ED2)	3
4.5	All other breakers on MCC 1EB2-1 should be OPENED per US direction.		

Comments / Reference: From SOP-604A, Attachment 5	Revision # 12

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-604A
480 VAC SWITCHGEAR and MCCs	REVISION NO. 12 CONTINUOUS USE	PAGE 77 OF 134

ATTACHMENT 5
PAGE 1 OF 10

GUIDELINES FOR DEENERGIZING 1E MCC LOADS

	EQUIPMENT NO.	NOMENCLATURE	NOTE

NOTE: Prior to deenergizing the following 1E MCC loads, EVALUATE each listed breaker for recommended action indicated under the applicable NOTE.

1.0	Motor Control Center 1EB1-1 (TRAIN A)		
1.1	1EB1-1/2M/BKR,	125V DC BATTERY CHARGER BC1ED1-1 SUPPLY BREAKER (SWBD 1ED1)	3
1.2	1EB1-1/9E/BKR-1,	480/277-480 VAC LTG TRANSFORMER (1EB1-1/S1) XF-S1 FEEDER BREAKER 1	4
	1EB1-1/9E/BKR-2,	480/277-480 VAC LTG TRANSFORMER (1EB1-1/S1) XF-S1 FEEDER BREAKER 2	4
1.3	1EB1-1/9G/BKR,	125V DC BATTERY CHARGER BC1ED3-1 SUPPLY BREAKER (SWBD 1ED3)	3
1.4	All other breakers on MCC 1EB1-1 should be OPENED per US direction.		

Comments / Reference: From SOP-604A, Attachment 6	Revision # 12

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-604A
480 VAC SWITCHGEAR and MCCs	REVISION NO. 12 CONTINUOUS USE	PAGE 87 OF 134

ATTACHMENT 6
PAGE 1 OF 6

GUIDELINES FOR DEENERGIZING COMMON 1E MCC LOADS

EQUIPMENT NO. - NOMENCLATURE — NOTE

NOTE: Prior to deenergizing the following common 1E MCC loads, EVALUATE each listed breaker for recommended action indicated under the applicable NOTE.

1.0 Motor Control Center XEB1-1 (TRAIN A)

1.1	XEB1-1/1C/BKR,	480/208-120 LTG XFMR (XEB1-1/EAB9-ECB11) XF-EAB9-ECB11 FDR BREAKER	1
1.2	XEB1-1/1BL/BKR,	480/208-120 TRANSFORMER (XEB1-1/XEC3-3) TXEC3 FEEDER BREAKER 1	2, 3
	XEB1-1/1BR/BKR,	480/208-120 REGULATING TRANSF (XEB1-1/XEC3) TXEC3 FEEDER BREAKER 2	
1.3	XEB1-1/5BL/BKR,	VENTILATION CHILLER X-03 OIL PUMP MOTOR BREAKER	4
1.4	XEB1-1/5F/BKR,	CONTROL ROOM A/C X-02 FAN/COMPRESSOR MOTOR BREAKER	4

Comments / Reference: From SOP-604A, Attachment 6	Revision # 12

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-604A
480 VAC SWITCHGEAR and MCCs	REVISION NO. 12 CONTINUOUS USE	PAGE 88 OF 134

ATTACHMENT 6
PAGE 2 OF 6

GUIDELINES FOR DEENERGIZING COMMON 1E MCC LOADS

	EQUIPMENT NO. - NOMENCLATURE		NOTE
3.0	Motor Control Center XEB1-3 (TRAIN A)		
3.1	XEB1-3/4DL/BKR,	WASTE EVAPORATOR UNIT X-01 FEEDER BREAKER	4
3.2	XEB1-3/4DR/BKR,	BORON RECYCLE EVAPORATOR PKG X-01 CONTROL PANEL SUPPLY BREAKER	4
3.3	All other breakers on MCC XEB1-3 should be OPENED per US direction.		
4.0	Motor Control Center XEB2-1 (TRAIN B)		
4.1	XEB2-1/2BR/BKR,	VENTILATION CHILLER X-04 OIL PUMP MOTOR SUPPLY BREAKER	4
4.2	XEB2-1/2F/BKR,	CONTROL ROOM A/C UNIT X-04 FAN/COMPRESSOR MOTOR BREAKER	4
4.3	XEB2-1/6F/BKR,	CONTROL ROOM VENTILATION S. INTAKE RAD DET 5896B PROCESS SKID SPLY BKR	11
4.4	All other breakers on MCC XEB2-1 should be OPENED per US direction.		

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 A2.18	
Importance Rating	2.6	

Emergency Diesel Generator System: 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 premature opening of breaker under load

Proposed Question: Common 20

Given the following conditions:

- Unit 1 is in MODE 1.
- OPT-214A, Diesel Generator Operability Test, is in progress with Emergency Diesel Generator (EDG) 1-01 paralleled to the grid.
- CS-1EA1-1, INCOMING BKR 1EA1-1 and CS-1EG1, DG BKR 1EG1 are closed.
- With the test in progress a loss of several other units connected to the grid results in grid frequency dropping to 58.6 Hertz.

Which of the following states the required operator action and the reason for that action?

The operator will...

A. Open CS-1EG1, DG BKR 1EG1 to prevent excessive loading on the EDG.

B. Pullout CS-1EA1-2, INCOMING BKR 1EA1-2 and CS-1EA1-1, INCOMING BKR 1EA1-1 to prevent excessive loading on the EDG.

C. Open CS-1EG1, DG BKR 1EG1 to prevent excessive loading on CS-1EA1-1, INCOMING BKR 1EA1-1.

D. Pullout CS-1EA1-2, INCOMING BKR 1EA1-2 and CS-1EA1-1, INCOMING BKR 1EA1-1 to prevent excessive loading on CS-1EA1-1, INCOMING BKR 1EA1-1.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, the output breaker is opened to prevent excessive loading on the EDG.

B. Incorrect. Plausible because preventing excessive loading on the EDG is the proper concern and opening both the alternate and preferred offsite breakers would disconnect the bus from the degrading grid. However, proper procedural guidance is to open the EDG output breaker.

C. Incorrect. Plausible because the output breaker is opened, however, the reason is not to prevent excessive loading on the incoming breaker but to prevent excessive loading on the EDG..

D. Incorrect. Plausible because preventing excessive loading is correct but not excessive loading on the incoming breaker. Opening both the alternate and preferred offsite breakers would disconnect the bus from the degrading grid. However, proper procedural guidance is to open the EDG output breaker.

Technical Reference(s) OPT-214A, Step 8.1.T Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Emergency Diesel Generator System and **PREDICT** the system response.

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 OPT-214A, Step 8.1.T	Revision # 22

CPNPP OPERATIONS TESTING MANUAL	UNIT 1	PROCEDURE NO. OPT-214A
DIESEL GENERATOR OPERABILITY TEST	REVISION NO. 22 CONTINUOUS USE	PAGE 37 OF 147

8.1

NOTE:	**"Continuous Action Step"** This step is a compensatory action for the possibility of excessive loading on the DG due to Offsite Power degradation. The termination criteria of Attachment 10.7, Section II apply to the following step.

T. IF the termination criteria of Attachment 10.7, Section II are met while the DG is synchronized with the offsite power source,
THEN
PERFORM the following:

☐ 1) OPEN CS-1EG1, DG1 BKR 1EG1.

☐ 2) Slowly ADJUST DG voltage to 6900 V (6831 V to 6969 V).

☐ 3) Slowly ADJUST DG frequency to 60.0 (59.7 to 60.3) Hz.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064 K4.10	
Importance Rating	3.5	

Emergency Diesel Generator System: Knowledge of the EDG design feature(s) and/or interlock(s) that provide for the following: Automatic load sequencer: blackout

Proposed Question: Common 21

Given the following conditions:

- Work is in progress to place XST2A, Spare 345 kV Startup Transformer, in service.
- XST2, 345 kV Startup Transformer, is currently tagged out to support the work activities.
- 1EA1, 6.9kV Safeguards Bus, is being supplied from XST1, 138 kV Startup Transformer, via CS-1EA1-2, INCOMING BKR 1EA1-2.

With this configuration, if a sudden pressure fault were to occur on XST1, Bus 1EA1 would be de-energized, Emergency Diesel Generator (EDG) 1-01 would start, CS-1EG1, DG BKR 1EG1 would ...

A. automatically CLOSE, and 1EA1 would have to be manually loaded by the operators.

B. automatically CLOSE, then the Blackout Sequencer would load 1EA1.

C. remain OPEN, may be manually closed, then the Blackout Sequencer would load 1EA1.

D. remain OPEN, may be manually closed, then manually loaded by the operators.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because when 1EA1 is de-energized, the EDG would auto start, the output breaker would automatically close, however, the BOS would automatically load the bus. If the BOS sequencer failed to run, the operator may manually load the bus per ABN-602.

B. Correct. When 1EA1 is de-energized, the EDG would auto start, the output breaker would automatically close and the BOS would automatically start the prescribed loads.

C. Incorrect. Plausible because when 1EA1 is de-energized, the EDG would auto start, the output breaker would automatically close and the BOS would automatically start the prescribed loads. However, if the EDG output breaker were to remain open, the operator may manually close the breaker and then the BOS would load the bus.

D. Incorrect. Plausible because when 1EA1 is de-energized, the EDG would auto start, the output breaker would automatically close and the BOS would automatically start the prescribed loads. However, if the EDG output breaker were to remain open, the operator may manually close the breaker and if the BOS sequencer failed to run, the operator may manually load the bus per ABN-602.

Technical Reference(s) ABN-602, Section 2.2 Attached w/ Revision # See Comments / Reference
ABN-602, Steps 2.3.4 & 2.3.5

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Emergency Diesel Generator System.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 5, 7
55.43 ____

Comments / Reference: From ABN-602, Section 2.2	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 4 OF 107

2.2 Automatic Actions

- Second level undervoltage at 6192 V (6.9 KV bus) or 442.4 V (480 V bus) will trip 6.9 KV safeguard bus preferred and alternate feeds. Attachment 3 lists specific undervoltage response.
- Diesel generators auto-start and energize affected safeguard bus, loading via blackout sequencer. Attachment 2 lists specific sequence.
- Transformer XST2 or XST2A faults isolate 1EA1 and 1EA2 preferred source, causing slow transfer. |
- Transformer 1ST fault momentarily de-energizes XST2 <u>OR</u> XST2A, causing a slow transfer. |
- Transformer XST1 fault isolates 2EA1 and 2EA2 preferred source, causing slow transfer.
- Both affected 6.9 KV safeguard bus offsite feeder breakers open.
- Associated diesel generator starts
- Equipment powered from affected bus de-energizes.

Comments / Reference: From ABN-602, Step 2.3.4	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 7 OF 107

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	5) IF bus needed immediately, THEN GO TO Step 4. 6) IF DG running AND SSW not available, THEN place the affected DG in PULL-OUT to shutdown the DG. ● CS-uDG1E ● CS-uDG2E 7) IF bus lockout actuated AND bus NOT needed immediately, THEN restore affected bus per Attachment 4 AND GO TO Step 5.

NOTE:
- Fully charged DG starting air receivers have sufficient air pressure for approximately five (5) start attempts.
- Following shutdown, a time delay associated with pneumatic logic board will prevent normal starts for approximately two minutes, but may be overridden with an emergency start.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 Restore power to affected 6.9 KV safeguard bus:	
☐ a. Verify affected DG - RUNNING	a. IF not previously performed, THEN start DG as follows:

Comments / Reference: From ABN-602, Step 2.3.4	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 8 OF 107

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 ☐ b. Check affected bus DG supply breaker - CLOSED	b. Perform the following for de-energized bus: 1) Turn synchroscope on AND manually close selected DG breaker. 2) Turn synchroscope OFF. 3) IF supply breaker still NOT closed AND not previously performed, THEN emergency start DG. 4) IF supply breaker still NOT closed, THEN place the affected DG in PULL-OUT to shutdown the DG. 5) IF DG will NOT stop remotely, THEN locally emergency stop DG. 6) IF breaker NOT closed, THEN restore affected bus per Attachment 4 WHILE continuing with Step 5.
☐ c. Check affected DG voltage, 6500 - 7100 volts	d. Perform the following:

Comments / Reference: From ABN-602, Step 2.3.4	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 9 OF 107

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4 ☐ d. Check affected DG frequency, 59.5 - 60.5 Hz	c. Perform the following: 1) Adjust DG frequency to 60 Hz (59.5 - 60.5 Hz). 2) IF frequency can NOT be maintained greater than 59 Hz, THEN place the affected DG in PULL-OUT to shutdown the DG. 3) IF DG will NOT stop remotely, THEN locally emergency stop DG. 4) IF DG STOPPED, THEN locally restore DG per Attachment 6, WHILE continuing with Step 5.
e. Re-energize BOS, if previously de-energized by Attachment 4 by closing the breaker in the bottom of the Sequencer Cabinet for the affected bus.	

Comments / Reference: From ABN-602, Step 2.3.5	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 10 OF 107

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5 Monitor blackout sequencer status:	
☐ a. Affected bus - ENERGIZED	a. Ensure all affected equipment - PULL OUT (Use Attachment 4, Step 6 for guidance, if necessary)
☐ b. Verify Blackout Sequencer - OPERATED • OUTPUT-STEP TIME lights - ALL LIT OR • Automatic lockouts AL light - LIT	b. Align necessary equipment to unaffected train, using Attachment 2 for guidance.
☐ c. Verify all required equipment actuated per Attachment 2.	c. Align equipment, as necessary.
☐ d. Recover from blackout sequencer operation per Section 8 while continuing.	

Comments / Reference: From CPNPP Exam Bank	Revision # 01/23/00

Given the following conditions:

- Work is in progress to place XST2A, Spare 345 kV Startup Transformer, in service.
- XST2, 345 kV Startup Transformer, is currently de-energized to support the work activities.
- 1EA1, 6.9kV Safeguards Bus, is being supplied from XST1, 345 kV Startup Transformer, via 1EA1-2, 6.9 kV Incoming Breaker.

With this configuration, if a sudden pressure fault were to occur on XST1, Bus 1EA1 would be load shed, Emergency Diesel Generator (EDG) 1-01 would start, 1EG1, EDG 1-01 Output Breaker would ...

A. automatically close, and the Blackout Sequencer (BOS) would load Bus 1EA1.

B. automatically close, but Bus 1EA1 would have to be manually loaded by the operators.

C. remain open but could be manually closed by the operators, and Bus 1EA1 would have to be manually loaded by the operators.

D. remain open and could **NOT** be manually closed by the operators, and Bus 1EA1 would remain de-energized.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073 K3.01	
	Importance Rating	3.6	

Process Radiation Monitoring System: Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases

Proposed Question: Common 22

Given the following conditions on Unit 1:

- A PC-11 High Radiation alarm was just received on X-RE-5251A (ABP074) LVW/EVAP POND VNT & DRN HDR RADIATION DETECTOR.

Which of the following describes any automatic actions that should occur and operator indications in the Control Room to verify these actions?

A. The Condensate Backwash Recovery Pump will stop and the Isolation Valve should automatically close stopping flow to the Low Volume Waste System. Sample flow to the radiation monitor will stop.

B. The Turbine Building Sumps will re-align from the Co-Current Waste System to the Low Volume Waste System. Valve position status is available on the Plant Computer and an annunciator alarm alerts the operator that this action occurred.

C. The Auxiliary Building drains will re-align from the Low Volume Waste System to the Co-Current Waste System. Valve position status is available on the Plant Computer and a Computer alarm alerts the operator that this action occurred.

D. The Turbine Building Sump Pumps will stop and the flow isolation valve will automatically close stopping flow to the Low Volume Waste System. Sample flow to the radiation monitor will stop.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this flow goes to the Low Volume Waste System (LVW) and it could be thought this path was monitored by this Radiation Monitor, however, this is an unmonitored path to the LVW.
B. Incorrect. Plausible because this flow goes to the Low Volume Waste System (LVW) and it could be thought this path was monitored by this Radiation Monitor, however, this path is monitored by 1-RE-5100, Turbine Building Sump 1-02 Radiation Monitor.
C. Correct. This path from the Auxiliary Building is monitored by this Radiation Monitor and these indications are the only ones available in the Control Room.
D. Incorrect. Plausible because this path is monitored by a Radiation Monitor, however, it is 1-RE-5100, Turbine Building Sump 1-02 Radiation Monitor, and the automatic actions from this monitor closes the path to LVW and opens the path to Co-Current Waste.

Technical Reference(s) ABN-903, Steps 2.1.B & 2.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operations of the Liquid Waste Systems.
STATE the functions and **EXPLAIN** the design criteria of the Waste Management System.

Question Source:	Bank #	X	
	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	11, 13
	55.43	

Comments / Reference: From ABN-903, Steps 2.1.B & 2.2	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-903
ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID	REVISION NO. 6	PAGE 3 OF 13

2.0 ACCIDENTAL RELEASE OF RADIOACTIVE LIQUID

2.1 Symptoms

a. Annunciator Alarms

Main Control Board

- LWPS PNL TRBL (6B-4.7)

LWPS PANEL

- LWPS EFFLUENT MONITOR ALERT (2.6)

b. Plant Indications

1) An unexpected increase in any of the following liquid process effluent monitors:

- X-RE-5251A (ABP074) LVW/EVAP POND VNT & DRN HDR RADIATION DETECTOR 5251A
- X-RE-5253 (LWE076) LIQUID WASTE PROCESSING DISCHARGE RADIATION DETECTOR
- u-RE-4269 (SSWu65) UNIT u STATION SERVICE WATER TRAIN A TO DISCH CANAL RAD DETECTOR
- u-RE-4270 (SSWu66) UNIT u STATION SERVICE WATER TRAIN B TO DISCH CANAL RAD DETECTOR
- 1-RE-5100 (TBD172) TURBINE BUILDING SUMP 1-02 RADIATION DETECTOR
- 2-RE-5100 (TBD272) TURBINE BUILDING SUMP 2-04 RADIATION DETECTOR

2) Waste Water Hold-up Tank or piping leak or spill reported by Plant Personnel.

2.2 Automatic Actions

a. A High Alarm on X-RE-5251A will realign sump discharge from the LVW system to the COW system.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	073 G 2.4.6	
Importance Rating	3.7	

Process Radiation Monitoring System: Emergency Procedures/Plan: Knowledge of EOP mitigation strategies

Proposed Question: Common 23

Given the following conditions:

- Unit 1 is at 100% power.
- Component Cooling Water (CCW) Non-Safeguards Loop Radiation Monitor (CCW-168/1-RE-4510) goes into ALERT.
- Component Cooling Water Surge Tank level is rising.

Which of the following components should be isolated in response to the Radiation Monitor alarm and rising CCW Surge Tank level?

A. Excess Letdown Heat Exchanger

B. Letdown Heat Exchanger

C. Seal Water Heat Exchanger

D. BTRS Letdown Chiller Heat Exchanger

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because RCS Excess Letdown is at a higher pressure than CCW pressure but is not aligned.
B. Correct. RCS Letdown is at a higher pressure than CCW pressure.
C. Incorrect. Plausible because CCW cools the Seal Water Heat Exchanger but is at a lower pressure than CCW.
D. Incorrect. Plausible because RCS flowing through the BTRS Letdown Chiller Heat Exchanger is at a higher pressure than CCW pressure but is not aligned.

Technical Reference(s): ABN-502, Step 4.1.b; ABN-502, Step 4.3.15 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Component Cooling Water System.
EXPLAIN the instrumentation and controls of the Digital Radiation Monitoring System and **PREDICT** the system response.

Question Source: Bank # X
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, 11
55.43 ______

Comments / Reference: From ABN-502, Step 4.1.b	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 18 OF 75

4.0 LEAKAGE INTO THE CCW SYSTEM

4.1 Symptoms

a. Annunciator Alarms

- CCW SRG TK TRN A LVL HI-HI/LO (3B-2.4)
- CCW SRG TK TRN B LVL HI-HI/LO (3B-3.4)
- ANY RCP THBR CLR CCW RET TEMP HI (3B-2.11)
- ANY RCP THBR CLR CCW RET FLO HI (3B-4.11)

b. Plant Indications

- CCW Surge Tank level increasing without make-up
- RCP thermal barrier CCW return temperature increase
- RCP thermal barrier CCW return flow increase
- CCW activity increasing on any of the following monitors:
- uRE-4509, (CCWu67) Unit u COMPONENT COOLING WATER TRAIN A RADIATION DETECTOR
- uRE-4510, (CCWu68) Unit u COMPONENT COOLING WATER NON-SFGD RADIATION DETECTOR
- uRE-4511, (CCWu69) Unit u COMPONENT COOLING WATER TRAIN B RADIATION DETECTOR

Comments / Reference: From ABN-502, Step 4.3.15	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 27 OF 75

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 14 Verify CCW Surge Tank Level - NOT INCREASING: • u-LI-4500, CCW SRG TK LVL • u-LI-4501, CCW SRG TK VLV	Perform the following: a. Start the idle RHR Pump. b. Shutdown the opposite train of RHR. c. Close the train associated RHR HX CCW RET VLV: • u-HS-4572, RHR HX 1 CCW RET VLV • u-HS-4573, RHR HX 2 CCW RET VLV
15 Isolate the affected component: ☐ a. Close the CCW supply isolation valve to the affected component. ☐ b. Close the CCW return isolation valve to the affected component. ☐ c. Close the heat exchanger tube side supply isolation valve to the affected component. ☐ d. Close the heat exchanger tube side return isolation valve to the affected component. ☐ e. Depressurize the affected heat exchanger. ☐ f. Monitor temperature of isolated component.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 A4.04	
	Importance Rating	3.5	

Service Water System: Ability to manually operate and/or monitor in the control room: Emergency heat loads

Proposed Question: Common 24

Given the following conditions:

- Unit 1 is operating at 100% power.
- Station Service Water Pump 1-01 trips.
- The crew is securing affected components per ABN-501, Station Service Water System Malfunction.

Which of the following components are secured using ABN-501, Station Service Water System Malfunction?

1. Centrifugal Charging Pump 1-01
2. Emergency Diesel Generator 1-01
3. Component Cooling Water Pump 1-01
4. Containment Spray Pumps 1-01 & 1-03
5. Control Room Air Conditioners 1-01 & 1-03
6. Uninterruptable Power Supply HVAC Unit 1-01
7. Safety Injection Pump 1-01
8. Safety Chiller 1-05

A. 1, 5, 6, 8

B. 2, 3, 4, 5

C. 2, 4, 6, 7

D. 1, 2, 4, 7

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because CCP 1-01 must be secured and CRACs 1-01 & 1-03, UPS HVAC Unit 1-01 and Safety Chiller 1-05 are secured if CCWP 1-01 trips.

B. Incorrect. Plausible because EDG 1-01 and CTPs 1-01 & 1-03 must be secured, CCWP 1-01 may be secured if HXR outlet temperature exceeds 122°F and CRACs 1-01 & 1-03 secured if CCWP 1-01 trips.

C. Incorrect. Plausible because EDG 1-01, CTP 1-01 & 1-03 and SIP 1-01 and UPS HVAC Unit 1-01 are secured if CCWP 1-01 trips.

D. Correct. CCP 1-01, EDG 1-01, CTPs 1-01 & 1-03 and SIP 1-01 must be secured.

Technical Reference(s) ABN-501, Steps 2.3.1 & 2.3.5 — Attached w/ Revision # See Comments / Reference
ABN-502, Step 2.3.6

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Station Service Water Pump Trip per ABN-501, Station Service Water System Malfunction.

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 ABN-501, Step 2.3.1	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
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2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- The diesel generator can be operated, with load, for approximately one minute without SSW flow and not affect diesel performance.
- When a fault exists on the 6.9KV safeguard bus, the SSW pump will not be available to supply cooling water to the DG.
- Diamond step 1 denotes Initial Operator Actions.

☐ ◇1 Place affected train diesel generator handswitch, CS-uDGuE (emergency stop/start) in PULLOUT.

Comments / Reference: From ABN-502, Step 2.3.5	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 5 OF 50

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: The CCW pump on the affected train may be left operating at the discretion of the Shift Manager. However, with this pump operating, the affected SSW Pump will have an Auto Start Signal to it.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 4 Verify equipment in the affected train - NOT REQUIRED FOR OPERATION: • CCP • Diesel Generator • CCW Pump • SI Pump • Containment Spray Pumps	Start equipment in the unaffected train as required to support Plant Operations: • CCP • Diesel Generator • CCW Pump • SI Pump • Containment Spray Pumps

CAUTION: Do not place pump handswitch in STOP if pump tripped (white TRIP light). This will reset 86M relay (white TRIP light) and may result in an automatic restart.

5 Shutdown equipment in the affected Train as follows:

- ☐ CCP - PULL OUT
- ☐ SI Pump - PULL OUT
- ☐ Containment Spray Pumps - PULL OUT
- ☐ SSW Pump - PULL OUT

Comments / Reference: From ABN-502, Step 2.3.6	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 5 OF 75

2.3 <u>Operator Actions</u>

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 5 Verify required equipment, for existing conditions, supplied by unaffected train - IN OPERATION: • Control Room A/C Units • Containment Spray System • UPS HVAC Unit • Excess Letdown • RHR System	Align <u>AND</u> start required equipment as necessary. <u>IF</u> CCW flow is <u>NOT</u> available to an operating RHR HX, <u>THEN</u> perform the following: a. Verify safeguard loop isolation valves open (<u>u</u>-HS-4512, 4513, 4514, and 4515) b. Ensure the idle CCW Pump Recirc Valve closed. • <u>u</u>-HS-4536, CCWP 1 RECIRC VLV • <u>u</u>-HS-4537, CCWP 2 RECIRC VLV c. Align RHR HX return valves as necessary to supply the operating heat exchanger. • <u>u</u>-HS-4572, RHR HX 1 CCW RET VLV • <u>u</u>-HS-4573, RHR HX 2 CCW RET VLV d. Ensure CCW flow requirements of Step 4 are maintained.
☐ 6 Stop equipment on the affected train as necessary: • Control Room A/C Units • Containment Spray System • UPS HVAC Unit • RHR System • Safety Chiller Recirc Pump	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	076 K1.15	
Importance Rating	2.5	

Service Water System: Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: FPS

Proposed Question: Common 25

Given the following conditions on Unit 1:

- Unit 1 is tripped due to a normal plant shutdown.
- Station Service Water Pump 1-01 was tagged out 1 hour ago due to the pump bearing overheating.
- Station Service Water Pump 1-02 tripped on overcurrent.
- During the recovery actions it becomes necessary to supply cooling water to the Unit 1 Safety Injection Pumps, Containment Spray Pumps, and Centrifugal Charging Pumps.

Which of the following answers completes the statement below per ABN-501, Station Service Water Malfunctions?

IF [1] can NOT supply Unit 1 Station Service Water, THEN supply cooling water to essential equipment from [2].

	[1]	[2]
A.	Demineralized Water;	Unit 2 Station Service Water
B.	Unit 2 Station Service Water;	Fire Protection Water
C.	Demineralized Water;	Fire Protection Water
D.	Unit 2 Station Service Water;	Demineralized Water

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Unit 2 Station Service Water is the preferred source, however, demineralized water is insufficient to supply the required loads.
B. Correct. As outlined in ABN-501.
C. Incorrect. Plausible because fire protection water is the secondary source, however, demineralized water is insufficient to supply the required loads
D. Incorrect. Plausible because Unit 2 Station Service Water is the preferred source, however, demineralized water is insufficient to supply the required loads.

Technical Reference(s) ABN-501, Step 5.3.7 Attached w/ Revision # See
ABN-501, Attachment 1 Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Station Service Water Pump Trip per ABN-501, Station Service Water System Malfunction.

Question Source: Bank # X
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, 8
55.43

Comments / Reference: From ABN-501, Step 5.3.7	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 23 OF 50

5.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Cross connecting Station Service Water between units will render cross connected trains of BOTH units INOPERABLE in MODE 1, 2, 3, or 4.
-0R-
Cross connecting Station Service Water between Trains within a unit will render BOTH trains INOPERABLE in MODE 1, 2, 3, or 4.

NOTE: IF barriers designated as Fire or Security Barriers, such as manways, doors, hatchcovers, slabs, etc. are to be breached, THEN the Shift Manager and Security shall be notified and approval obtained prior to affecting the breach.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7 Restore SSW cooling flow by CROSS-TIEING Train A AND Train B as follows:	Perform the following: 1) Cross-tie SSW between units per SOP-501A/B, as directed.
☐ a. Verify at least one SSW Train - AVAILABLE.	2) Refer to TS 3.7.8

Comments / Reference: From ABN-501, Attachment 1	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
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ATTACHMENT 1
PAGE 1 of 5

FIRE PROTECTION WATER ALIGNMENT TO DIESEL GENERATORS

This contingency method is intended to be used when other means of responding within design capability to the current situation are unavailable. The intent is to supply enough cooling to support one MDAFWP, one CCP, and one battery charger. Any loading greater than this must be considered on a case by case basis.

The limitation is in cooling water supply which must be carefully monitored. If installed diesel driven fire pumps are being used, maximum flows are expected on the order of 2,000 gpm per pump. One pump should supply sufficient water for all four diesels (~250 gpm), however, it may also be necessary to supply makeup to the SFPs. Any running fire pump will also require a makeup source of fuel oil; minimum level in the tank holds about two hours worth of fuel.

Multiple cross connections between SSW and Fire Protection systems are made in this attachment. Because SSW is designed to operate at a lower pressure than FP, prior to initiating FP water flow to the SSW system a flow path to atmosphere (the lake or an open drain valve) must be verified.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078 K2.01	
	Importance Rating	2.7	

Instrument Air System: Knowledge of bus power supplies to the following: Instrument air compressor

Proposed Question: Common 26

Given the following conditions:

- Both Units are operating at full power.
- Instrument Air (IA) System is aligned as follows:
  - IA Compressors 1-01 & 2-02 are running, supplying their respective system demand.
  - IA Compressors 1-02 & 2-01 are in BACKUP with their handswitches in AUTO.
  - IA Compressor X-01 is in STANDBY, aligned to Unit 1.
  - IA Compressor X-02 is in STANDBY, aligned to Unit 2.
- A Large Break Loss of Coolant Accident occurs on Unit 1.
- All systems function automatically as designed.
- The following Annunciators are in alarm:
  - 1-ALB-1, Window 2.1 – INSTR AIR COMPR 1/2 TRIP.
  - 1-ALB-1, Window 2.4 – CNTMT INSTR AIR HDR PRESS LO.
  - 1-ALB-1, Window 4.1 – COMMON INSTR AIR DRYER PNL TRBL.

Which of the following describes the final status of the Instrument Air Compressors on both Units?

A. 1-01 and 2-02 are in operation.

B. 2-02 and X-01 are in operation.

C. Only X-02 is in operation.

D. Only X-01 is in operation.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because IA Compressor 2-02 is operating, however, 1-01 would not be running for the conditions listed.
B. Correct. Given the conditions listed, these are the IA Compressors running.
C. Incorrect. Plausible if thought SI affected both units, however, X-02 would not be running for the conditions listed.
D. Incorrect. Plausible because IA Compressor X-01 is operating, however, IA Compressor 2-02 would also be operating.

Technical Reference(s) LO21.SYS.IA1.LN, Page 32 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Instrument Air System including interrelations with other systems to include interlocks and control loops.

Question Source:		
Bank #	X	
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 LO21.SYS.IA1.LN, Page 32	Revision # 05/07/11

SAFETY INJECTION (FIGURE 18)

The X-01/X-02 Compressors are unaffected by a unit-related Safety Injection signal being processed.

The 1-01/1-02 and 2-01/2-02 Compressors breakers on uEB3 and uEB4 are shunt tripped by a safety injection signal via u-KXA-0147A/u-KXA-0147B which will shutdown the compressors. An open contact will electrically prevent reclosing the breaker until the SI signal has been reset and the breaker is reset locally. The 1-01 and 1-02 Air Dryers control power breakers are also tripped in the event of a Safety Injection signal.

The 2-01 Air dryer control power breaker are load shed by 2-K615A. To reclose the breaker and restart the Safety Injection signal must be reset.

The 2-02 Air dryer control power breaker are load shed by 2-K639B. To reclose the breaker and restart the Safety Injection signal must be reset.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	078 K3.03	
Importance Rating	3.0	

Instrument Air System: Knowledge of the effect that a loss or malfunction of the IAS will have on the following: Cross-tied units

Proposed Question: Common 27

Given the following conditions:

- In response an instrument air malfunction Unit 1 and Unit 2 instrument air headers are cross-connected per SOP-509A, Instrument Air System.
- Instrument Air compressors 1-01 and 1-02 are not operable and instrument air compressor X-01 is aligned to Unit 1 running at reduced capacity.
- The air leak has NOT been found.
- Instrument Air compressors 2-01, 2-02 and X-02 are all aligned to Unit 2 as follows;
  - Instrument Air Compressor 2-01 – BACKUP
  - Instrument Air Compressor 2-02 – LEAD
  - Instrument Air Compressor X-02 – STBY
- 2-CI-0050, INST AIR RCVR 2-01 U1 XTIE VLV and 1-CI-0051, INST AIR RCVR 1-01 U2 XTIE VLV are open.
- Instrument Air header pressure dropped to 83 psig and rose to 102 psig and has been stable for 25 minutes.

Which of the following describes the status of the Unit 2 Instrument Air compressors?

Instrument Air compressor 2-01 is __________________, Instrument Air compressor 2-02 is __________________, and Instrument Air compressor X-02 is __________________.

A.	running loaded	running loaded	running loaded
B.	running unloaded	running unloaded	running unloaded
C.	running loaded	running unloaded	running unloaded
D.	running loaded	running unloaded	not running

Proposed Answer: A

Explanation:

A. Incorrect. Plausible because IA Compressor 2-02 is operating, however, 1-01 would not be running for the conditions listed.
B. Correct. Given the conditions listed, these are the IA Compressors running.
C. Incorrect. Plausible if thought SI affected both units, however, X-02 would not be running for the conditions listed.
D. Incorrect. Plausible because IA Compressor X-01 is operating, however, IA Compressor 2-02 would also be operating.

Technical Reference(s) LO21.SYS.IA1.LN, Page 13 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Instrument Air 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, 7
55.43

Comments / Reference: From LO21.SYS.IA1.LN, Page 13	Revision # 05/07/11

Lead Setpoints

- 105 PSIG Loads
- 115 PSIG Unloads
- 20 Second Delay on Start of Compressor to load
- Auto Shutdown if Running Unloaded > 20 Minutes (1-01 or 1-02 only)

Backup Setpoints

- 100 PSIG Loads (Common IACs load 95 psig)
- 115 PSIG Unloads
- 20 Second Delay on Start of Compressor to load
- Auto Shutdown if Running Unloaded > 20 Minutes

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	103 G 2.2.42	
Importance Rating	3.9	

Containment System: Equipment Control: Ability to recognize system parameters that are entry- level conditions for Technical Specifications

Proposed Question: Common 28

Which of the following would cause entry into the Containment Isolation Valves Limiting Condition for Operation while in MODE 1?

Damage to the ______________________, which will NOT allow the valve to close.

A. stem for 1-HV-3487, CNTMT INST AIR ISOL VLV

B. operator for 1-LCV-459, U1 LTDN ISOL VLV

C. stem for 1MS-0357, SG 1-03 BLDN DNSTRM ISOL VLV

D. operator for 1-HV-2134, SG 1-01 FW ISOL VLV

Proposed Answer: A

Explanation:

A. Correct. This valve is a Containment Penetration boundary outside of Containment.
B. Incorrect. Plausible because this valve is part of the CVCS System, however, there are 2 automatic downstream isolation valves (1/1-8160 & 8152) providing Containment Isolation.
C. Incorrect. Plausible because this valve is part of the Blowdown System, however, there are 2 automatic upstream isolation valves (1-HV-2399 & 2399A) providing Containment Isolation.
D. Incorrect. Plausible because this valve looks like a Containment Isolation Valve but is excluded from Technical Specification LCO 3.6.3 in NOTE “above the line”.

Technical Reference(s):
Technical Specification LCO 3.6.3
Flow Diagram M1-0202, Sheet 02
Flow Diagram M1-0203, Sheet 01
Flow Diagram M1-0253, Sheet A
Flow Diagram M1-0216, Sheet A

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Containment System including Technical Specifications, TRM and ODCM.

Question Source: Bank # X

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 9

55.43 ______

Comments / Reference: From Technical Specification LCO 3.6.3	Amendment # 150

Containment Isolation Valves
3.6.3

3.6 CONTAINMENT SYSTEMS

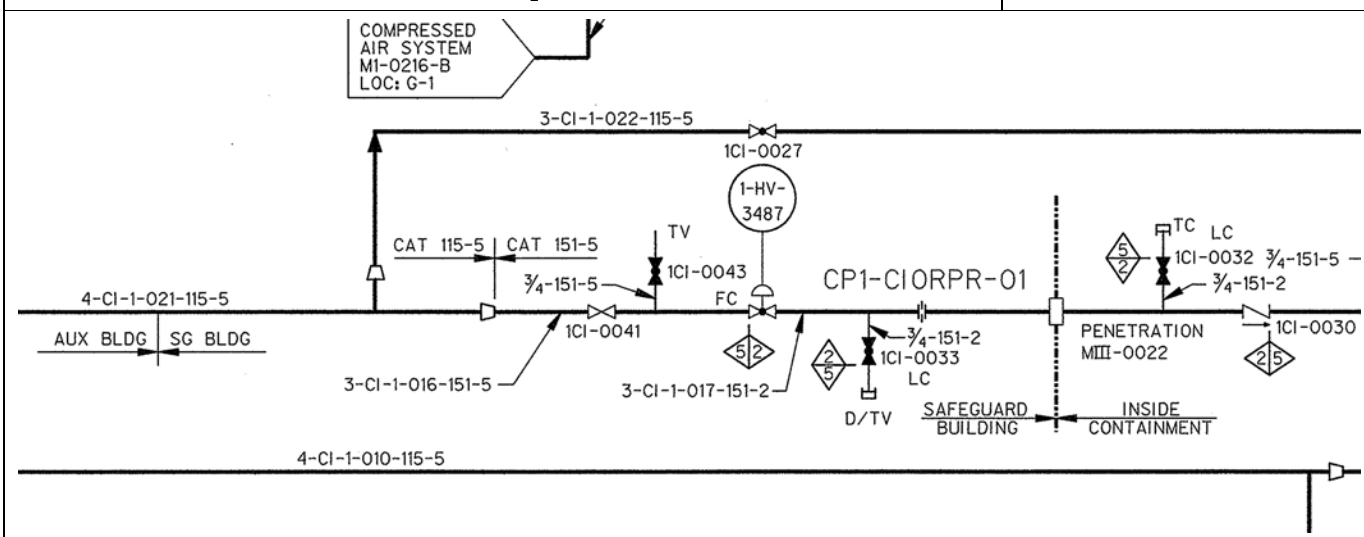
3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

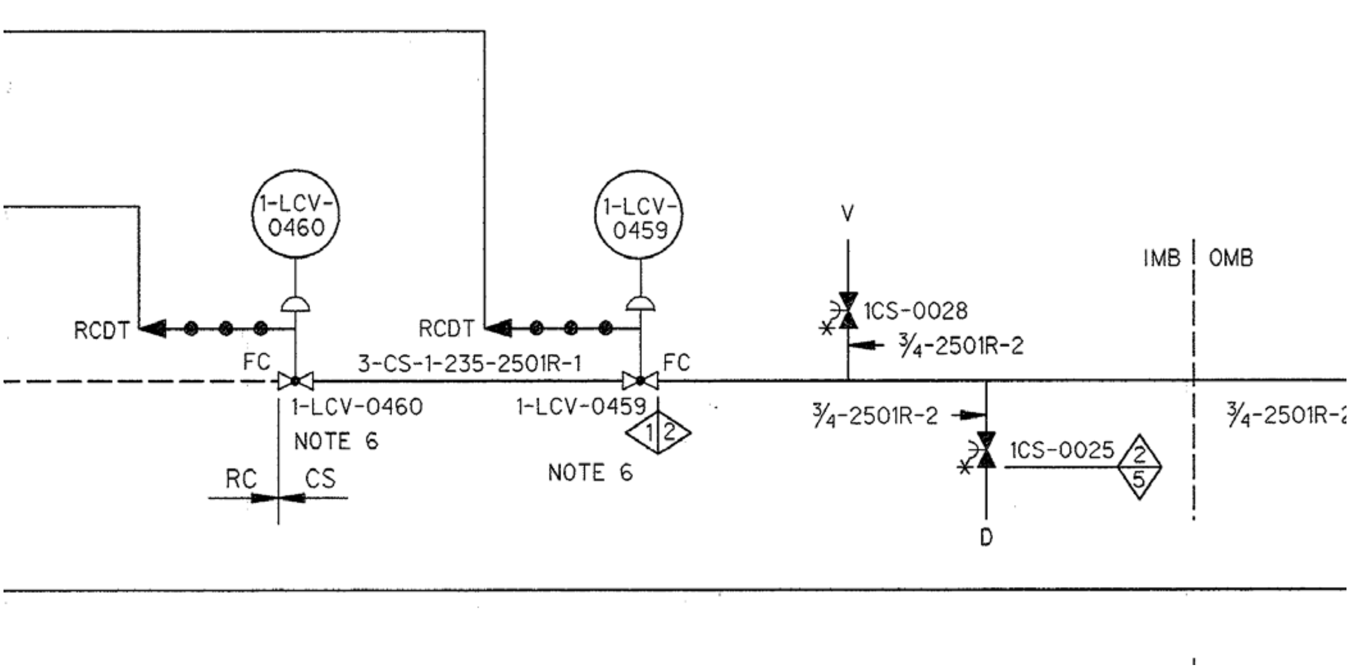
---NOTE---
Not applicable to Main Steam Safety Valves (MSSVs), Main Steam Isolation Valves (MSIVs), Feedwater Isolation Valves (FIVs) and Associated Bypass Valves, and Steam Generator Atmospheric Relief Valves (ARVs).

APPLICABILITY: MODES 1, 2, 3, and 4

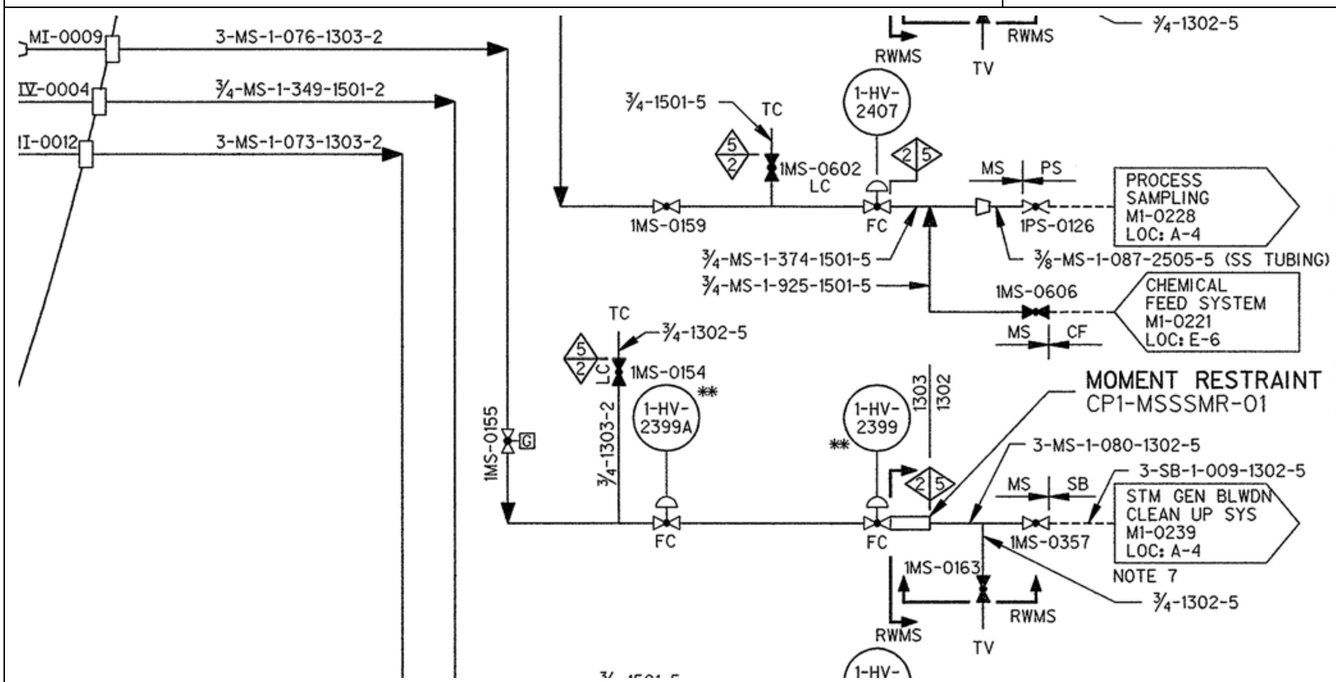
Comments / Reference: From Flow Diagram M1-0216, Sheet A	Revision # CP-41



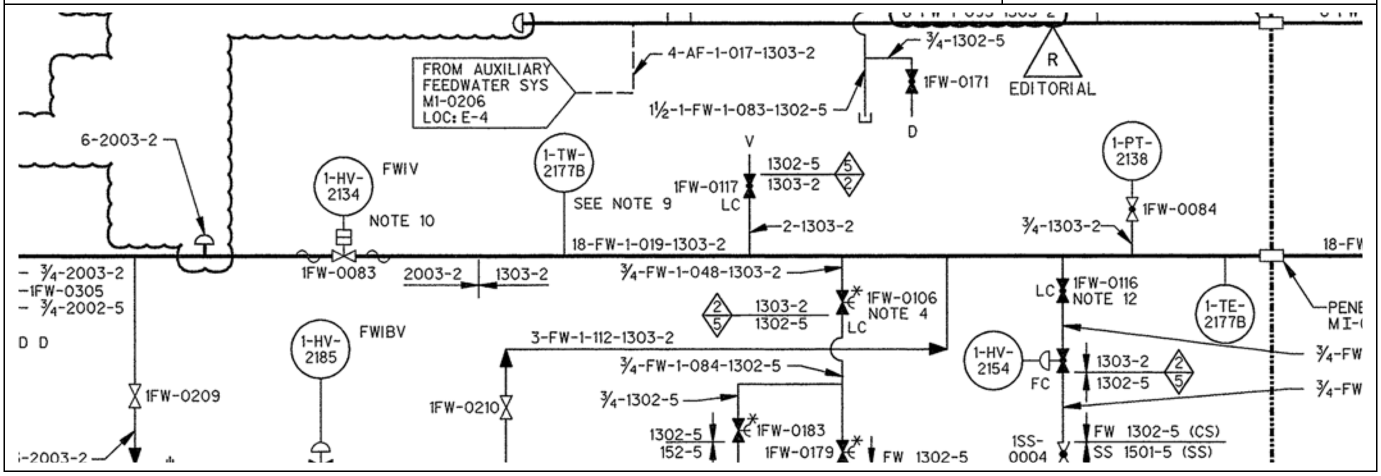
Comments / Reference: From Flow Diagram M1-0253, Sheet A	Revision # CP-10



Comments / Reference: From Flow Diagram M1-0202, Sheet 02	Revision # CP-21



Comments / Reference: From Flow Diagram M1-0203, Sheet 01	Revision # CP-25



Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	001 K6.12	
Importance Rating	2.9	

Control Rod Drive System: Knowledge of the effect that a loss or malfunction of the following will have on the CRDS: Location and interpretation of CRDS AC/DC status alarms

Proposed Question: Common 29

Given the following conditions:

- Unit 1 is at 100% power.
- Annunciator 1-ALB-10B, Window 4.12 – 480V ANY NON-1E BUS VOLT LOSS is in alarm.
- The following conditions are report by the Field Support Supervisor:
  - CV-1MG1 & CV-1MG2, GROUND PROTECTION RELAY flags are NOT actuated.
  - CRDM Generator 1-01 Motor Breaker is closed.
  - CRDM Generator 1-02 Motor Breaker is open.
  - Both MG 1 and 2 DIRECTIONAL OVERCURRENT A and B relays are NOT actuated.
  - MG GENERATOR OVERVOLTAGE TRIP light NOT lit.
  - CRDM Generator 1-01 Generator Breaker is closed.
  - CRDM Generator 1-02 Generator Breaker is closed.
  - CRDM GENERATOR LINE VOLTS are 261 volts and stable.

Which of the following describes the status of the Control Rod Drive System?

The Reactor is [1] and CRDM Generator [2] is tripped due to the loss of 480 Volt Bus [3].

	[1]	[2]	[3]
A.	tripped	1-01	1B4
B.	not tripped	1-02	1B1
C.	tripped	1-01	1B1
D.	not tripped	1-02	1B4

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because 480 V Bus 1B4 was lost, however, the Reactor remains online.
B. Incorrect. Plausible because the Reactor is not tripped and CRDM Generator 1-02 is tripped, however, the loss is due to 480 V Bus 1B4.
C. Incorrect. Plausible if thought that indications provided resulted in this condition.
D. Correct. Given the conditions listed, the Reactor is not tripped but CRDM Generator 1-02 is lost with Bus 1B4 de-energized.

Technical Reference(s): LO21.SYS.CR1.LN, Page 24; ALM-0102A, 1-ALB-10B, Window 4.12 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Rod Control System and **PREDICT** the system response. **COMPREHEND** the normal, abnormal and emergency operation of the Rod 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, 6
55.43 ______

Comments / Reference: From LO21.SYS.CR1.LN, Page 24	Revision # 05/02/11

MOTOR GENERATORS (MGS)

Power for the CRDMs is supplied by two motor-generator sets. They are each comprised of a three phase induction motor directly coupled to a solid steel flywheel and a synchronous alternator. Each MG is operating from separate 480-volt, three-phase buses (uB3 and uB4). The generators are paralleled through Westinghouse type DB-416 circuit breakers. Each generator is a synchronous type, rated at 438 KVA, 260 VAC phase to phase, 150 VAC phase to neutral, zig-zag-wye connected, with brushless excitation from a static voltage regulator and 58.5-59.7 Hertz. The generator output breakers are in the bottom of the MG set switchgear panel. Control switches are on the panel and an automatic synchronizing circuit is installed inside.

Comments / Reference: From LO21.SYS.CR1.LN, Page 25	Revision # 05/02/11

Motor Generator Set Control Panel

These panels are located in the Safeguards Building 832' level near the Rod Control Logic and Power Cabinets. Each houses the associated Motor Generator output breakers. The associated controls for the generator output breaker as well as the **motor generator** motor breaker are also on the MG set panel. When both output breakers are open, either will close when operated by their respective handswitch. The second must always be closed by the synchronizer. Either breaker may be tripped by its control switch, overexcitation, or by phase overcurrent.

Comments / Reference: From ALM-0102A, 1-ALB-10B, Window 4.12	Revision # 12

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0102A
ALARM PROCEDURE 1-ALB-10B	REVISION NO. 12	PAGE 330 OF 366

ANNUNCIATOR NOM./NO.: **480V ANY NON-1E BUS VOLT LOSS** **4.12**

PROBABLE CAUSE:

Overcurrent trip on 480V bus supply OR bus tie breaker
Overcurrent trip on 6.9KV transformer supply breaker
Loss of associated 6.9KV bus

AUTOMATIC ACTIONS:

Trips the following component supply breakers on affected bus.

1B1	1B2	1B3	1B4
A CEV	B CEV	BTRS CHILLER	C CEV
A EHC	B EHC	SLOP	C EHC
A ALOP	B ALOP	C ALOP	CRDM MG 2
		CRDM MG 1	IAC X-01 (If selected)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	011 A1.03	
	Importance Rating	2.8	

Pressurizer Level Control System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PZR LCS system controls including: VCT level

Proposed Question: Common 30

Given the following conditions:

- Unit 1 is at 100% power.
- Centrifugal Charging Pump (CCP) 1-01 is running.
- Letdown is established at 120 gpm.
- 1-LCV-112A, VCT LVL CTRL VLV, is in AUTO position.
- Actual Volume Control Tank (VCT) level is 58% and lowering.
- VCT level channel 1-LT-112 fails high.

Which of the following describes VCT level response?

The Reactor Makeup System will...

A. ...initiate automatic makeup at 46% and cause level to cycle between 46% and 56% with CCP 1-01 maintaining suction.

B. ...initiate automatic makeup at 46% and cause level to cycle between 46% and 58% with CCP 1-01 maintaining suction.

C. ...NOT initiate automatic makeup and CCP 1-01 suction will automatically shift to the Refueling Water Storage Tank.

D. ...NOT initiate automatic makeup and CCP 1-01 suction will NOT automatically shift to the Refueling Water Storage Tank.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because automatic makeup would be initiated when both level channels lowered to 46% and the cycling setpoint is correct, however, 1-LT-112 will continue to indicate 100% VCT level. Even though 1-LCV-112A, VCT LVL CTRL VLV shifts to the Reactor Holdup Tank automatic makeup will not initiate because both channels must indicate 46%.

B. Incorrect. Plausible because automatic makeup would be initiated when both level channels lowered to 46%, however, the cycling setpoint is incorrect and 1-LT-112 will continue to indicate 100% VCT level.

C. Incorrect. Plausible because automatic makeup will not be initiated, however, the Centrifugal Charging Pump suction remains aligned to the VCT.

D. Correct. As outlined in ALM-0061A, VCT level is above the setpoint for automatic makeup and suction will not shift to the RWST until a level of 2% is reached in the VCT on both channels.

Technical Reference(s) ALM-0061A, 1-ALB-6A, Window 2.5 & 4.5 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Chemical and Volume Control System including interrelations with other systems to include interlocks and control loops.

Question Source: Bank # X
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 ALM-0061A, 1-ALB-6A, Window 2.5	Revision # 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 37 OF 79

ANNUNCIATOR NOM./NO.: **VCT LVL HI HI** **2.5**

PROBABLE CAUSE:

1-LT-0185, CVCS VOLUME CONTROL TANK 1-01 LEVEL TRANSMITTER 0185 malfunction
1-LT-0112, CVCS VOLUME CONTROL TANK 1-01 LEVEL TRANSMITTER 0112 malfunction
1-LK-112C, VCT LVL CTRL malfunction

NOTE: 1/1-LCV-112A fails to the VCT position on loss of air or power. 1-8120, VCT 1-01 TO REHUT RLF VLV set pressure is 75 psig.

AUTOMATIC ACTIONS:

IF in AUTO, 1/1-LCV-112A, VCT LVL CTRL VLV diverts letdown flow to the recycle holdup tank.

OPERATOR ACTIONS:

NOTE: Normal VCT level is maintained between 46% and 56%.

NOTE: Normal position for 1/1-LCV-112A is VCT.

1. Monitor VCT level on 1-LI-112A, VCT LVL and 1-LI-185, VCT LVL.
   A. IF both VCT levels indicate high, THEN stop makeup by placing 1/1-MU, RCS MU MAN ACT in STOP.
   B. Reduce VCT level by positioning 1/1-LCV-112A, VCT LVL CTRL VLV to HUT.
   C. Verify 1-LK-112C, VCT LVL CTRL, potentiometer setpoint is correct per TDM-203A.
   D. If an overdilution is suspected refer to ABN-105.

Comments / Reference: From ALM-0061A, 1-ALB-6A, Window 4.5	Revision # 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 71 OF 79

ANNUNCIATOR NOM./NO.: **VCT LVL LO-LO** **4.5**

PROBABLE CAUSE:

Excessive RCS leakage
1-LK-112C, VCT LVL CTRL malfunction
1/1-LCV-112A, VCT LVL CTRL VLV malfunction
1-LT-0112, CVCS VOLUME CONTROL TANK 1-01 LEVEL TRANSMITTER 0112 malfunction
1-LT-0185, CVCS VOLUME CONTROL TANK 1-01 LEVEL TRANSMITTER 0185 malfunction

AUTOMATIC ACTIONS:

When both VCT level channels are LO-LO (2%), the charging pump suction shifts to the RWST.

- 1/1-LCV-112D and 1/1-112E, RWST TO CHRG PMP SUCT VLV open
- 1/1-LCV-112B and 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV close

When 1/1-LCV-112B and 1/1-LCV-112C receive a close signal, 1-HV-8220 and 1-HV-8221, CHRG PMP SUCT HI POINT VENT VLVs close.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	014 A2.06	
Importance Rating	2.6	

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

Proposed Question: Common 31

Given the following conditions:

- Both Units are at 100% power.
- The following annunciators are in alarm:
  - 1-ALB-6D, Window 3.6 – DRPI URGENT FAIL.
  - 1-ALB-6D, Window 4.6 – DRPI NON-URGENT FAIL.
  - 1-ALB-6D, Window 3.5 – DRPI ROD DEV.
- Control Bank A Group 2 Rods K8 and F8 have flashing GENERAL WARNING lights.
- Unit 1 T_{AVE} is stable.
- Unit 1 indicated power distribution is normal.
- The following lights are illuminated on the DRPI Bezel:
  - URGENT.
  - ROD DEVIATION.
  - DATA A FAILURE.
  - DATA B FAILURE.

Which of the following actions is required per ABN-712, Rod Control System Malfunction?

A. Immediately trip the Reactor.

B. Place Rod Control in MANUAL.

C. Have Core Performance verify Control Bank A Group 1 rod positions.

D. Commence a plant shutdown to 20% then manually trip the Reactor.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this would be correct for two dropped rods, however, T_{AVE} is stable.
B. Correct. This is the appropriate action given the conditions listed.
C. Incorrect. Plausible because this is the correct action for a single Control Rod, however, two Control Rods are affected.
D. Incorrect. Plausible because Technical Specification LCO 3.1.7.B allows 24 hours to repair or place the Unit in MODE 3.

Technical Reference(s) ABN-712, Sections 4.1, 4.2 & 4.3 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Rod Control Indication and Rod Insertion Limit (RIL) Monitor Systems.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 6, 10
55.43

Comments / Reference: From ABN-712, Section 4.1	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 22 OF 52

4.0 DIGITAL ROD POSITION INDICATION MALFUNCTION

4.1 Symptoms

a. Annunciator Alarms

- DRPI URGENT FAIL (6D-3.6)
- DRPI NON-URGENT FAIL (6D-4.6)
- DRPI ROD DEV (6D-3.5)
- ANY ROD AT BOT (6D-3.7)
- ≥2 ROD AT BOT (6D-4.7)

b. Plant Indications

- DRPI disagrees with step counter by greater than 12 steps
- CONTROL ROD POSN bezel - DARK

Comments / Reference: From ABN-712, Section 4.2 & 4.3	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 23 OF 52

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE:
- Half accuracy is indicated by a DRPI NON-URGENT alarm and a flashing general warning light above that indicator. The discrepancy between indicated position and control board step counter for that group should be within the ±12 steps Technical Specification limit, unless rod is actually misaligned. Therefore, either A OR B DRPI operable is sufficient for TS 3.1.7 and TR 13.1.39 position verification.
- An actual misaligned rod could appear to be a DRPI malfunction. DRPI malfunctions or other possible malfunction(s) may be eliminated using appropriate section(s) of this procedure.
- uC14 may be powered from uC1 (Normal) or uC4 (Alternate).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify CONTROL ROD POSN bezel - INDICATING	a. IF ALL of the following exist, THEN immediately OPEN Reactor Trip Breakers AND GO TO EOP-0.0A/B while other operators continue this procedure. • Unit in MODE 3, 4 or 5 AND • Affected rod(s) NOT fully inserted (CBO) AND • Rod drop time measurements NOT being performed b. Check power supply breakers on uC14 (SFGD 832' wall behind the rod control cabinets) - ON • uC14/1/BKR, DATA CAB A • uC14/2/BKR, DATA CAB B • uC14/3/BKR, Control Board Display c. If necessary, shift uC14 power supply per SOP-608A/B.
☐ 2 Check Unit - IN MODE 1 OR 2	GO TO Step 7.

Comments / Reference: From ABN-712, Section 4.2 & 4.3	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-712
ROD CONTROL SYSTEM MALFUNCTION	REVISION NO. 10	PAGE 24 OF 52

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Check Inoperable DRPI(s) - ≤ ONE PER GROUP.	Perform the following: a. Ensure 1/u-RBSS, CONTROL ROD BANK SELECT - MANUAL b. Monitor and record RCS Tavg once per hour. c. Determine position of non-indicating rod(s) by core power distribution measurement information: 1) Once per 8 hours 2) Within 4 hours after any motion of non-indicating rod which exceeds 24 steps in one direction since last determination of rod's position. OR 3) Reduce THERMAL POWER to less than 50% of RTP per IPO-003A/B within 8 hours. d. Within 24 hours restore inoperable DRPI(s) to operable status such that a maximum of one per group is inoperable.

Comments / Reference: From CPNPP Exam Bank	Revision # 03/12/11

Given the following conditions:

- Both Units are at 100% power.
- Annunciator 1-ALB-6D, Window - 3.6, DRPI URGENT FAIL, is in alarm.
- Annunciator 1-ALB-6D, Window - 4.6, DRPI NON-URGENT FAIL, is in alarm.
- Annunciator 1-ALB-6D, Window - 3.5, DRPI ROD DEV, is in alarm.
- Control Bank A Control Rods K8 and F8 have flashing GENERAL WARNING lights.
- Unit 1 T_{AVE} is stable.
- The following lights are illuminated on the Digital Rod Position Indication Bezel:
    - URGENT.
    - ROD DEVIATION.
    - DATA A FAILURE.
    - DATA B FAILURE.

Which of the following actions is required?

A. Manually trip the Reactor and enter EOP-0.0, Reactor Trip or Safety Injection immediately.

B. Verified Control Bank A Rods (K8 and F8) positions using Core Power Distribution Measurement AND reduce RTP to ≤ 50% in 8 hours.

C. Within one hour commence a controlled plant shutdown per IPO-003A, Power Operations to place the Unit in MODE 3 within 6 hours.

D. Place Rod Control in MANUAL and restore at least one Control Bank A Rod (K8 or F8) position indication to OPERABLE status within 24 hours.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	017 K5.02	
Importance Rating	3.7	

In Core Temperature Monitor System: Knowledge of the operational implications of the following concepts as they apply to the ITM system: Saturation and subcooling of water

Proposed Question: Common 32

Given the following condition with Unit 1 operating at 100% power:

- All four (4) Pressurizer pressure channels indicate 2235 psig and stable.
- 1-PI-3616, RCS PRESS (WR) indicates 2285 psig and stable.
- All four (4) T_{AVE} channels indicate 585ºF and stable.
- Core Exit Thermocouple T7401A (TRAIN A) has an open circuit.
- Core Exit Thermocouple T7435A (TRAIN A) indicates 640ºF.
- Core Exit Thermocouple T7444A (TRAIN A) indicates 639ºF.
- Core Exit Thermocouple T7404A (TRAIN A) indicates 628ºF.
- All remaining TRAIN A Core Exit Thermocouples read between 638ºF and 629ºF.

Which of the following describes the Core Cooling Monitoring System indication resulting from the failure of Core Exit Thermocouple T7401A?

Train A...

A. ...CORE EXIT TEMP, 1-TI-3611-2 (on CB05) will indicate 2300ºF until Core Exit Thermocouple T7401A is manually removed from scan.

B. ...RCS SAT MARGIN, 1-TI-3611-1 (on CB05) will indicate 25ºF Subcooled with no manual action required.

C. ...RCS SAT MARGIN, 1-TI-3611-1 (on CB05) will indicate 300ºF Subcooled until Core Exit Thermocouple T7401A is manually removed from scan.

D. ...CORE EXIT TEMP, 1-TI-3611-2 (on CB05) will indicate 640ºF with no manual action required.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that an open circuit caused the CET indication to fail high, however, the indication would fail low and the display would show the next highest Core Exit Thermocouple (T7435A).
B. Incorrect. Plausible if thought that RCS SAT MARGIN is based on highest reading pressure input, however, it is based on the lowest pressure input and highest temperature.
C. Incorrect. Plausible if thought that the open circuit caused the CET indication to fail low, and the lowest CET indication was used in calculating the RCS SAT MARGIN, however, the highest CET and the lowest pressure are used.
D. Correct. The Core Cooling Monitor microprocessor monitors for failed detectors and only allows valid outputs to be used on the CORE EXIT TEMP and RCS SAT MARGIN indications.

Technical Reference(s) LO21.SYS.RC3.LN, Page 14 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** the input signals to the Core Cooling Monitoring System and **DESCRIBE** how these signals are utilized in determining the thermodynamic condition of the RCS/Reactor Vessel fluid.

Question Source: Bank # X
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, 14
55.43

Comments / Reference: From LO21.SYS.RC3.LN, Page 14	Revision # 02/10/04

COMPONENTS

CORE COOLING MONITOR SYSTEM (CCM)

Two qualified, redundant CCM's are used for Inadequate Core Cooling (ICC) monitoring. Each CCM is designed to indicate Core Exit Thermocouple temperatures (CET function) and to monitor the RCS Subcooling Margin Monitor (SMM function).

CORE EXIT THERMOCOUPLES

To provide input temperature data to the CCM microprocessor, the NSSS-supplied array of fifty CET's has been divided into two separate, redundant trains with each set having a distribution representative of all four quadrants of the reactor core exit area. The planar locations of the CET's with respect to the core fuel assembly position are illustrated on Figure 2. All CET's are axially located just above the Upper Core Plate as illustrated in Figure3.

Each CET is a type K (chromel-alumel) thermocouple contained within an aluminum-oxide insulated, stainless steel sheathed cable (1/8" OD). Each cable passes through one of four vessel head penetrations (located 90º apart and near the core periphery) which contain pressure-boundary sealing assemblies. Figure 2 includes indication of head penetration assignments for the various T/C cables, separated into groups of either twelve or thirteen cables per penetration.

Above the vessel head, the CET cables are grouped into two separate trains. Each train is routed into a separate reference junction box which contains three platinum resistance temperature detectors (RTD's): two used for reference temperature measurements plus one installed spare. These reference measurements permit the transition from chromel-alumel leads to copper conductors for signal transmission to the CCM microprocessor (Figure 4).

The CET signals are used in the CCM to monitor coolant temperatures over the entire range including normal operating conditions and extending to beyond accident extremes. Each thermocouple is constantly checked, by the CCM computer, for open or shorted conditions, and the signal is adjusted to account for the inside containment cold reference junction conditions based on the reference RTD measurements. The highest valid CET signal is displayed on the Control Board and is also employed by the microprocessor to determine the RCS saturation margin.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	027 K2.01	
Importance Rating	3.1	

Containment Iodine Removal System: Knowledge of bus power supplies to the following: Fans

Proposed Question: Common 33

Which of the following describes how the Safeguards 480 VAC Motor Control Centers that supply power to the Containment Pre-Access Filtration System Fans are affected by a Blackout and by a Safety Injection?

A. Unaffected by either a Blackout or a Safety Injection.

B. Tripped on undervoltage by a Blackout, but are unaffected by a Safety Injection.

C. Unaffected by a Blackout, but are load shed by a Safety Injection.

D. Tripped on undervoltage by a Blackout and load shed by a Safety Injection.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible since these fans are typically only run prior to Containment entry which normally occurs in MODE 5 and ESF actuations are for the most part blocked in MODE 5, but they are powered by Safeguards MCCs which are tripped on undervoltage on a Blackout Signal and load shed on A Safety Injection Signal.
B. Incorrect. Plausible since these fans are powered by Safeguards MCCs and are tripped on undervoltage on a Blackout, but they are also load shed on a Safety Injection.
C. Incorrect. Plausible since these fans are powered by Safeguards MCCs and are load shed on a Safety Injection, but they are tripped on undervoltage on a Blackout.
D. Correct. Train A fan is powered from MCC 1EB1-2 and Train B from MCC 1EB2-2. Both of these MCCs are tripped on undervoltage on a Blackout Signal and are load shed on a Safety Injection Signal.

Technical Reference(s): EOP-0.0A, Attachment 8; SOP-801A-VAC-C05; ABN-602, Attachment 1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Containment Ventilation System.

Question Source:
- Bank # X
- 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, 9
- 55.43 ____

Comments / Reference: From EOP-0.0A, Attachment 8	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 50 OF 115

ATTACHMENT 8
PAGE 1 OF 10

LOAD SHEDDING
1-MLB-9

> **NOTE:** The load shedding of the following components is initiated by a shunt trip at the power supply breaker or by a control circuit contact. Unless otherwise identified, loads provided with a breaker location are shunt tripped and should be locally positioned. Loads provided with a handswitch location are control circuit tripped and should be positioned at the handswitch.

MLB	LOAD DESCRIPTION	CONTROL LOCATION
1.1	LOAD SHEDDING COMPLETE 1EA1 (SFGDs 810 Train A Swgr)	
☐	• VENTILATION CHILLER X-01 COMPRESSOR MOTOR BREAKER	XCICE1(1EA1/4/BKR)
2.1	LOAD SHEDDING COMPLETE 1EB1-1 (SFGDs 810 North Wall)	
☐	• 480/120 VAC TRANSFORMER (SPACE HEATER) 1EB1-1/8M/TR FEEDER BREAKER	1EB1-1/8M/BKR

Comments / Reference: From EOP-0.0A, Attachment 8	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 55 OF 115

ATTACHMENT 8
PAGE 6 OF 10

LOAD SHEDDING
1-MLB-10

MLB	LOAD DESCRIPTION	CONTROL LOCATION
1.1	LOAD SHEDDING COMPLETE 1EA2 (SFGDs 852 Train B Swgr)	
☐	• VENTILATION CHILLER X-02 COMPRESSOR MOTOR BREAKER	XCICE2(1EA2/7/BKR)
2.1	LOAD SHEDDING COMPLETE 1EB2-1 (SFGDs 852 HP Chem Feed Room)	
☐	• 125 VDC BATTERY CHARGER BC1D2 SUPPLY BREAKER	1EB2-1/1FR/BKR
☐	• 480/120 VAC TRANSFORMER (SPACE HEATER) 1EB2-1/10M/TR FEEDER BREAKER	1EB2-1/10M/BKR

Comments / Reference: From SOP-801A-VAC-C05	Revision # 14

01/07/05 14:10 Page 1 of 3

Revised Procedure: SOP-801A-VAC-C05, CONTAINMENT PRE-ACCESS FILTRATION SYSTEM LINEUP

Type: OPS, Unit: 1, Revision: 0

Step: SOP-801A-VAC-C05 ==> CONTAINMENT PRE-ACCESS FILTRATION SYSTEM LINEUP

Equip Operator Id:	Equip Description:	Equip Location:	Required Config:	Actual Config:	Verif:	Initials
1-HS-5429	PREACC FILT FN 11	UNIT 1 & 2 CONTROL ROOM // 1-CB-03	NEUTRAL (STOP)		IV	
1-HS-5432	PREACC FILT FN 12	UNIT 1 & 2 CONTROL ROOM // 1-CB-03	NEUTRAL (STOP)		IV	
1EB1-2/3M/BKR-1	PREACCESS FILTRATION FAN 1-11 MOTOR BREAKER 1	UNIT 1 TRAIN A SWITCHGEAR ROOM // E. END	ON		IV	
1EB1-2/3M/BKR-2	PREACCESS FILTRATION FAN 1-11 MOTOR BREAKER 2	UNIT 1 TRAIN A SWITCHGEAR ROOM // E. END	ON		IV	
1EB1-2/10B/BKR-1	CONTAINMENT PREACCESS FILTER UNIT 1-17 480 VAC SUPPLY BREAKER 1	UNIT 1 TRAIN A SWITCHGEAR ROOM // E. END (CP1-VAFUPK-17)	ON		IV	

Comments / Reference: From SOP-801A-VAC-C05	Revision # 14

01/07/05 14:10 Page 2 of 3

Revised Procedure: SOP-801A-VAC-C05, CONTAINMENT PRE-ACCESS FILTRATION SYSTEM LINEUP

Type: OPS, Unit: 1, Revision: 0

Step: SOP-801A-VAC-C05 ==> CONTAINMENT PRE-ACCESS FILTRATION SYSTEM LINEUP

Equip Operator Id:	Equip Description:	Equip Location:	Required Config:	Actual Config:	Verif:	Initials
1EB1-2/10B/BKR-2	CONTAINMENT PREACCESS FILTER UNIT 1-17 480 VAC SUPPLY BREAKER 2	UNIT 1 TRAIN A SWITCHGEAR ROOM // E. END (CP1-VAFUPK-17)	ON		IV	
1EB2-2/3M/BKR-1	PREACCESS FILTRATION FAN 1-12 MOTOR BREAKER 1	UNIT 1 TRAIN B SWITCHGEAR ROOM // E. END	ON		IV	
1EB2-2/3M/BKR-2	PREACCESS FILTRATION FAN 1-12 MOTOR BREAKER 2	UNIT 1 TRAIN B SWITCHGEAR ROOM // E. END	ON		IV	
1EB2-2/12H/BKR-1	CONTAINMENT PREACCESS FILTER UNIT 1-18 480 VAC SUPPLY BREAKER 1	UNIT 1 TRAIN B SWITCHGEAR ROOM // E. END (CP1-VAFUPK-18)	ON		IV	
1EB2-2/12H/BKR-2	CONTAINMENT PREACCESS FILTER UNIT 1-18 480 VAC SUPPLY BREAKER 2	UNIT 1 TRAIN B SWITCHGEAR ROOM // E. END (CP1-VAFUPK-18)	ON		IV	

Comments / Reference: From ABN-602, Attachment 1	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 66 OF 107

ATTACHMENT 1
PAGE 2 OF 12

6900/480 V SWITCHGEAR UNDERVOLTAGE LOAD SHEDDING

1. b. Bus 1EB1
   1) PDP
   2) CNTMT FN CLR FN 1
   * 3) MCC XEB1-3
   * 4) MCC 1EB1-2
   * 5) MCC 1EB1-3
   ** 6) PRZR CTRL HTR GROUP C

   * Supply breakers only trip if the following occur: 1) Train A DG is ready to load; 2) Breakers 1EA1-1 AND 1EA1-2 are open; 3) An undervoltage condition exists on 1EB1 OR Train A DG is supplying Train A power.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	029 G 2.1.32	
	Importance Rating	3.8	

Containment Purge System: Conduct of Operations: Ability to explain and apply all system limits and precautions

Proposed Question: Common 34

Given the following conditions:

- Unit 1 is in MODE 6.
- Preparations for core off-load are in progress.
- The Fuel Transfer Tube Gate Valve is open.
- The Containment Equipment Hatch is removed.
- Both doors of the Personnel Airlock are open.
- The Control Room is starting a Containment Purge per SOP-801A, Containment Ventilation System.
- 1-HV-5572, CNTMT AIR PRG SPLY DMPR 1-01, will remain closed for the Containment Purge.

Which of the following describes the reason for leaving 1-HV-5572, CNTMT AIR PRG SPLY DMPR 1-01, closed during the Containment purge per SOP-801A, Containment Ventilation System?

A. Ensures water from the Spent Fuel Pools is NOT transferred to the Refueling Cavity.

B. Ensures air flow into Containment to prevent contaminating the Safeguards Building.

C. Ensures water from the Refueling Cavity is NOT transferred to the Spent Fuel Pools.

D. Ensures air flow into the Safeguards Building so it can be monitored by Vent Stack radiation monitors.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because with the Fuel Transfer Tube Gate Valve open there is a potential for transferring water, however, there is insufficient differential pressure developed because both doors of the Personnel Airlock are open.
B. Correct. As outlined in the NOTE in SOP-801A.
C. Incorrect. Plausible because with the Fuel Transfer Tube Gate Valve open there is a potential for transferring water, however, there is insufficient differential pressure developed because both doors of the Personnel Airlock are open.
D. Incorrect. Plausible because this condition could exist with the Personnel Airlock open, however, only if 1-HS-5572 was also open.

Technical Reference(s) SOP-801A, Step 5.1.6.E NOTE & CAUTION Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Containment Purge 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, 9	
	55.43		

Comments / Reference: From SOP-801A, Step 5.1.6.E NOTE & CAUTION	Revision # 14

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. SOP-801A
CONTAINMENT VENTILATION SYSTEM	REVISION NO. 14 CONTINUOUS USE	PAGE 38 OF 56

5.6.1

[CV] D. REPLACE fuses listed below for the Containment Purge Supply AND Exhaust dampers.

Termination Rack 1-TC-04

Damper 1-HV-5537

- ☐ Fuse FB1-13
- ☐ Fuse FB1-15

Damper 1-HV-5539

- ☐ Fuse FB1-9
- ☐ Fuse FB1-11

Termination Rack 1-TC-05

Damper 1-HV-5536

- ☐ Fuse FB1-17
- ☐ Fuse FB1-19

Damper 1-HV-5538

- ☐ Fuse FB1-13
- ☐ Fuse FB1-15

CAUTION: WHEN the transfer tube is open, THEN differential pressure between the Containment AND Fuel Building may cause level to change in the Refueling Cavity AND Spent Fuel Pools.

NOTE: IF requested by RP OR Decon, THEN the air purge supply damper (1-HS-5572) may be left closed during periods that the equipment hatch is off AND personnel airlock is open to ensure air flow through the personnel airlock is into Containment. This will prevent the spread of contamination into the Safeguards Building.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	035 K4.05	
Importance Rating	2.9	

Steam Generator System: Knowledge of the SGS design feature(s) and/or interlock(s) that provide for the following: Amount of reserve water in SG

Proposed Question: Common 35

Given the following conditions on Unit 2:

- Main Feedwater Regulating Valve to Steam Generator 2-01 closed and the Unit tripped on Low-Low Steam Generator level.

Which of the following describes the setpoint, logic, and the basis for when Auxiliary Feedwater flow was automatically initiated?

A.
- 35.4% Narrow Range Level;
- 2 out of 4 on Steam Generator 2-01;
- Prior to tubes uncovering to ensure a secondary heat sink.

B.
- 35.4% Narrow Range Level
- 2 out of 4 Steam Generators less than setpoint;
- Keeps the tubes covered on a tube rupture.

C.
- 38% Narrow Range Level;
- 2 out of 4 on Steam Generator 2-01;
- Keeps the tubes covered on a tube rupture.

D.
- 38% Narrow Range Level;
- 2 out of 4 Steam Generators less than setpoint;
- Prior to tubes uncovering to ensure a secondary heat sink.

Proposed Answer: A

Explanation:

A. Correct. A single Steam Generator below 35.4% (U-2) initiates Auxiliary Feedwater to keep the tubes covered for secondary heat removal.
B. Incorrect. Plausible because the setpoint is correct and the logic is correct for starting the Turbine Driven Auxiliary Feedwater Pump and it could be thought that maintaining a level above the tubes was for the partitioning affect during a tube rupture, however, the reason for the level is for covering the tubes for heat removal.
C. Incorrect. Plausible because 38% is the Unit 1 setpoint and the logic is correct for starting the Turbine Driven Auxiliary Feedwater Pump and it could be thought that maintaining a level above the tubes was for the partitioning affect during a tube rupture, however, the reason for the level is for covering the tubes for heat removal. The U-2 setpoint is 35.4% and the logic for initiation of the Motor Driven Auxiliary Feedwater Pumps is 2/4 on a single Steam Generator.
D. Incorrect. Plausible because 38% is the Unit 1 setpoint and the logic is correct for starting the Turbine Driven Auxiliary Feedwater Pump and the basis is correct, however, the U-2 setpoint is 35.4%.

Technical Reference(s) LO21.SYS.AF1.LN, Pages 11, 15, & 34 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Auxiliary Feedwater System.

Question Source: Bank # X
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 4, 7
55.43 ______

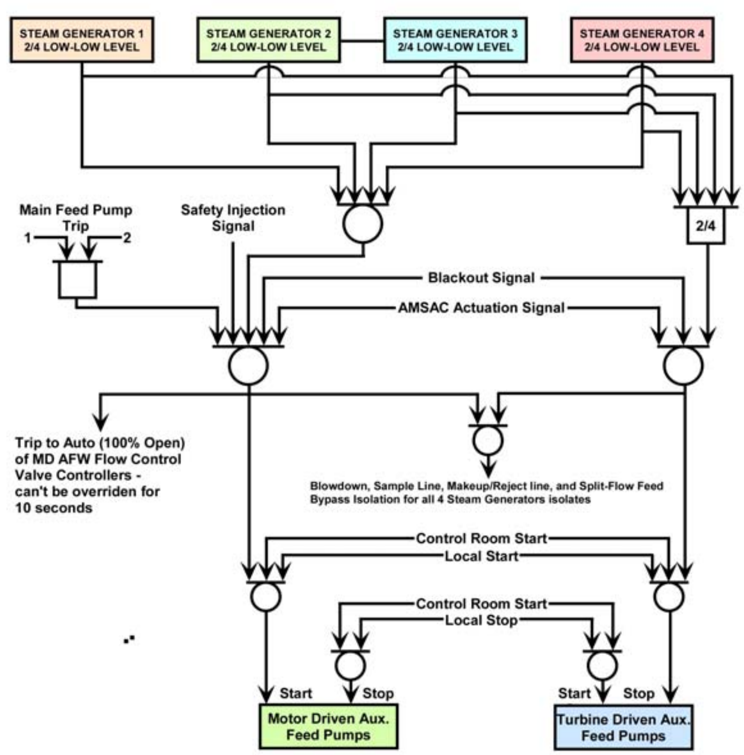
Comments / Reference: From LO21.SYS.AF1.LN, Page 11	Revision # 05/11/11

When Control Room switches are inaccessible, manual operation from the Remote Shutdown Panel (RSP) is provided. Local manual control from the RSP overrides all other signals. Manual control is switched from control board to the RSP with transfer switches located on the Shutdown Transfer Panel (STP) (Train "A") or on the RSP (Train "B"). When control is transferred, an alarm for local override is actuated in the Control Room.

The MDAFWPs will automatically start due to (**Figure 2**):

- Low-low Steam Generator narrow range level at 38% (35.4% for Unit 2) in two out of four detectors on any one Steam Generator,
- Trip of both main feed pumps,
- Safety injection sequence signal (SI),
- Blackout (BO) sequence signal, or
- AMSAC signal

AUXILIARY FEEDWATER PUMP START LOGIC



OP51.SYS.AF1.FG02

03-31-2008

Comments / Reference: From LO21.SYS.AF1.LN, Page 15	Revision # 05/11/11

The TDAFW Pump may be started or stopped from the Control Room by opening or closing the steam supply valves. The TDAFW Pump steam supply valves, HV-2452-1 and HV-2452-2, are operated using three position (OPEN, AUTO, CLOSE) switches on CB-09 which spring return to center position and pull-to-lock in the STOP position.

When Control Room switches are inaccessible, manual operation from the RSP is provided. Local manual control from the RSP overrides all other signals. Manual control is switched from the Main Control Board to the RSP with installed hand switches on the Switch Transfer Panel (STP) for the Train A valve (main steam line 4) or the RSP for the Train B valve (main steam line 1). When control is transferred, an alarm for local override is sounded in the Control Room.

The TDAFWP steam supply valves will automatically open, admitting steam to the TDAFW Pump turbine, due to:

- Low-low SG NR level at 38% (35.4% for Unit 2) on two of four detectors in any two SGs,
- Blackout Sequencer operator lockout signal, or
- AMSAC signal

Comments / Reference: From LO21.SYS.AF1.LN, Page 34	Revision # 05/11/11

The SGs remove decay heat from the reactor core following a reactor trip or shutdown. Proper operation of the AFW System to maintain SG inventory is necessary in order to provide an adequate heat sink. If the AFW pump discharge flowpath is not properly aligned then part of the required AFW flow will be diverted prior to reaching the SGs. Failure to maintain an adequate heat sink by establishing the minimum required AFW feed flow to the SGs will result in core damage due to fuel and clad over heating.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	045 A3.05	
Importance Rating	2.6	

<u>Main Turbine Generator System</u>: Ability to monitor automatic operation of the MTG System, including: Electrohydraulic control

Proposed Question: Common 36

Given the following conditions:

- Unit 1 Main Turbine startup is in progress as follows:
    - Turbine Speed is 1800 rpm.
    - Exhaust Hood temperature is 174°F.
    - Turbine Stress Evaluator (TSE) Margin is GREEN.
    - No operator action has been taken since establishing 1800 rpm.

Given the following Main Turbine auxiliary components:

1. Shaft Lift Oil Pump
2. Turbining Gear Valves
3. Auxiliary Lube Oil Pumps
4. LP Turbine Control Valves
5. HP Turbine Control Valves

Which of the following indicates the status of the components at this point in Main Turbine startup?

A. 1 – OFF
2 – CLOSED
3 – RUNNING
4 – OPEN
5 – CLOSED

B. 1 – ON
2 – OPEN
3 – RUNNING
4 – OPEN
5 – NOT FULLY OPEN

C. 1 – OFF
2 – CLOSED
3 – STOPPED
4 – OPEN
5 – NOT FULLY OPEN

D. 1 – ON
2 – CLOSED
3 – STOPPED
4 – NOT FULLY OPEN
5 – NOT FULLY OPEN

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because all conditions are correct with the exception of the HP Control Valves. These valves would be partially open to maintain speed at 1800 RPM.
B. Incorrect. Plausible because the HP and LP Control Valve positions are correct, however, the Shaft Oil Pump automatically stops at 540 RPM, the Turning Gear Valve closes at 260 RPM, and the Auxiliary Lube Oil Pumps are stopped by the operator at 1765 rpm.
C. Correct. Given the conditions listed and with no operator action once the Main Turbine reaches 1800 RPM, this is the correct condition of the Main Turbine components.
D. Incorrect. Plausible because Turning Gear Valve position and HP Control Valve positions are correct, however, the other components are incorrect for the conditions listed.

Technical Reference(s): IPO-003A, Section 5.1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** and **DESCRIBE** the purpose of in-plant and Control Room System controls, indications, and alarms for the Main Turbine components.

Question Source: Bank # X
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 IPO-003A, Section 5.1	Revision # 28

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-003A
POWER OPERATIONS	REVISION NO. 28 **CONTINUOUS USE**	PAGE 32 OF 195

NOTE: The TSE Margin Display has 2 Bar Graphs. Each of the 2 Bar Graphs has a positive and a negative temperature scale which represent Upper and Lower TSE Margins. At this point the upper bar graphs should be green and above 60°F.

5.1.21 B. Verify upper TSE Margin is above 60°F and Upper Admission Bar is green on the "TSE Margin" Display.

_______/_______
Initials Date

NOTE:
- Hold Setpoint Function on the "TG" Display may be used at anytime during the Turbine Roll-up if problems occur.
- Initiating the Hold Setpoint Function will automatically reduce (ramp down) the turbine speed to 500 rpm. The turbine then remains at warm-up speed until the Operator resumes startup.

CAUTION: If the Upper TSE Margin stops the Main Turbine rollup prior to attaining at least 1765 RPM, Main Turbine speed should immediately be reduced to approximately 500 RPM to allow the Main Turbine to continue soaking.

C. In the "Speed Control" Section, roll the Main Turbine to 1800 RPM by raising the "Speed Target" Controller to 1800 RPM.

_______/_______
Initials Date

D. Verify Lube Oil Temperature is maintained at approximately 113°F as indicated on the TURB BRG TEMP RCDR 1 recorder (1-SB10T010.G01 recorder point 12 on 1-CB-10) while Main Turbine speed is increased.

_______/_______
Initials Date

E. Perform the following:
- Verify 1-HS-6579, TURB SHAFT LIFT OIL PMP automatically stops at a Main Turbine speed of approximately 540 RPM.
- Place 1-HS-6579 in AUTO AFTER STOP

_______/_______
Initials Date

F. WHEN turbine speed is approximately 1400 rpm, THEN ensure the EXCITER AIR DRIER and EXCITER HEATER in OFF at the Unit 1 GENERATOR AUXILIARIES CABINET JC91 (TB 778, U1 GAC).

_______/_______
Initials Date

G. Verify 1-PI-6558, TURB L/O PMP DISCH PRESS is between 155 and 175 psig.

_______/_______
Initials Date

H. WHEN Main Turbine speed increases above 1765 RPM, THEN stop ALL running Auxiliary Oil Pumps and place in AUTO.

_______/_______
Initials Date

Comments / Reference: From IPO-003A, Section 5.1	Revision # 28

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-003A
POWER OPERATIONS	REVISION NO. 28 **CONTINUOUS USE**	PAGE 30 OF 195

5.1.19 E. On the “Lube Oil” Display, verify “Turning Gear Valve # 1” closes at a Main Turbine speed of approximately 260 RPM.

_______ / _______
Initials Date

F. Verify no unexpected or sudden increase in vibration is indicated:

- ☐ Turbine Display or Turbine Vibration Display
- ☐ Generator Display or Generator Vibration Display
- ☐ Alarm Summary Display (Asd)

_______ / _______
Initials Date

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	055 K1.06	
Importance Rating	2.6	

Condenser Air Removal System: Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: PRM system

Proposed Question: Common 37

Given the following conditions on Unit 1:

- A small Steam Generator tube leak is in progress.
- 1-RE-2959 (COG-182), CONDENSER OFF GAS Radiation Monitor is rising.
- One Condenser Exhausting Vacuum Pump is in service.

Which of the following describes the effect on Condenser vacuum?

Condenser vacuum should [1] due to an increase in [2].

	[1]	[2]
A.	worsen;	non-condensable gases
B.	improve;	condensable gases
C.	worsen;	condensable gases
D.	improve;	non-condensable gases

Proposed Answer: A

Explanation:

A. Correct. Condenser vacuum should rise due to an increase in non-condensable gases brought on by the Steam Generator tube leak.

B. Incorrect. Plausible because absolute pressure will rise but vacuum will lower because dissolved gases leaking from the Reactor Coolant System are non-condensable gases.

C. Incorrect. Plausible because vacuum will lower, however, it is the presence of non-condensable gases that causes vacuum to lower. Condensable gases, such as steam, act to improve vacuum.

D. Incorrect. Plausible because it is the presence of non-condensable gases that causes Condenser pressure to change, however, absolute pressure will rise but vacuum will lower.

Technical Reference(s) LO21.SYS.CV1.LN, Pages 4 & 8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the physical connections and **EVALUATE** the cause-effect relationships between the Condenser Vacuum and Water Box Priming System and the following systems, components or events:

- Main Turbine

Question Source: Bank # X
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, 14
55.43 ______

Comments / Reference: From LO21.SYS.CV1.LN, Page 4	Revision # 05/25/11

CONDENSER VACUUM SYSTEM (CV)

Provide initial evacuation of main condenser shells and auxiliary condenser shells (steamside) during startup by removing air and non-condensable gases. (Hogging Mode)

Provide for removal of non-condensable gases from steam side of main and auxiliary condensers during operation. (Holding Mode)

Provide a vacuum breaker arrangement for the Main and Auxiliary Condenser shells.

Prevent an unmonitored release of radioactive material to the environment through the use of the Radiation Monitoring System.

Comments / Reference: From LO21.SYS.CV1.LN, Page 8	Revision # 05/25/11

Normal Operation (holding):

At approximately 20 inches of vacuum, the second stage can handle the entire first stage discharge and the internal bypass arrangement closes. The two-stage combination continues to pull the system down to operating pressure. The second stage exhaust is then separated and discharged. If, for some reason, the condenser shell pressures increase during the holding operation beyond this setting, the CEV pumps will automatically revert to the hogging operation. Details of pump operation are covered in the COMPONENTS section.

During the "holding" sequence the number of CEV pumps in operation can be reduced. Since all three CEV pumps are initially used, if available, the operator must manually stop unneeded pumps by switching them to OFF. One of those pumps is then placed in STANDBY. This standby pump is brought automatically into service on the loss of the operating pump due to an increase in suction pressure. If the standby pump starts, it must be manually removed from service to avoid hunting, the rapid on-off-on operation of pumps operating near their automatic setpoints.

Condenser Vacuum System Flow Path (Fig. 2)

Air and non-condensable gases are drawn from the main condenser shell thru the 8-inch piping and individual isolation valves to common isolation valve u-CV-0020. Air and non-condensable gases are drawn from the auxiliary condenser shells thru the 8-inch piping and individual isolation valves to common isolation valve u-CV-0022. These lines join to form the suction of the CEV pumps. Each pump discharges through its own seal water tank (Separator) and silencer to a common, 10" discharge header .Air and non-condensable gases in the discharge header are monitored for radiation by the condenser off-gas radiation monitor (u-RE-2959), located in a bypass line, and then discharged (in the Aux Building) to the Primary Plant Ventilation System.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	2	
K/A #	068 A4.04	
Importance Rating	3.8	

Liquid Radwaste System: Ability to manually operate and/or monitor in the control room: Automatic isolation

Proposed Question: Common 38

Which of the following will cause an AUTO closure of X-RV-5253, Liquid Waste Processing System Discharge Isolation Valve while a release is in progress?

A. PC-11 channel not responding to POLL (MAGENTA).

B. Only 2 of 4 Circulating Water Pumps running on associated Unit.

C. PC-11 channel in ALERT alarm (YELLOW).

D. Loss of counts on X-RE-5253, Liquid Effluent Radiation Monitor.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that a monitor not responding to POLL would be INOPERABLE.
B. Incorrect. Plausible because Circulating Water Pumps must be running for the valve to remain open, however, a 2 of 4 coincidence allows release to Unit aligned for discharge.
C. Incorrect. Plausible because radiation level has increased, however, it requires a high radiation level alarm to close X-RV-5253.
D. Correct. A loss of counts on the Liquid Effluent Radiation Monitor will trip X-RV-5253 (OPERATE FAILURE).

Technical Reference(s): ALM-3200, Page 38; ALM-3200, Attachment 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the functions, operation and interlocks of the Liquid Waste Processing System components.

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

Question History: Last NRC Exam CPNPP 2011

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

10 CFR Part 55 Content: 55.41 11
55.43

Comments / Reference: From ALM-3200, Page 38	Revision # 4

CPSES CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 4	PAGE 38 OF 117

ALARM: **OPERATE FAILURE-CHANNEL NO PULSES RECEIVED** COLOR: BLUE

AFFECTED MONITORS:

All monitors may display this alarm.

PROBABLE CAUSES:

Loss of high voltage to the detector
Damaged signal cable
Failed detector
Failed detector pre-amplifier
Channel Item 020 LOSS OF COUNTS TIME value is too short

MONITOR RESPONSE:

Automatic actions for monitors which actuate due an OPERATE FAILURE will be initiated

OPERATOR ACTION:

1. Determine the affected monitor.

   A. IF any of the following monitors are affected, THEN notify Radwaste personnel of the alarm condition.

      - LWE076 (X-RE-5253)
      - TBDu72 (u-RE-5100)

Comments / Reference: From ALM-3200, Attachment 3	Revision # 4

CPSES CPSES ALARM PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ALM-3200
ALARM PROCEDURE DRMS	REVISION NO. 4	PAGE 102 OF 117

ATTACHMENT 3
Page 1 of 1

AUTOMATIC ACTIONS

> NOTE: A loss of power to the RM-80 will result in the Automatic Actions for the associated monitor.

TITLE	CHANNEL	FUNCTION	PRINT
Plant Vent Stack Wide Range Gas Monitor	X-RE-5570A S. X-RE-5570B N.	Closes HCV-014 on High Radiation or any OPERATE FAILURE	E1-0046 Sh 62/63
Auxiliary Building Exhaust	X-RE-5701	Closes HCV-014 on High Exhaust Radiation or any OPERATE FAILURE	E1-0065 Sh 22
Liquid Waste to Circulating Water	X-RE-5253	Closes discharge to Circulating Water Circulating Water (X-RV-5253) on High Radiation or any OPERATE FAILURE	E1-0065 Sh 29

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:

- Unit 1 is at 75% power.
- Solid State Protection System (SSPS) Train B Actuation Logic testing is being performed.
- Train B SSPS Mode Selector switch is in the TEST position.
- Train B SSPS Input Error Inhibit switch is in the INHIBIT position.

Which of the following describes the status of the Reactor if a loss of one of the two 48 VDC instrument power supply were to occur on Train A SSPS?

A. Reactor at 75% power with a General Warning for Train A SSPS ONLY.

B. Reactor at 75% power with a General Warning for Train B SSPS ONLY.

C. Reactor Trip with a General Warning for BOTH Train A and Train B SSPS and a First Out Alarm illuminated.

D. Reactor Trip with a General Warning for BOTH Train A and Train B SSPS and NO First Out Alarm illuminated.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because a General Warning is generated for a loss of either 48 VDC power supply. If this were the only General Warning the Unit would remain at power, but performing testing on the other train generates a General Warning for both trains and the Unit trips.
B. Incorrect. Plausible because a General Warning is generated while performing testing on SSPS. If this were the only General Warning the Unit would remain at power, but a loss of either 48 VDC power supply on the other train generates a General Warning for both trains and the Unit trips.
C. Correct. Testing on one train of SSPS generates a General Warning. A loss of any of the four DC power supplies in the other train of SSPS also generates a General Warning. General Warnings in both trains of SSPS causes the Reactor Trip Breakers to open, which then causes the Turbine to trip. Since the power level is above 50%, the Turbine trip then causes a Reactor trip signal to be generated which causes the First Out annunciator. The First Out annunciator would NOT alarm if power were below 50%.
D. Incorrect. Plausible because a Reactor Trip is generated, but a First Out annunciator occurs due to the Unit being above P-9 (50%) power for RX > 50% PWR TURB TRIP.

Technical Reference(s) ALM-0064A, 1-ALB-6D, Window 1.5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Solid State Protection System including interrelations with other systems to include interlocks and control loops.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content: 55.41 6, 7
55.43

Comments / Reference: From ALM-0064A, 1-ALB-6D, Window 1.5	Revision # 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 19 OF 147

ANNUNCIATOR NOM./NO.: **SSPS TRN A GEN WARNING** 1.5

PROBABLE CAUSE:

Surveillance testing
Loss of power
Internal power supply failure

NOTE: Controlled evolutions for authorized testing should not require an alarm response.

AUTOMATIC ACTIONS: None

NOTE:
- The SSPS trouble alarm generates a GENERAL WARNING condition on the associated train. If a GENERAL WARNING condition exists on both trains, a Reactor trip is actuated.
- If a GENERAL WARNING condition exists on both trains and power $<$P-9, no first out annunciator will be in alarm.
- If a GENERAL WARNING condition exists on both trains and power $\geq$P-9, a RX $>$50% PWR TURB TRIP first out alarm will be illuminated.

Comments / Reference: From CPNPP Exam Bank	Revision # 03/11/09

Given the following conditions:

- Unit 1 is at 40% power.
- Solid State Protection System (SSPS) Train B Actuation Logic testing is being performed.
- Train B SSPS Mode Selector Switch is in the TEST position.
- Train B SSPS Input Error Inhibit Switch is in the INHIBIT position.

Which of the following describes the status of the Reactor if a loss of Distribution Panel 1PC1 were to occur on Train A SSPS?

A. Reactor at 40% power with a GENERAL WARNING for Train A SSPS only.

B. **Reactor Trip with a GENERAL WARNING for both Train A and Train B SSPS with the First Out annunciator NOT illuminated**.

C. Reactor at 40% power with a GENERAL WARNING for Train B SSPS only.

D. Reactor Trip with a GENERAL WARNING for both Train A and Train B SSPS and the First Out annunciator illuminated.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008 AK2.03	
Importance Rating	2.5	

Pressurizer Vapor Space Accident: Knowledge of the interrelations between Pressurizer Vapor Space Accident and the following: Controllers and positioners

Proposed Question: Common 40

Given the following conditions:

- A Loss of Coolant Accident (LOCA) is in progress on Unit 1.
- 1-PI-455A, PRZR PRES CHAN I, is indicating 2500 psig.
- The other three Pressurizer Pressure Channels are indicating 2100 psig and lowering.
- 1EA1, Train A 6.9KV Safeguards Bus is de-energized.
- Containment pressure is 4 psig and rising.
- All Reactor Coolant Pumps are secured.
- Pressurizer valve status is as follows;
    - Pressurizer Safety Valves are CLOSED.
    - 1-PCV-455A, PRZR PORV, is OPEN.
    - 1-PCV-456, PRZR PORV, is CLOSED.
- Pressurizer level is 100%.
- Reactor Vessel Level Indicating System (RVLIS) lights at 33", 21" and 11" are LIT and the remaining RVLIS lights are DARK.

Which of the following control channels should have initiated an automatic action to terminate the LOCA?

A. 1-PT-0455, PRZR PRESS XMTR 0455 PROT CHAN I

B. 1-PT-0456, PRZR PRESS XMTR 0456 PROT CHAN II

C. 1-PT-0457, PRZR PRESS XMTR 0457 PROT CHAN III

D. 1-PT-0458, PRZR PRESS XMTR 0458 PROT CHAN IV

Proposed Answer: D

Explanation:

A. Incorrect. Plausible as the indications are consistent with those which would exists if a high failure of PT-455 were to occur with no operator action per ABN-705.
B. Incorrect. Plausible if thought that the indications are consistent with those which would exist if a failure of PCV-456 were to occur with no operator action per ABN-705.
C. Incorrect. Plausible because this answer would be correct if PORV-456 were failed in the mid position.
D. Correct. When pressure drops below 2185 psig the interlock associated with PT-0458 should have closed PORV-0455A.

Technical Reference(s): ALM-0052A, 1-ALB-5B, Windows 1.6 & 2.6; ABN-705, Section 2.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: During abnormal or emergency events, **ANALYZE** indications to determine the cause of the abnormal or emergency 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

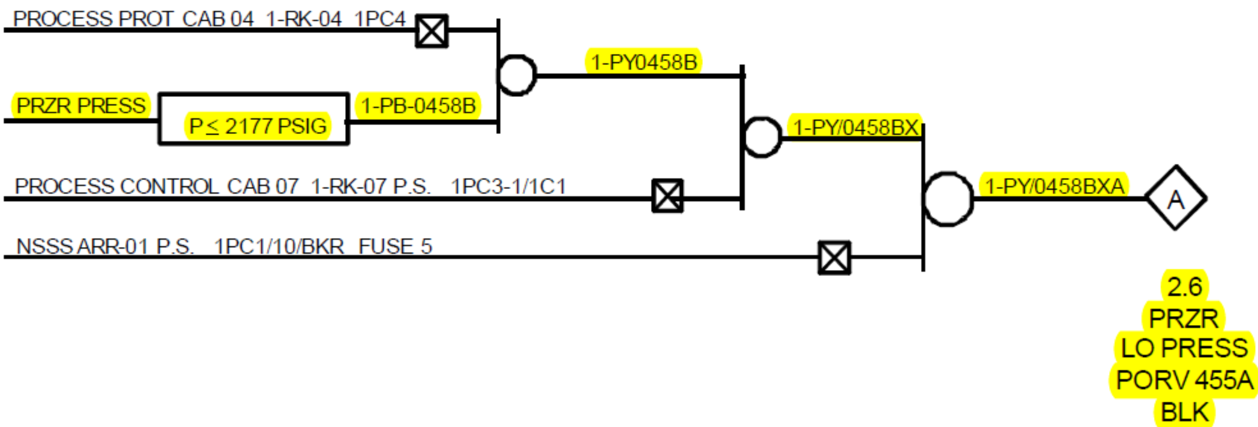
10 CFR Part 55 Content:
55.41 7
55.43 ______

Comments / Reference: From ALM-0052A, 1-ALB-5B, Window 2.6	Revision # 5

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0052A
ALARM PROCEDURE 1-ALB-5B	REVISION NO. 5	PAGE 33 OF 72

ANNUNCIATOR NO.: 2.6

LOGIC:



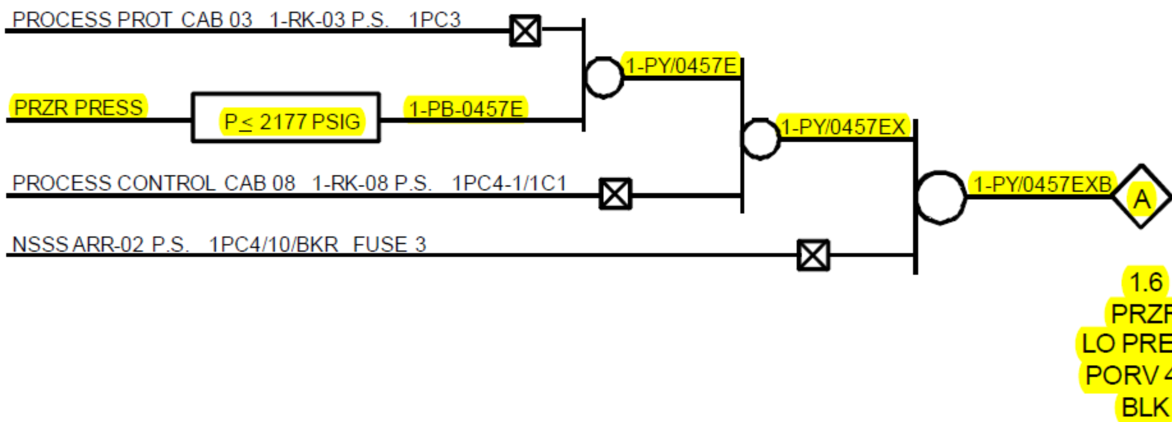
NOTE: The PORV permissive is enabled >2185 psig (increasing), but the alarm comes on <2177 psig (decreasing), this allows for a 1% deadband for the bistable.

Comments / Reference: From ALM-0052A, 1-ALB-5B, Window 1.6	Revision # 5

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0052A
ALARM PROCEDURE 1-ALB-5B	REVISION NO. 5	PAGE 17 OF 72

ANNUNCIATOR NO.: 1.6

LOGIC:



NOTE: The PORV permissive is enabled >2185 psig (increasing), but the alarm comes on <2177 psig (decreasing), this allows for a 1% deadband for the bistable.

Comments / Reference: From ABN-705, Section 2.2	Revision # 12

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-705
PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 12	PAGE 4 OF 26

2.2 Automatic Actions

NOTE: Control responses will only occur if failure occurs in a channel selected for control.

a. Control response for a pressurizer pressure channel failure HIGH.

1) PORV will open until pressure is reduced to 2185 psig, then the other channel will close the PORV.

- 1/u-PCV-455A, PRZR PORV
- 1/u-PCV-456, PRZR PORV

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	009 EA2.37	
	Importance Rating	4.2	

Small Break LOCA: Ability to determine and interpret the following as they apply to the Small Break LOCA: Existence of adequate natural circulation

Proposed Question: Common 41

Given the following conditions:

- A Small Break Loss of Coolant Accident has occurred on Unit 1.
- Conditions to start a Reactor Coolant Pump cannot be established.
- Containment pressure is 8 psig and slowly rising.
- Reactor Coolant System (RCS) pressure is 1085 psig and stable.
- All Steam Generator narrow range levels are approximately 55% and stable.
- Steam Generator pressures are as follows:
    - 1-01 is 785 psig and stable.
    - 1-02 is 790 psig and stable.
    - 1-03 is 850 psig and stable.
    - 1-04 is 780 psig and stable.
- RCS Cold Leg temperatures are as follows:
    - Loop 1 is 518°F and stable.
    - Loop 2 is 519°F and stable.
    - Loop 3 is 360°F and lowering.
    - Loop 4 is 517°F and stable.
- All RCS Hot Leg temperatures are approximately 540°F and stable.
- Core Exit Thermocouples are reading approximately 550°F and slowly rising.

What is the status of natural circulation and the expected operator action?

A. Adequate natural circulation does NOT exist and the Steam Dump Valves should be opened farther.

B. Adequate natural circulation does exist and the Atmospheric Relief Valve 1-03 should be closed farther.

C. Adequate natural circulation does NOT exist and the Atmospheric Relief Valves should be opened farther.

D. Adequate natural circulation does exist and Auxiliary Feedwater flow to Steam Generator 1-03 should be reduced.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible as Natural Circulation does NOT exist, however, the MSIVs would have closed at 6.2 psig Containment pressure and thus the Steam Dump Valves are isolated.
B. Incorrect. Plausible as three of the four Steam Generators appear to be coupled. However, adequate subcooling does not exist and one Steam Generator is uncoupled. Additionally, the CETs are increasing which indicates that adequate Natural Circulation does NOT exist. Steam Generator 3 appears to be overcooling, thus the concept to decrease steam dumping from this generator is plausible, however, the low temperature indicates lack of circulation and Cold Leg temperature lowering as a result of ECCS flow.
C. Correct. Since adequate subcooling does NOT exist, CETs are increasing and SG 1-03 is uncoupled, which indicate that Natural Circulation does NOT exist. EOS-1.2A, states to increase dumping steam to promote Natural Circulation. The Steam Dump Valves are isolated by the MSIVs which only leave the Atmospheric Relief Valves for dumping steam.
D. Incorrect. Plausible as three of the four Steam Generators appear to be coupled. However, adequate subcooling does not exist and one Steam Generator is uncoupled. Additionally, the CETs are increasing which indicates that adequate Natural Circulation does NOT exist. Steam Generator 3 appears to be overcooling, thus the concept to decrease steam dumping from this generator is plausible, however, the low temperature indicates lack of circulation and Cold Leg temperature lowering as a results of ECCS flow.

Technical Reference(s) EOS-1.2A, Attachment 3 & Step 22.a RNO Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the stagnant Reactor Coolant System Loops generic issue in the Emergency Response Guideline network and proper operator response.

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, 14
55.43 ___

Comments / Reference: From EOS-1.2A, Attachment 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.2A
POST LOCA COOLDOWN AND DEPRESSURIZATION	REVISION NO. 8	PAGE 36 OF 68

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- ☐ RCS subcooling - GREATER THAN 25°F (55°F FOR ADVERSE CONTAINMENT).
- ☐ SG pressures - STABLE OR DECREASING.
- ☐ RCS hot leg temperatures - STABLE OR DECREASING.
- ☐ Core exit TCs - STABLE OR DECREASING.
- ☐ RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE.

Comments / Reference: From EOS-1.2A, Step 22.a RNO	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.2A
POST LOCA COOLDOWN AND DEPRESSURIZATION	REVISION NO. 8	PAGE 18 OF 68

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		3) Start RCP 4 per Attachment 2. IF RCP 4 can NOT be started, THEN start other RCP(s) per Attachment 2 as necessary to provide normal spray. IF RCP(s) can NOT be started, THEN refer to Attachment 3 to verify natural circulation. IF natural circulation NOT verified, THEN increase dumping steam.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	015/17 AK1.04	
Importance Rating	2.9	

RCP Malfunctions: Knowledge of the operational implications of the following concepts as they apply to the RCP Malfunctions: Basic steady-state thermodynamic relationship between RCS loops and SGs resulting from unbalanced loop flows

Proposed Question: Common 42

Given the following conditions:

- Unit 2 is at 35% power.
- Reactor Coolant Pump (RCP) 2-02 trips.

In the 30 seconds following the trip of RCP 2-02, and assuming NO operator action, an automatic Reactor Trip will ...

A. ...occur, and the affected SG water level will shrink.

B. ...NOT occur, but the affected SG water level will shrink.

C. ...occur, and the affected SG water level will swell.

D. ...NOT occur, but the affected SG water level will swell.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because SG level will shrink, however, the Reactor will not automatically trip unless power level is greater than 48%.
B. Correct. A Reactor Trip will occur when one Reactor Coolant Pump trips with Reactor power greater than 48%. When the RCP trips, Steam Generator pressure will rise and result in SG shrink.
C. Incorrect. Plausible if thought that the P-8 permissive had been met, however, the Reactor does not trip and Steam Generator water level will shrink.
D. Incorrect. Plausible because the Reactor will not trip, however, steam pressure in Loop #2 will rise and result in shrink of SG water level.

Technical Reference(s): ABN-101, Section 2.2; ABN-101, Step 2.3.1 NOTE — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Reactor Coolant System.
ANALYZE the response to an RCP Trip per ABN-101, Reactor Coolant Pump Trip/Malfunction.

Question Source: Bank # X
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, 14
55.43 ______

Comments / Reference: From ABN-101, Section 2.2	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 3 OF 48

2.0 REACTOR COOLANT PUMP TRIP

2.1 Symptoms

a. Annunciators Alarm

- ANY RCP TRIP (5B-1.1)
- 1 OF 4 RCP UNDRVOLT (5B-1.2)
- RC LOOP 1 1 OF 3 FLO LO (5A-1.3)
- 1 OF 4 RCP UNDRFREQ (5B-2.2)
- RC LOOP 2 1 OF 3 FLO LO (5A-2.3)
- RC LOOP 3 1 OF 3 FLO LO (5A-3.3)
- RC LOOP 4 1 OF 3 FLO LO (5A-4.3)

b. Plant Indications

- Low flow indication on any reactor coolant loop.
- Breaker TRIP or MISMATCH light illuminated on any RCP handswitch.
- Motor amps on any RCP motor reading zero.

2.2 Automatic Actions

- Reactor trip occurs in the event of one reactor coolant pump trip with reactor power greater than 48% (P-8 permissive annunciator NOT LIT).
- Reactor trip occurs in the event of two reactor coolant pumps trip with reactor power or turbine power greater than 10% (P-7 permissive annunciator NOT LIT).

Comments / Reference: From ABN-101, Step 2.3.1 NOTE	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 4 OF 48

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

[C]
CAUTION: A Reactor Coolant Pump shall NOT be started with the reactor in MODE 1 or 2.

NOTE:
- Diamond step 1 denotes Initial Operator Actions.
- With a Reactor Coolant Pump stopped, the affected loop will stop steaming.

1 Check Plant status

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	025 AK3.03	
Importance Rating	3.9	

<u>Loss of RHR System</u>: Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Immediate actions contained in EOP for Loss of RHRS

Proposed Question: Common 43

Given the following conditions:

- Unit 1 has experienced a Large Break Loss of Coolant Accident.
- Safeguards 6.9KV Bus 1EA2 is de-energized.
- A transition has been made to EOS-1.3A, Transfer to Cold Leg Recirculation.
- ECCS has been transferred to Cold Leg Recirculation and the RWST level is 8% and lowering.
- Residual Heat Removal Pump (RHRP) 1-01 begins to cavitate.

Which of the following sequence of actions is required per EOS-1.3A, Transfer to Cold Leg Recirculation?

A. Stop RHRP 1-01, Close 1-8811A CNTMT SMP TO RHRP 1 SUCT ISOL VLV, Open 1-8812A RWST TO RHRP 1 SUCT ISOL VLV, then Restart RHRP 1-01.

B. Open 1-8812A RWST TO RHRP 1 SUCT ISOL VLV, if cavitation exists then stop RHRP 1-01.

C. Stop Centrifugal Charging Pump 1-01, Stop Safety Injection Pump 1-01, then Stop RHRP 1-01.

D. Stop RHRP 1-01, Stop Centrifugal Charging Pump 1-01, then Stop Safety Injection Pump 1-01.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible as these actions are similar to the actions necessary if the Containment Sump to RHR Pump Suction Valve will not open in EOS-1.3A.
B. Incorrect. Plausible as this action should provide increased NPSH to the RHR Pump, however, this is not the procedural guidance.
C. Correct. Because the CAUTION in EOS-1.3A states to stop any ECCS Pump which loses suction or shows indication of cavitation should be stopped. Additionally, the CAUTION states that the CCP and SIP should be stopped before stopping the RHRP.
D. Incorrect. Plausible as these pumps should be stopped per EOS-1.3A, however, the CCP and SIP should be stopped prior to the RHRP.

Technical Reference(s) EOS-1.3A, Step 3 CAUTION Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from EOS-1.3, Transfer to Cold Leg Recirculation, **STATE** the purpose/basis for the step(s).
Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOS-1.3, Transfer to Cold Leg Recirculation.

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 EOS-1.3A, Step 3 CAUTION	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 4 OF 53

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: Any ECCS pump taking suction from RWST should be stopped at RWST EMPTY. Any Containment Spray pump taking suction from RWST should be stopped when RWST level reaches 0%.

CAUTION: Any ECCS or Containment Spray pump that loses suction or shows indication of cavitation should be stopped. The CCP and SI pump should be stopped before stopping the RHR pump.

CAUTION: SI pumps should be stopped if RCS pressure is greater than their shutoff head pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 3	Align ECCS For Cold Leg Recirculation:	IF at least one flow path from the sump to the RCS can NOT be established or maintained, THEN go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1. Implement FRGs if required.
	a. Check open CNTMT SMP TO RHRP 1 AND RHRP 2 SUCT ISOL VLVS: • 1/1-8811A • 1/1-8811B	a. IF ONE RHR sump suction valve failed to open, THEN stop RHR pump with valve closed AND go to Step 3b to align operating RHR pump. IF BOTH RHR sump suction valves failed to open AND ECCS running for injection, THEN realign RHR pump suction one pump at a time: 1) Stop RHR pump 1(2). 2) Close 1/1-8812A(B).

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	027 AK1.01	
Importance Rating	3.1	

Pressurizer Pressure Control Malfunction: Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Definition of saturation temperature

Proposed Question: Common 44

Given the following conditions:

- Unit 2 Pressurizer, Power Operated Relief Valve (PORV) 2-PCV-455A, PRZR PORV 455A, is partially stuck open discharging into the Pressurizer Relief Tank.
- Pressurizer pressure is 1400 psig.
- Pressurizer Relief Tank pressure is 5 psig.

Which of the following is the condition of the fluid downstream of 2-PCV-455A, PRZR PORV 455A?

A. Superheated steam

B. Dry saturated steam

C. Wet vapor

D. Subcooled liquid

Proposed Answer: A

Explanation:

A. Correct. With a pressure of 1400 psig, the isenthalpic expansion occurs at approximately ~1174 BTUs/lbm. Following this enthalpy to that point on the Mollier Diagram where 5 psig (20 psia) intersects is located in the superheated steam region.

B. Incorrect. Plausible because errors using the Steam Tables or Mollier Diagram could result in determining that the fluid is dry saturated steam.

C. Incorrect. Plausible because errors using the Steam Tables or Mollier Diagram could result in determining that the fluid is wet vapor.

D. Incorrect. Plausible because errors using the Steam Tables or Mollier Diagram could result in determining that the fluid is subcooled liquid.

Technical Reference(s) LO21.GFE.PRO.LN, Pages 22 & 23; Mollier Diagram — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Mollier Diagram

Learning Objective: **SOLVE** throttling process problems, applying the General Energy Equation.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 14
55.43 ______

Comments / Reference: From LO21.GFE.PRO.LN, Pages 22 & 23	Revision # 09/24/07

Figure 4-18 Property Diagrams of a Steam Throttling Process

A steam safety relief valve at 2250 psia is leaking a small amount. The relief tank is at 20 psia. Essentially the entire pressure drop occurs across the valve. Using a Mollier diagram, the quality of the steam exiting the valve is 66.5%.

When a throttling process is analyzed, a small opening of the valve is assumed. When the valve is nearly shut, the entire pressure drop is across the valve, and the process is considered isenthalpic. The more open the valve, the smaller the pressure drop across the valve. A greater percentage of the pressure drop becomes a function of the head loss in the piping. Pressure reduction due to frictional head loss is not isenthalpic.

The temperature-enthalpy diagram indicates an interesting behavior of water with respect to enthalpy. The specific enthalpy of saturated water vapor increases with temperature until approximately 450°F. Above 450°F, the specific enthalpy decreases with increasing temperature up to the critical point.

This phenomenon is particularly important during rapid pressure drops occurring during plant operations. At initially high temperatures (i.e., 650°F), an isenthalpic drop to atmospheric pressure results in high quality steam at 212°F. At initially lower temperatures (i.e., 400°F), an isenthalpic drop to atmospheric pressure results in superheated steam at approximately 320°F. For example, if a relief valve in the main steam line fails partially open, steam is throttled through the valve. The steam exiting is initially at a temperature of approximately 545°F. The steam experiencing an isenthalpic pressure drop to atmospheric pressure exits the throttled valve as superheated steam at a temperature of 290°F. On the other hand, steam is throttled through a valve at a temperature of 653°F and enters the piping to the relief tank (20psia) as a saturated mixture at 230°F. Because the processes are isenthalpic, an originally higher steam temperature exits the throttle valve at a lower temperature.

Comments / Reference: From CPNPP Exam Bank	Revision # 08/14/07

Given the following conditions:

- Unit A pressurizer power-operated relief valve is stuck partially open with the fluid being discharged into a pressurizer relief tank. The pressurizer pressure is 2200 psia and the relief tank pressure is 5 psig.

Which of the following is the condition of the fluid downstream of the relief valve?

A. Superheated steam

B. Dry saturated steam

C. <u>Wet vapor</u>

D. Subcooled liquid

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	038 G 2.4.8	
Importance Rating	3.8	

Steam Generator Tube Rupture: Emergency Procedures/Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOPs

Proposed Question: Common 45

Given the following conditions:

- Unit 1 Main Steam Line 1-04 Radiation Monitor (MSL 181) alarms 10 seconds prior to a Reactor trip and Safety Injection.
- Auxiliary Feedwater flow has been secured to Steam Generator 1-04.
- Narrow range level in Steam Generator 1-04 continues to increase.
- During performance of EOP-0.0A, Reactor Trip or Safety Injection it is observed that all Radiation Monitors on PC-11, Digital Radiation Monitoring System are GREEN.

Which of the following would be appropriate when Step 13, Check if SG Tubes are Not Ruptured, is reached in EOP-0.0A, Reactor Trip or Safety Injection?

A. Recognize Steam Generator 1-04 level is increasing in an uncontrolled manner and transition to EOP-3.0A, Steam Generator Tube Rupture.

B. Since all PC-11, Digital Radiation Monitoring System Radiation Monitors are GREEN; continue in EOP-0.0A, Reactor Trip or Safety Injection.

C. Wait until a Steam Generator sample from Chemistry confirms a tube leak. Until then, continue in EOP-0.0A, Reactor Trip or Safety Injection.

D. Monitor Main Steam Line N16 Radiation Monitors and if an increase is seen, transition to EOP-3.0A, Steam Generator Tube Rupture.

Proposed Answer: A

Explanation:

A. Correct. Even with no confirming Radiation Monitor alarms EOP-0.0A requires that if any Steam Generator level is increasing in an uncontrolled manner then EOP-3.0A, SGTR entry is required.

B. Incorrect. Plausible because PC-11 indications are GREEN, however, with Steam Generator level rising a transition to EOP-3.0A is required.

C. Incorrect. Plausible because this action would be performed during a Steam Generator tube leak, however, other indications are used to verify a tube rupture.

D. Incorrect. Plausible because the N16 Radiation Monitors are designed to detect Steam Generator tube leaks during operation, however, in a post-trip condition these instruments are no longer useful due to loss of N16 production. See EOP-0.0A, Step 13, Basis reference.

Technical Reference(s) EOP-0.0A, Step 13
EOP-0.0A, Step 13 Bases

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the diagnostic steps of EOP-0.0, Reactor Trip or Safety Injection.

Question Source: Bank # X
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 EOP-0.0A, Step 13	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 11 OF 114

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13	Check If SG Tubes Are Not Ruptured: • Condenser off gas radiation - NORMAL (COG-182, 1RE-2959) • Main steamline radiation - NORMAL (MSL-178 through 181, 1RE-2325 through 2328) • SG blowdown sample radiation monitor - NORMAL (SGS-164, 1RE-4200) • No Steam Generator level increasing in an uncontrolled manner	Go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.

Comments / Reference: From EOP-0.0A, Step 13, Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 91 OF 114

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BASES

STEP 13: Abnormal condenser off gas, main steamline,or SG blowdown sample radiation indicates primary to secondary leakage. "Normal" means the value of a process parameter experienced during routine plant operations. Trending of secondary radiation monitors ensures that any changes in secondary radiation levels can be compared to previous plant conditions. This will aid in primary to secondary leak determination.

In addition, an uncontrolled steam generator level increase is indicative of secondary leakage. "Uncontrolled" means not under the control of the operator and incapable of being controlled by the operator using available equipment.

Steam Generator Leak Rate Monitors are installed upstream of the MSIVs and are provided to detect a slow-propagating SG tube leak during unit operation. The SG Leak Rate Monitors detect N-16 gammas to provide a correlation of primary to secondary leakage. The N-16 gamma monitored by the SG Leak Rate Monitors will no longer exist following a reactor trip even though primary to secondary leakage will continue. The Leak Rate Monitor trends may be used to confirm a steam generator tube rupture, but the parameter is not listed as a main indication since the reading will cease following a reactor trip.

Optimal recovery in dealing with a steam generator tube rupture is provided in EOP-3.0A, STEAM GENERATOR TUBE RUPTURE.

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	054 AK1.01	
Importance Rating	4.1	

Loss of Main Feedwater: Knowledge of the operational implications of the following concepts as they apply to the Loss of Main Feedwater: MFW line break depressurizes the S/G (similar to a steam line break)

Proposed Question: Common 46

Given the following conditions:

- Unit 1 has received an automatic Reactor Trip and Safety Injection.
- Containment pressure is 4 psig and rising.
- Containment Sump level and pump run alarms are locked in.
- Steam Generator 1-01 level is lowering.
- Steam Generators 1-02, 1-03 and 1-04 levels are rising.
- All Steam Generator pressures are 1080 psig and stable.

Which of the following events initiated the Reactor Trip and Safety Injection?

A. Main Steam Line break on Steam Generator 1-01

B. Main Feed Line break on Steam Generator 1-01.

C. Small Break Loss of Coolant Accident on Reactor Coolant System Loop 1 Cold Leg.

D. Small Break Loss of Coolant Accident on Reactor Coolant System Loop 1 Hot Leg.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Containment pressure and dew point are both rising, however, Steam Generator pressures are stable which would differentiate a steam line from a feed line break.
B. Correct. Given the conditions listed, this is the correct diagnosis.
C. Incorrect. Plausible because Containment Sump level and pressure are both rising and Steam Generator pressures are stable, however, lowering level in Steam Generator 1-01 with the other conditions listed is indicative of a feed line break.
D. Incorrect. Plausible because Containment Sump level and pressure are both rising and Steam Generator pressures are stable, however, lowering level in Steam Generator 1-01 with the other conditions listed is indicative of a feed line break.

Technical Reference(s): EOP-2.0A, Step 3; EOP-2.0A, Attachment 3, Step 3 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the symptoms for the entry conditions of EOP-2.0, Faulted Steam Generator Isolation.

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 3, 5
55.43 ____

6

Comments / Reference: From EOP-2.0A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 3 OF 14

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: At least one SG must be maintained available for RCS cooldown.

CAUTION: Any faulted SG or secondary break should remain isolated during subsequent recovery actions unless needed for RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check Main Steamline Isolation Valves - CLOSED	Manually close MSIVs.
2	Check At Least One SG Pressure - STABLE OR INCREASING	IF all SG pressures decreasing in an uncontrolled manner or completely depressurized, THEN go to ECA-2.1A, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS, Step 1.
3	Identify Faulted SG(s):	
	a. Check pressures in all SGs: • ANY SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER -OR- • ANY SG COMPLETELY DEPRESSURIZED	a. Search for initiating break: • Main steamlines • Main feedlines • Other secondary piping Go to Step 5.

Comments / Reference: From EOP-2.0A, Attachment 3, Step 3 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 10 OF 14

ATTACHMENT 3
PAGE 2 OF 6

BASES

decreasing is small or the cause is known, it should not be considered DECREASING IN AN UNCONTROLLED MANNER. Non-faulted SG pressures may decrease slightly due to RCS cooldown from a faulted SG. This non-faulted SG pressure response can be controlled by the operator when the faulted SG depressurizes.

STEP 3: An uncontrolled SG pressure decrease (following MSIC closure and FW isolation) or a completely depressurized SG indicates an unisolated failure of the secondary pressure boundary. If the faulted SG is not isolated following MSIV closure, the operator is directed to search for the initiating break in main steamlines, feedlines, or other secondary piping such as blowdown lines, sample lines, etc. The operator should also check for stuck open atmospheric relief valves or safety valves. If the faulted SG is isolated following MSIV closure, the operator is directed to identify where the break was (may require local checks since break has been isolated) while continuing with this procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055 EA2.03	
	Importance Rating	3.9	

Station Blackout: Ability to determine and interpret the following as they apply to the Station Blackout: Actions necessary to restore power

Proposed Question: Common 47

A system wide grid blackout has occurred. Which of the following is the preferred Black Start corridor for Comanche Peak?

Align the...

A. ...138 KV Switchyard to be powered from Stephenville.

B. ...345 KV Switchyard east bus to be powered from DeCordova.

C. ...345 KV Switchyard west bus to be powered from Wolf Hollow.

D. ...138 KV Switchyard to be powered from DeCordova.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the second option for black start is the 138 KV Stephenville corridor.
B. Incorrect. Plausible because if the 138 KV switchyard cannot be aligned, then the DeCordova 345 KV line is the next available option.
C. Incorrect. Plausible because the Wolf Hollow plant is the nearest in proximity to CPNPP and has been the black start unit in the past.
D. Correct. The 138 KV DeCordova line is the preferred Black Start corridor.

Technical Reference(s) ABN-601, Attachment 20 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Switchyard 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 __10__
55.43 ______

Comments / Reference: From ABN-601, Attachment 20	Revision # 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 11 CONTINUOUS USE	PAGE 191 OF 203

ATTACHMENT 20
PAGE 1 OF 3

Alignment of the Black Start Corridor

A. 138 KV SWITCHYARD BLACK START CORRIDOR

NOTE:
- The primary source of offsite power during a system blackout will be 138 KV line from Decordova. The TGM Transmission Grid Controller will also energize the 345 KV transmission line as soon as practical.
- Electrical power may be available from black start combustion turbines as early as 90 minutes following a system blackout, therefore the 138 KV switchyard corridor should be setup in the first 90 minutes, if possible.
- Breaker manipulations may be performed by Transmission personnel or by NEOs if Transmission is unavailable.
- Stephenville is an alternate 138KV blackout source if the primary source is not available (maintenance, etc). Transmission personnel perform breaker switching operations whether the primary or alternate source is used for black start. If an alternate source is NOT one of the 3 listed below, then alternate source actions will be as directed / performed by Transmission personnel.

1. DO NOT perform this attachment until communication is established with Transmission. Check the Black Start Corridor alignment desired:

☐ • 138KV DeCordova (N/A steps A.4b and A.6b)

☐ • 138KV Stephenville (N/A steps A.4a and A.6a)

☐ • 345 DeCordova (N/A remaining steps in Section A).

Comments / Reference: From ABN-601, Attachment 20	Revision # 11

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-601
RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION	REVISION NO. 11 CONTINUOUS USE	PAGE 193 OF 203

ATTACHMENT 20
PAGE 3 OF 3

Alignment of the Black Start Corridor

B. 345 KV SWITCHYARD BLACK START CORRIDOR

NOTE: The 138KV corridor is the preferred Black Start Corridor. Glen Rose Transmission will normally align the corridor either remotely or locally. This section of the attachment is to be used as a backup if Glen Rose Transmission is not available AND when the 345KV DeCordova line is being utilized as a back up black start power source.

☐ 1. Establish communications with the Transmission Grid Controller to confirm desired corridor alignment.

☐ 2. Align the 345 KV Switchyard as directed by the Transmission Grid Controller; the following steps provide guidance.

3. OPEN the following breakers from the Control Room:

☐ • CS-E3, GEN BKR 8000 AND	☐ • CS-W3, GEN BKR 8010
☐ • CS-E10, GEN BKR 8020 AND	☐ • CS-W10, GEN BKR 8030

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	056 AA1.05	
Importance Rating	3.8	

Loss of Offsite Power: Ability to operate and/or monitor the following as they apply to a Loss of Offsite Power: Initiation (manual) of safety injection process

Proposed Question: Common 48

Given the following conditions:

- A Loss of Offsite Power has occurred.
- Bus 1EA1 is de-energized.
- EOS-0.1A, Reactor Trip Response, is in progress.
- Reactor Coolant System (RCS) temperature is 561ºF and stable.
- RCS pressure is 1920 psig and trending down slowly.
- Pressurizer level cannot be maintained greater than 6%.

Which of the following actions is required per EOS-0.1A, Reactor Trip Response?

A. Increase Condenser Steam Dump to maintain T_{AVE} at 557ºF.

B. Attempt to restore Bus 1EA1 per ABN-602, Response to a 6900V/480V System Malfunction.

C. Manually actuate Safety Injection and return to EOP-0.0A, Reactor Trip or Safety Injection.

D. Isolate Letdown and verify Natural Circulation per EOS-0.1A, Reactor Trip Response, Attachment 3.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this is a Step 1 RNO action of EOS-0.1A, however, Steam Dump will not be available without Circulating Water Pumps. T_{AVE} is where it should be for the conditions.
B. Incorrect. Plausible because it would have been performed in EOP-0.0A, however, priority is Safety Injection (SI).
C. Correct. Foldout Page requires initiation of SI when PRZR level cannot be maintained greater than 6%.
D. Incorrect. Plausible because Natural Circulation would be verified if SI was not required per the Foldout Page.

Technical Reference(s) EOS-0.1A, Attachments 1.A & 3
EOS-0.1A, Steps 1 RNO & 6 RNO

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the recovery technique used and the procedure steps of EOS-0.1, Reactor Trip Response.

Question Source: Bank # X
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 EOS-0.1A, Attachment 1.A	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 8	PAGE 18 OF 40

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-0.1A, REACTOR TRIP RESPONSE

1. SI ACTUATION CRITERIA

Actuate SI and go to EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, Step 1, if EITHER condition listed below occurs:

- RCS subcooling - LESS THAN 25°F
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 6%

Comments / Reference: From EOS-0.1A, Attachment 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 8	PAGE 25 OF 40

ATTACHMENT 3
PAGE 1 OF 1

NATURAL CIRCULATION VERIFICATION

The following conditions support or indicate natural circulation flow:

- ☐ RCS subcooling - GREATER THAN 25°F (55°F FOR ADVERSE CONTAINMENT).
- ☐ SG pressures - STABLE OR DECREASING.
- ☐ RCS hot leg temperatures - STABLE OR DECREASING.
- ☐ Core exit TCs - STABLE OR DECREASING.
- ☐ RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE.

Comments / Reference: From EOS-0.1A, Step 1 RNO	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 8	PAGE 3 OF 40

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: If SI actuation occurs during this procedure, EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, shall be performed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1	Check RCS Temperature - • RCS AVERAGE TEMPERATURE STABLE AT OR TRENDING TO 557°F	IF temperature less than 557°F and decreasing, THEN perform the following: a. Stop dumping steam. b. IF cooldown continues, THEN reduce total AFW flow as necessary to minimize the cooldown: • Maintain a minimum of 460 gpm UNTIL narrow range level greater than 43% in at least one SG. • As necessary to maintain SG levels WHEN narrow range level greater than 43% in at least one SG. • IF Turbine Driven AFW pump is not required to maintain greater than 460 gpm flow, THEN stop Turbine Driven AFW pump. c. IF cooldown continues, THEN perform the following as necessary to maintain RCS temperature: • Ensure SG blowdown isolated. • Trip both MFW pumps. • Close main steamline isolation valves. IF temperature greater than 557°F and increasing, THEN dump steam: • To condenser using steam dumps. -OR- • To atmosphere using SG atmospherics.

Comments / Reference: From EOS-0.1A, Step 1 RNO	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.1A
REACTOR TRIP RESPONSE	REVISION NO. 8	PAGE 14 OF 40

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		b. Restore offsite power per ABN-601, RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION or ABN-602, RESPONSE TO A 6900/480 VOLT SYSTEM MALFUNCTION while continuing with this procedure.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	057 AA1.05	
Importance Rating	3.2	

Loss of Vital AC Instrument Bus: Ability to operate and/or monitor the following as they apply to a Loss of Vital AC Instrument Bus: Backup instrument indications

Proposed Question: Common 49

Given the following conditions:

- Unit 1 is at 100% power.
- 1-ALB-10B, Window 4.16 – 118V CHAN IV INV TRBL is lit.
- Multiple indications on the Main Control Board are not indicating actual parameter values.
- Multiple Main Control Board annunciators are LIT.

Which of the following is an alternate method for determining which 118V Protection Bus is de-energized?

A. Row four lights on the Trip Status Light Boxes are LIT.

B. Row four lights on the PCIP are LIT.

C. Row four lights on the Trip Status Light Boxes are DARK.

D. Row four lights on the PCIP are DARK.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because a loss of Channel 1, 2 or 3 results in the associated Trip Status Light Boxes lighting for all trips that are de-energize to actuate.
B. Incorrect. Plausible because the PCIP provides important information on Permissives and Control logics. It is plausible that an operator could attempt to assess the plant status from the PCIP Windows as a change of status does occur on the panel, however, the operator cannot determine that a loss of Channel 4 has occurred from this panel alone.
C. Correct. As identified in the NOTE box of ABN-603, Section 2.0.
D. Incorrect. Plausible because the PCIP provides important information on Permissives and Control logics. It is plausible that an operator could attempt to assess the plant status from the PCIP Windows as a change of status does occur on the panel, however, the operator cannot determine that a loss of Channel 4 has occurred from this panel alone.

Technical Reference(s) ABN-603, Section 2.0 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to Loss of a Protection Bus per ABN-603, Loss of Protection or Instrument Bus.

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 ABN-603, Section 2.0	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 3 OF 34

2.0 LOSS OF PROTECTION BUS

2.1 Symptoms

a. The affected inverter trouble alarm

- 118V CHAN 1 INV TRBL (10B-1.16)
- 118V CHAN 2 INV TRBL (10B-2.16)
- 118V CHAN 3 INV TRBL (10B-3.16)
- 118V CHAN 4 INV TRBL (10B-4.16)
- 118V INV IVuEC1/3 TRBL (10B-1.18)
- 118V INV IVuEC2/4 TRBL (10B-4.18)

b. The affected channel instruments failing or alarming (refer to Attachments 1 and 2).

c. Possible reduced charging flow due to Ave Tave.

2.2 Automatic Actions

a. Possible reactor trip

NOTE:
- The Process Control Group Cabinets have a backup power supply from Panel uC1 Assumption: Only Card Frame 9 solid state relay cards (NASs) will deenergize on the loss of protection bus.
Assumption: Preferred feedwater control channels are selected.
- On a uPC4 failure all Channel IV bistables will NOT be lit. Steam Generator alarms associated with Channel IV instruments will annunciate and immediately clear. These indications are due to a loss of a control board multiplexer.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058 AK3.02	
	Importance Rating	4.0	

Loss of DC Power: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of DC power

Proposed Question: Common 50

Given the following conditions on Unit 1:

- Actions of ECA-0.0A, Loss of All AC Power, are being performed.
- Safeguards DC Bus voltage is 118 VDC and slowly lowering.

Which of the following actions should be taken?

A. Monitor DC bus voltage and begin load shedding operation of small essential DC loads to permit monitoring and control of the plant until AC power can be restored.

B. Begin load shedding of large non-essential loads as soon as practical, consistent with preventing damage to plant equipment.

C. Monitor DC bus voltage and begin load shedding of small non-essential DC loads to permit monitoring and control of the plant until AC power can be restored.

D. Begin load shedding of large essential loads as soon as practical, consistent with preventing damage to plant equipment.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because loads are shed, however, they are large non-essential loads.
B. Correct. Shedding of non-essential loads is started at Step 16 per ECA-0.0.
C. Incorrect. Plausible because DC bus voltage should be monitored, however, the maximum battery voltage must be maintained as it is the only power source available.
D. Incorrect. Plausible because large loads must be shed, however, they are non-essential loads. The shedding of additional loads would occur if DC voltage was 110 VDC or less.

Technical Reference(s): ECA-0.0A, Attachment 7, Step 16 Bases; ECA-0.0A, Step 16 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-0.0, Loss of All AC Power.

Question Source: Bank # X
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 ECA-0.0A, Attachment 7, Step 16 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 71 OF 88

ATTACHMENT 7
PAGE 13 OF 30

BASES

"Level increase in an uncontrolled manner" means that the operator cannot control level using available equipment, i.e., level continues to rise even when all feed flow valves to that SG are fully closed.

This is a Continuous Action Step.

Step 16: Following loss of all AC power, the station batteries are the only source of electrical power. The station batteries supply the DC busses and the AC vital instrument busses. Since AC emergency power is not available to charge the station batteries, battery power supply must be conserved to permit monitoring and control of the plant until AC power can be restored.

The intent of load shedding is to remove all large non-essential loads as soon as practical, consistent with preventing damage to plant equipment. Prioritized shedding of additional loads is performed in case AC power cannot be restored within the projected life of the station batteries. CPSES analysis for Station Blackout has identified that even without load shedding, the heaviest loaded battery has sufficient capacity to not only carry its loads for a four (4) hour period, but also provide sufficient DC power for AC power restoration. DC voltage may be required to flash the diesel generator field or close safeguards bus supply breakers during the power restoration evolution.

Since the remaining battery life cannot be monitored from the control room, Step requires personnel to be dispatched to locally monitor the DC power supply. This is intended to provide the operator information on remaining battery life and the need to shed additional DC loads.

Comments / Reference: From ECA-0.0A, Step 16	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 15 OF 88

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

	b. Control AFW flow to maintain narrow range level between 43% (50% FOR ADVERSE CONTAINMENT) and 60%.	b. <u>IF</u> narrow range level in any SG continues to increase in an uncontrolled manner, <u>THEN</u> isolate ruptured SG:
		• Manually isolate AFW flow <u>AND</u> dispatch operator to locally isolate AFW flow.
		• <u>IF</u> SG 1 or 4 ruptured, <u>THEN</u> locally close steam supply isolation valve to TDAFW pump:
		  • SG 1 - 1MS-0101 (SG 881' Rm 1-109A, Main Steam Penetration Area)
		  -OR-
		  • SG 4 - 1MS-0128 (SG 881' Rm 1-108B, Main Steam Penetration Area)
		• Adjust ruptured SG(s) atmospheric controller setpoint to 1160 psig.
		  <u>WHEN</u> SG pressure less than 1160 psig, <u>THEN</u> verify SG atmospheric closed. <u>IF</u> <u>NOT</u>, <u>THEN</u> manually close. <u>IF</u> SG atmospheric can <u>NOT</u> be closed, <u>THEN</u> locally isolate atmospheric.

16	Check DC Bus Loads:	
	a. Initiate shedding of DC loads per Attachment 2.	
	b. Voltage - GREATER THAN 110 VOLTS	b. Determine necessity of shedding additional DC loads per Attachment 2.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	062 AA2.03	
Importance Rating	2.6	

Loss of Nuclear Service Water: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition

Proposed Question: Common 51

Given the following conditions:

- Unit 1 is at 100% power.
- A leak has developed on Station Service Water (SSW) 10 inch header upstream of 1SW-0336, DG-1-02 JKT WTR CLR SSW IN ISOL VLV.
- 1SW-0021, U1 SSW TRN B SPLY HDR IN ISOL VLV and 1SW-0007, U1 SSW TRN B RET HDR ISOL VLV have been closed to isolate the leak.

Which of the following components or systems, in addition to the Diesel Generator, have been affected by isolating the SSW leak?

1. 1-02 Centrifugal Charging Pump lube oil cooler
2. 1-02 Safety Injection Pump lube oil cooler
3. 1-02 Component Cooling Water Heat Exchanger
4. X-02 Station Service Water Screenwash Pump suction
5. Auxiliary Feedwater System alternate suction

A. 1, 2, 3

B. 2, 3, 4

C. 2, 4, 5

D. 1, 2, 5

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because 1 & 2 are supplied from Station Service Water 10 inch header while 3 is not.
B. Incorrect. Plausible because 2 is supplied from Station Service Water 10 inch header while 3 & 4 are not.
C. Incorrect. Plausible because 2 & 5 are supplied from Station Service Water 10 inch header while 4 is not.
D. Correct. The CCP, SIP and the AFW alternate suction are all loads on the 10 inch SSW header.

Technical Reference(s): SOP-501A, Step 5.5.2 CAUTION; ABN-501, Attachment 3; LO21.SYS.SW1.LN, Pages 14 & 15 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Loss Station Service Water per ABN-501, Station Service Water System Malfunction.

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

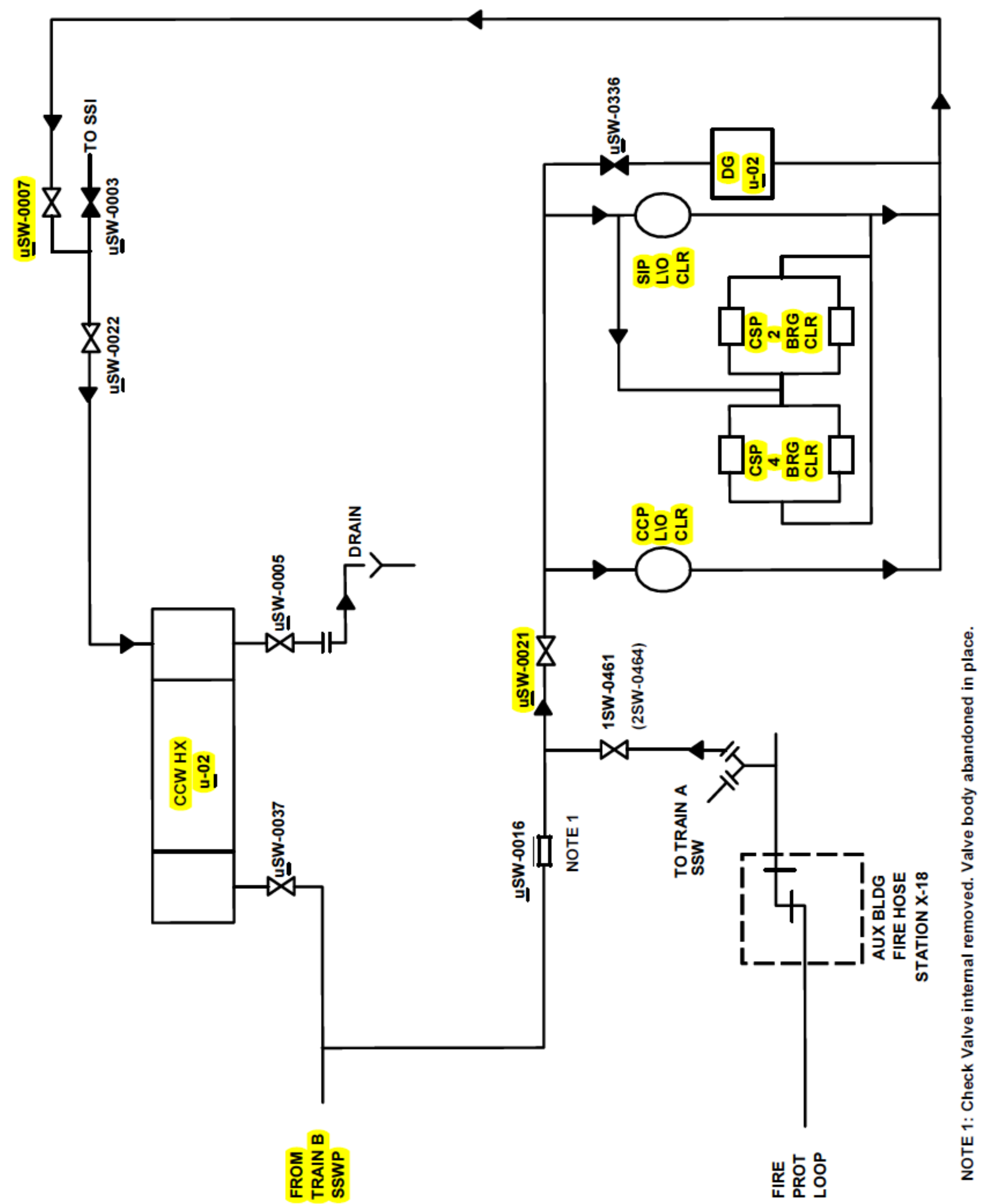
10 CFR Part 55 Content:
55.41 4, 10
55.43 ____

Comments / Reference: From ABN-501, Attachment 3	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-501
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 50 OF 50

ATTACHMENT 3
PAGE 9 OF 9

UNIT 1 AND UNIT 2 TRAIN B ECCS PUMP EMERGENCY COOLING LINEUP



Comments / Reference: From SOP-501A, Step 5.5.2 CAUTION	Revision # 19

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 & COMMON	PROCEDURE NO. SOP-501A
STATION SERVICE WATER SYSTEM	REVISION NO. 19 CONTINUOUS USE	PAGE 35 OF 104

5.5.2 D. Service Water Pump 1-02 (Train B)

- ☐ 1/1-APSI2, SIP 2
- ☐ 1-HS-4766, CSP 2
- ☐ 1-HS-4767, CSP 4
- ☐ 1/1-APCH2, CCP 2
- ☐ CS-1DG2E, DG 2 EMER STOP/START

CAUTION:
- The Service Water Pump is placed in PULL-OUT to prevent any auto start. WHEN 1SW-0001 OR 1SW-0007 is CLOSED in the following steps, THEN the associated SSW Pump will have no flowpath.
- PERFORMANCE of the following step renders the selected Service Water Pump inoperable. (TS 3.7.8)

E. ENSURE the selected Service Water Pump handswitch is in PULL-OUT:

- ☐ 1-HS-4250A, SSWP 1
- ☐ 1-HS-4251A, SSWP 2

Comments / Reference: From LO21.SYS.SW1.LN, Page 15	Revision # 04/05/05

1.1 A separate 10-inch line (**Figure 2**) branches from each of the main 30-inch SSW supply headers and runs through the Auxiliary and Safeguards Building. In the Auxiliary Building, the 10-inch branch supplies cooling water to the centrifugal charging pumps lube oil coolers and in the Safeguards Building it supplies cooling water to the safety injection pumps lube oil coolers, the containment spray pump bearing oil coolers and the diesel generator system. A backup water supply to the auxiliary feedwater pumps is provided from the SSW discharge piping downstream of the diesel generator cooler. The return flow joins the main SSW discharge line on the downstream side of the CCW heat exchanger. Heated service water is piped underground to the service water discharge canal which carries it back to the SSI.

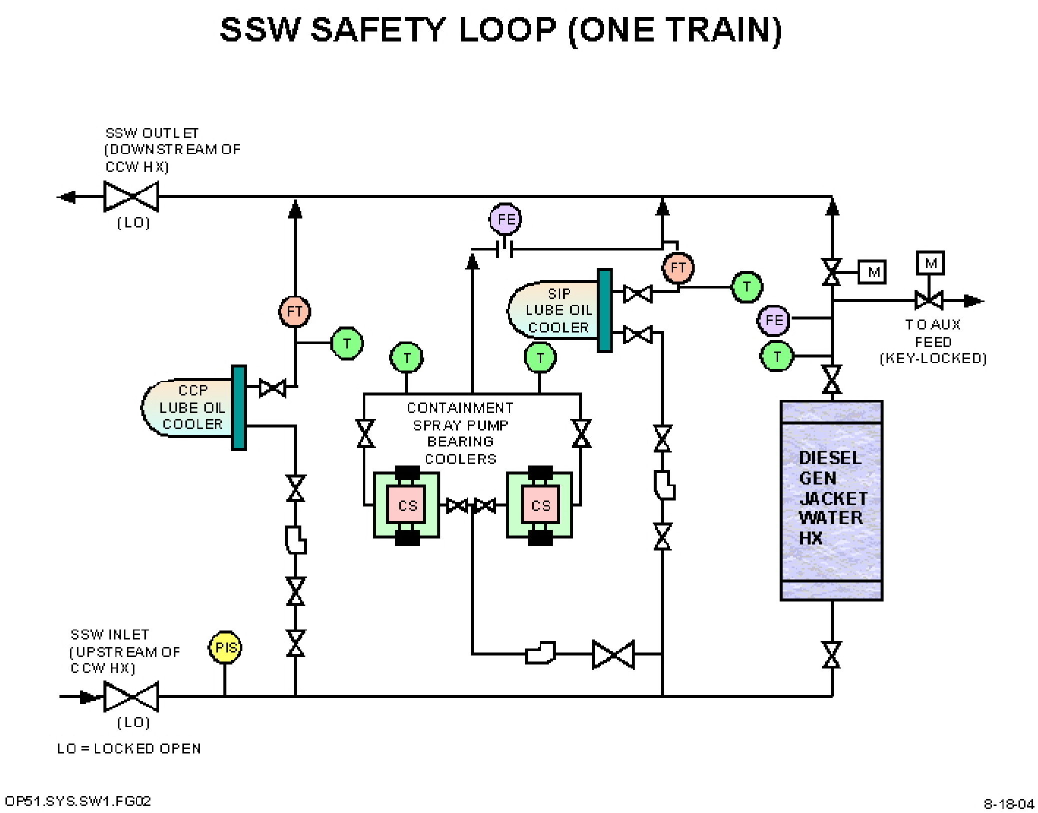


Figure 2 - SSW Safety Loop (One Train)

1.2 The SSW discharge canal is common to both units. If the canal is blocked, the water will spill into the yard and drain to the SSI without impairing operation of the SSW system.

Comments / Reference: From LO21.SYS.SW1.LN, Page 14	Revision # 04/05/05

FLOWPATHS

1.3 The SSW pumps take suction from the Safe Shutdown Impoundment (SSI). SSI water enters the SSW system through trash racks then traveling screens to the intake bay of the Service Water Intake Structure (SWIS) (**Figure 1**).

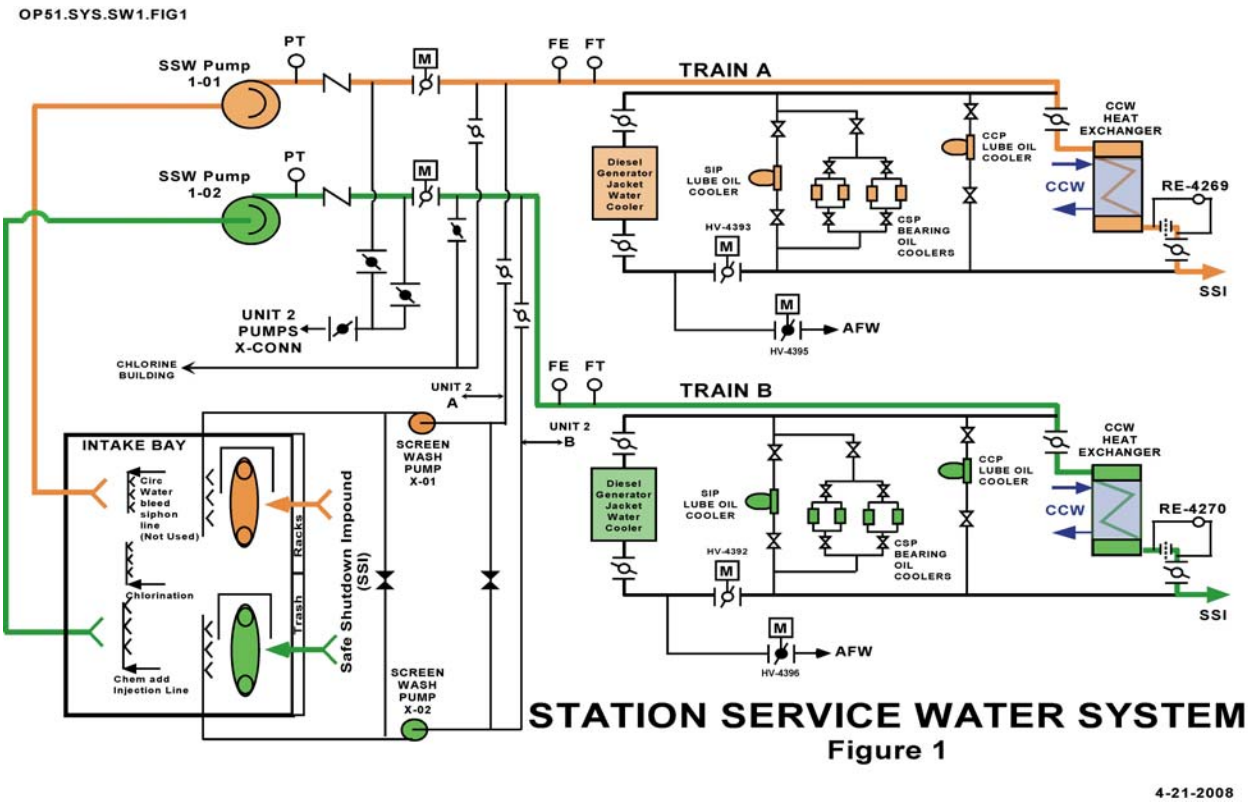


Figure 1 - Station Service Water System

1.4 The SSW pumps discharge to the CCW HX, Diesel Generator (DG) Jacket Water Coolers, Safety Injection Pump (SIP) oil cooler, Centrifugal Charging Pump (CCP) oil cooler and the Containment Spray Pump (CSP) bearing oil coolers, and then return the SSW returns water back to the SSI.

1.5 The alternate or emergency source of water to Auxiliary Feedwater (AFW) is tapped off the outlet side of the Diesel Generator Jacket Water Coolers. This is not only a convenient location but also performs some preheating should SSW be required to supply the AFW system.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	065 AK3.04	
Importance Rating	3.0	

Loss of Instrument Air: Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Crossover to backup air supplies

Proposed Question: Common 52

Which of the following provides the means for shutting the Auxiliary Feedwater Pump Flow Control Valves should the normal supply to the valve operator be lost?

A. Hydraulic oil supply.

B. Electric motor operator.

C. Nitrogen pressure backup.

D. Instrument Air Accumulators.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Turbine Driven Auxiliary Feedwater Pump uses a hydraulically operated speed changer mechanism.
B. Incorrect. Plausible because the Motor Driven Auxiliary Feedwater Pump Isolation Valves are motor operated, however, the Flow Control Valves are air operated.
C. Incorrect. Plausible because nitrogen is associated with the Auxiliary Feedwater System, however, its use is in the Condensate Storage Tank.
D. Correct. The Instrument Air Accumulators allow for 5 full cycles of the valves plus leakage and steady-state consumption for 30 minutes.

Technical Reference(s) LO21.SYS.AF1.LN, Pages 9, 13, & 17 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the basic design and flow path of the Auxiliary Feedwater System.

Question Source: Bank # X
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 4, 7
55.43 ______

Comments / Reference: From LO21.SYS.AF1.LN, Page 13	Revision # 05/11/11

MDAFWP FLOW CONTROL VALVES

Each MDAFW pump discharge line branches into individual lines feeding its two associated SGs. The individual AFW line to each SG is provided with a normally open, pneumatically operated flow control valve. Manual isolation valves are provided for maintenance and local flow control.

MDAFW pump flow to each SG is controlled by flow control valves, PV-2453A and B for the Train A pump, PV-2454A and B for the Train B pump. The flow control valves fail open on loss of air or electrical power.

Each flow control valve is provided with a safety class air accumulator sized for five full cycles, plus leakage and steady state consumption for 30 minutes. This allows the valve to control AFW flow following a loss of Instrument Air coincident with a plant condition which requires AFW operation, or to isolate a faulted SG when the normal motor operated isolation valves are not available. The manual isolation valves are then used to control the flow in the event of loss of air to the flow control valves.

Manual/auto (M/A) controllers on the Main Control Board enable the operator to control flow manually from the Control Room. Upon automatic start of the MDAFW pumps, flow control valves PV-2453 A&B and PV-2454 A&B will automatically trip from manual to automatic control and position full open to ensure flow to the SGs. After a 10 second time delay the flow control valves can be manually positioned by the operator to adjust flow to the SGs. M/A controllers for these valves on the RSP enable the operator to control flow from the RSP when the RSP controllers are placed in manual. When in automatic, these controllers allow feed control to be accomplished at the Main Control Board.

A flow restricting orifice is provided downstream of each feed regulator valve. The orifice is designed to limit the maximum flow to a faulted SG to 700 gpm and prevent a pump runout condition.

MDAFWP ISOLATION VALVES

A check valve is located downstream of each feed regulator valve. RTDs in thermowells are provided on each discharge line just upstream of the check valve. These RTDs provide input to a dual indication temperature instrument with a range of 0-300°F located on CB-09. These temperature instruments are used to monitor for potential check valve backleakage from the Main Feedwater System and SGs into the AFW System piping.

A normally open, motor operated Containment isolation valve is located downstream of each check valve. Motor operated valves HV-2491A/B, -2492A/B, -2493A/B and -2494A/B are used to isolate AFW flow to the SGs. These valves are operated with two-position OPEN-CLOSE switches on the Main Control Board. Each switch operates two motor operated valves associated with the same SG. For example, 1-HS-2491 operates both 1-HV-2491A and 1-HV-2491B. The valve from both the MDAFW pump and the valve from the TDAFW Pump operate simultaneously to isolate AFW to one SG.

Comments / Reference: From LO21.SYS.AF1.LN, Page 17	Revision # 05/11/11

TDAFWP TURBINE SPEED GOVERNOR

The turbine speed governor is equipped with a pneumatically operated hydraulic speed changer mechanism which is used to control the turbine governor valve. The governor actuator is also driven by the turbine rotor via spiral reduction gears driven by the same worm gear as the oil pump. The governor servomotor receives oil from the actuator and is the lowest point in the oil system. To ensure proper operation, the oil level in the governor oil sight glass should be maintained at the levels specified in the operating procedures (U1 and U2 require slightly different levels). Obviously, a low oil level would damage the regulator or prevent operation, but high oil levels are also undesirable. This is because the oil may foam causing the regulator to operate improperly.

Comments / Reference: From LO21.SYS.AF1.LN, Page 9	Revision # 05/11/11

The tank is provided with a diaphragm to prevent oxygenation of the stored auxiliary feedwater. The tank is also provided with two 12" vents on top of the tank above the floating diaphragm for atmospheric relief. Nitrogen connections below the diaphragm allow for inerting the tank.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	077 G 2.4.1	
Importance Rating	4.6	

Generator Voltage and Electric Grid Disturbances: Emergency Procedures/Plan: Knowledge of EOP entry conditions and immediate action steps

Proposed Question: Common 53

Given the following conditions:

- The Unit 1 Reactor was tripped following a 345 kV grid disturbance.
- Immediate Actions of EOP-0.0A, Reactor Trip or Safety Injection are being performed.
- The Balance of Plant Operator reports the following:
  - Buses 1EA1 and 1EA2 are 6250 Volts and lowering.
  - Both buses are powered from their Alternate Offsite source.
- All other Immediate Action, Action/Expected Responses are Normal.

What is the appropriate action?

A. Transition to ECA-0.0A, Loss of All AC Power.

B. Transition to EOS-0.1A, Reactor Trip Response.

C. Perform 6900 V load shedding per ABN-602, Response to a 6900/480 V System Malfunction.

D. Perform 480 V load shedding per ABN-602, Response to a 6900/480 V System Malfunction.

Proposed Answer: A

Explanation:

A. Correct. Per EOP-0.0A, Step 3 if voltage is less than 6500 V on both busses a transition to ECA-0.0A is required.

B. Incorrect. Plausible as both buses are energized, that all Action/Expected Responses are satisfied, and a transition to EOS-0.1A is appropriate.

C. Incorrect. Plausible because when DC voltage is dropping, load shedding is performed to sustain the bus for as long as possible, however, per ABN-602A, the bus should be transferred to a different source.

D. Incorrect. Plausible because when DC voltage is dropping, load shedding is performed to sustain the bus for as long as possible, however, per ABN-602A, the bus should be transferred to a different source.

Technical Reference(s): EOP-0.0A, Step 3; ABN-602, Section 9 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the immediate operator actions of EOP-0.0, Reactor Trip or Safety Injection.

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 EOP-0.0A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 3 OF 115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Verify Reactor Trip:	
	a. Verify the following: • Reactor trip breakers - AT LEAST ONE OPEN -AND- • Neutron flux - DECREASING	a. Manually trip reactor from both trip switches. IF reactor will not trip, THEN momentarily de-energize 480V normal switchgear 1B3 AND 1B4. IF reactor NOT tripped, THEN go to FRS-0.1A, RESPONSE TO NUCLEAR POWER GENERATION/ATWT, Step 1.
	b. All control rod position rod bottom lights - ON	
2	Verify Turbine Trip: • All HP turbine stop valves - CLOSED	Manually trip turbine. IF the turbine will NOT trip, THEN pull-out all EHC fluid pumps. IF turbine still NOT tripped, THEN close or verify closed main steamline isolation valves.
3	Verify Power To AC Safeguards Busses:	
	a. AC safeguards busses - AT LEAST ONE ENERGIZED • AC safeguards bus voltage- 6900 Volts(6500-7100 Volts)	a. Go to ECA-0.0A, LOSS OF ALL AC POWER, Step 1.
	b. AC safeguards busses - BOTH ENERGIZED	b. Restore power to de-energized AC safeguards bus per ABN-601, RESPONSE TO A 138/345 KV SYSTEM MALFUNCTION or ABN-602, RESPONSE TO A 6900/480 VOLT SYSTEM MALFUNCTION when time permits.

Comments / Reference: From ABN-602, Step 9.3.5	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-602
RESPONSE TO A 6900/480V SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 53 OF 107

9.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Attachment 3 lists specific undervoltage response if needed.

		ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐	1	Check Blackout Sequencer - NOT OPERATED • OUTPUT - STEP TIME light - DARK • Automatic lockouts AL light - DARK	Recover from Blackout Sequencer operation per Section 8.0 while continuing.
☐	2	Verify Diesel Generators - BOTH STOPPED	IF the DG is NOT required to supply the bus, THEN stop any unloaded DG by placing DG EMER STOP/START handswitch in STOP OR run the DG per SOP-609A/B, Diesel Generator Startup if run unloaded greater than 30 minutes.
☐	3	Verify 138 KV Supply Voltage Normal - 135 KV - 144 KV	GO TO ABN-601, if not previously performed.
☐	4	Verify 345 KV Supply Voltage Normal - 340 KV - 361 KV	GO TO ABN-601, if not previously performed.
☐	5	Verify 6.9 KV and 480V Safeguard Bus Voltage - WITHIN LIMITS • 6.9 KV, 6480V - 7150V • 480V, 455V - 508V	Perform the following: a. Transfer affected 6.9 KV safeguard bus to available power supply per SOP-603A/B. b. IF either 6.9 KV OR 480V voltages NOT restored within limits, THEN GO TO Section 2.0.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	W/E04 EK2.2	
Importance Rating	3.8	

LOCA Outside Containment: Knowledge of the interrelations between LOCA Outside Containment 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 54

Given the following conditions:

- Unit 1 is responding to a Loss of Coolant Accident (LOCA) per ECA-1.2A, LOCA Outside Containment.
- The crew believes that the leak outside of Containment has been isolated but Reactor Coolant System pressure is not rising.

Which of the following alternate indications may be used to determine if the break has been isolated per ECA-1.2A, LOCA Outside Containment?

A. Refueling Water Storage Tank level stable.

B. Reactor Vessel Level Indicating System indication rising.

C. Emergency Core Cooling System flows rising.

D. Emergency Core Cooling System alignment verification.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because RWST inventory could be lost to the Safeguards Building from an interfacing system break outside Containment and RWST level stable could indicate that break flow has ceased, however, RWST level may still be lowering with the break isolated as ECCS flow refills the RCS. Therefore, RWST level change is not a good indicator of break isolation.

B. Correct. As stated in Attachment 2, Step 3 Bases, RCS pressure may not initially rise once the break is isolated, due to plant cooldown or when the RCS is saturated. RVLIS indication rising shows that ECCS flow is not leaving the RCS via the break, but rather it is refilling the RCS. This indicates that the break is isolated from the RCS.

C. Incorrect. Plausible because ECCS flows could indicate that the break is isolated if it were lowering. Rising ECCS flow is indicative of the RCS break becoming worse.

D. Incorrect. Plausible because an ECCS valve alignment is performed in Step 1 of ECA-1.2A, and this may isolate the break, however, verifying ECCS alignment alone does not ensure that the break is isolated. RCS parameters, such as pressure and RVLIS trend must be evaluated to verify break isolation.

Technical Reference(s)	ECA-1.2A, Attachment 2, Step 3 Bases ECA-1.2A, Steps 1, 2, & 3	Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the bases for operator actions, notes and cautions from ECA-1.2, LOCA Outside Containment.

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	2, 5, 10, 14
	55.43	

Comments / Reference: From ECA-1.2 A, Attachment 2, Step 3 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 6 OF 6

ATTACHMENT 2
PAGE 2 OF 2

BASES

STEP 3: This step instructs the operator to check RCS pressure to determine if the break has been isolated by previous actions. If the break is isolated in Step 2, a significant RCS pressure increase will occur due to the ECCS flow filling up the RCS with break flow stopped.

The operator transfers to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there may be no inventory in the sump.

It should be noted that for some breaks ECCS flow may cause an RCS pressure increase independent of break isolation. It should also be noted that for larger breaks, RCS repressurization may be delayed following break isolation. Additionally, if the RCS is saturated or a cooldown is in progress, RCS repressurization will proceed more slowly. Other means of verifying break isolation should be checked. For example increasing RVLIS trend due to injection flow, decreasing trends in local abnormal conditions and local observation (if practical) may be useful.

Comments / Reference: From ECA-1.2A, Steps 1 & 2	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 3 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] 1	Verify Proper Valve Alignment:	Manually close valve(s). IF valve(s) can NOT be manually closed, THEN locally close valve(s).
	a. RHRP 1 & 2 HL RECIRC ISOL VLVS - CLOSED • 1/1-8701A • 1/1-8702A • 1/1-8701B • 1/1-8702B	
	b. RHR TO HL 2 & 3 INJ ISOL VLV - CLOSED • 1/1-8840	
	c. SI TO HL INJ ISOL VLVS - CLOSED • 1/1-8802A • 1/1-8802B	
2	Identify And Isolate Break:	
	a. Sequentially close and open the following valves and monitor for an RCS pressure increase: 1) RHR TO CL INJ ISOL VLVS: • 1/1-8809A • 1/1-8809B 2) SI to CL 1•4 INJ ISOL VLV • 1/1-8835	

Comments / Reference: From ECA-1.2A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 4 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Check If Break Is Isolated:	
	a. RCS pressure - INCREASING	a. Go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	

-END-

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	W/E05 G 2.2.3	
Importance Rating	3.8	

Inadequate Heat Transfer - Loss of Secondary Heat Sink: Equipment Control: (multi-unit) Knowledge of the design, procedural, and operational differences between units

Proposed Question: Common 55

Given the following conditions:

- Unit 2 has experienced a Loss of All Feedwater Flow following a Reactor Trip from 100% power.
- 1/2-8000A, PRZR PORV BLK VLV, was previously closed due to 1/2-PCV-455A, PRZR PORV, seat leakage.
- FRH-0.1B, Response to Loss of Secondary Heat Sink, is in progress.
- Power can NOT be restored to 480V MCC 2EB3-2 which provides power to 1/2-8000A, PRZR PORV BLK VLV.

Which of the following is required per FRH-0.1B, Response to Loss of Secondary Heat Sink?

In order for Bleed and Feed to be initiated, Steam Generator (SG) wide range levels must be less than…

A. 27% in at least 3 SGs. Adequate core cooling will be achieved during Bleed and Feed operations.

B. 35% in at least 3 SGs. Adequate core cooling will be achieved during Bleed and Feed operations.

C. 27% in at least 3 SGs. Inadequate core cooling will be experienced during Bleed and Feed operations without opening Reactor Head and Pressurizer vents.

D. 35% in at least 3 SGs. Inadequate core cooling will be experienced during Bleed and Feed operations without opening Reactor Head and Pressurizer vents.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because 27% is the required Steam Generator level for Unit 2, however, adequate core cooling will not be achieved during Bleed and Feed operations.
B. Incorrect. Plausible if thought that 35% is the required Steam Generator level for Unit 2 and that adequate core cooling would be achieved during Bleed and Feed operations with only one PORV available.
C. Correct. This is the required Steam Generator level for Unit 2. Inadequate core cooling will be experienced without opening Reactor Head and Pressurizer vents because only one PORV is available.
D. Incorrect. Plausible because inadequate core cooling will be experienced without opening Reactor Head and Pressurizer vents, however, a Steam Generator level of 35% is required for Unit 1.

Technical Reference(s) FRH-0.1B, Steps 3, 20, & 21
FRH-0.1A, Steps 3, 20, & 21
Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRH-0.1, Response to Loss of Secondary Heat Sink.

Question Source: Bank # X
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 FRH-0.1B, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRH-0.1B
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 4 OF 64

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Check CCP Status - BOTH AVAILABLE	Immediately perform the following: a. STOP ALL RCPs. b. Verify power to PRZR PORV block valves - AVAILABLE Locally restore power to block valve(s). c. Go to Step 12. OBSERVE CAUTION PRIOR TO STEP 12.
* 3	Check Bleed And Feed - REQUIRED:	
	a. Check the following:	a. Go to Step 4.
	• Actual wide range level (per Attachment 2) in at least 3 SGs - LESS THAN 27% (30% FOR ADVERSE CONTAINMENT)	

Comments / Reference: From FRH-0.1B, Steps 20 & 21	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRH-0.1B
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 23 OF 64

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1D] 19	Establish Instrument Air And Nitrogen To Containment:	
	a. Establish instrument air:	
	1) Verify air compressor running. -AND-	1) Manually start air compressor and align valve as appropriate.
	2) Establish instrument air to containment.	
	b. Establish nitrogen:	
	1) Verify ACCUM 1•4 VENT CTRL, 2-HC-943 - CLOSED	1) Manually close valve.
	2) Open SI/PORV ACCUM N_2 ISOL VLV, 1/2-8880	
20	Establish RCS Bleed Path:	
	a. Verify power to PRZR PORV block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. Verify PRZR PORV block valves - BOTH OPEN	b. Manually open both block valve(s).
	c. Open PRZR PORVs.	
21	Verify Adequate RCS Bleed Path:	Perform the following:
	• PRZR PORVs - BOTH OPEN	a. Open vents on reactor vessel head and on the PRZR to containment.
	• PRZR PORV block valves- BOTH OPEN	b. Align any available low pressure water source to the SG(s). <u>IF</u> no low pressure water source can be aligned, <u>THEN</u> go to Step 22.

Comments / Reference: From FRH-0.1A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 4 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 2	Check CCP Status - BOTH AVAILABLE	Immediately perform the following: a. STOP ALL RCPs. b. Verify power to PRZR PORV block valves - AVAILABLE Locally restore power to block valve(s). c. Go to Step 12. OBSERVE CAUTION PRIOR TO STEP 12.
* 3	Check Bleed And Feed - REQUIRED:	
	a. Check the following:	a. Go to Step 4.
	• Actual wide range level (per Attachment 2) in at least 3 SGs - LESS THAN 35% (40% FOR ADVERSE CONTAINMENT)	

Comments / Reference: From FRH-0.1A, Step 20 & 21	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRH-0.1A
RESPONSE TO LOSS OF SECONDARY HEAT SINK	REVISION NO. 8	PAGE 19 OF 60

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[1D] 19	Establish Instrument Air And Nitrogen To Containment:	
	a. Establish instrument air:	
	1) Verify air compressor running. -AND-	1) Manually start air compressor and align valve as appropriate.
	2) Establish instrument air to containment.	
	b. Establish nitrogen:	
	1) Verify ACCUM 1•4 VENT CTRL, 1-HC-943 - CLOSED	1) Manually close valve.
	2) Open SI/PORV ACCUM N_2 ISOL VLV, 1/1-8880	
20	Establish RCS Bleed Path:	
	a. Verify power to PRZR PORV block valves - AVAILABLE	a. Locally restore power to block valve(s).
	b. Verify PRZR PORV block valves - BOTH OPEN	b. Manually open both block valve(s).
	c. Open PRZR PORVs.	
21	Verify Adequate RCS Bleed Path:	Perform the following:
	• PRZR PORVs - BOTH OPEN	a. Open vents on reactor vessel head and on the PRZR to containment.
	• PRZR PORV block valves- BOTH OPEN	b. Align any available low pressure water source to the SG(s). IF no low pressure water source can be aligned, THEN go to Step 22.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	W/E11 EA1.3	
	Importance Rating	3.7	

Loss of Emergency Coolant Recirculation: Ability to operate and/or monitor the following as they apply to a Loss of Emergency Coolant Recirculation: Desired operating results during abnormal and emergency situations

Proposed Question: Common 56

Given the following condition:

- ECA-1.2A, LOCA Outside Containment, Step 3, directs checking Reactor Coolant System pressure to determine if the break has been isolated by previous actions.

If the break has NOT been isolated, which of the following identifies the effect that a transition to ECA-1.1A, Loss of Emergency Coolant Recirculation, has on mitigating the accident?

Actions are taken to...

A. ...transfer Safeguards Building Sump contents to the Refueling Water Storage Tank as directed by Technical Support Center.

B. ...minimize Refueling Water Storage Tank depletion by reducing total injection flow.

C. ...increase the injection flow rate to restore Reactor Coolant System pressure.

D. ...stabilize Reactor Coolant System pressure to prevent the Safety Injection Accumulators from discharging out the break.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this action could be performed to recover lost sump water, however, it is not directed by ECA-1.1A.
B. Correct. Reducing injection flow will minimize RWST depletion.
C. Incorrect. Plausible if thought RCS pressure is restored, however, actions are taken to restore RCS mass in ECA-1.1A.
D. Incorrect. Plausible because multiple ERGs depressurize SI Accumulators, however, ECA-1.1A utilizes SI Accumulator inventory.

Technical Reference(s): ECA-1.1A, Steps 17 & 22; ECA-1.2A, Step 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural step, or sequence of steps from ECA-1.1, Loss of Emergency Coolant Recirculation, **STATE** the purpose/basis for the step(s).

Question Source:
Bank # X
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 2, 10
55.43 ______

Comments / Reference: From ECA-1.1A, Step 17	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 9 OF 79

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
16	Check If ECCS Is In Service: • SI pumps - ANY RUNNING -OR- • CCP injection line isolation valves - OPEN -OR- • RHR pumps - ANY RUNNING IN INJECTION MODE	Go to Step 25.
17	Establish One Train Of ECCS Flow:	
	a. CCP - ONLY ONE RUNNING	a. Start or stop CCP to establish only one pump running.
	b. SI pump - ONLY ONE RUNNING	b. Start or stop SI pump to establish only one pump running.
	c. RCS pressure - LESS THAN 325 PSIG (425 PSIG FOR ADVERSE CONTAINMENT)	c. Stop RHR pumps. Go to Step 18.
	d. RHR pump - ONLY ONE RUNNING	d. Start or stop RHR pump to establish only one pump running.

Comments / Reference: From ECA-1.1A, Step 22	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.1A
LOSS OF EMERGENCY COOLANT RECIRCULATION	REVISION NO. 8	PAGE 12 OF 79

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*20	Check If ECCS Can Be Terminated:	
	a. RVLIS indication - GREATER THAN OR EQUAL TO 11 IN ABOVE CORE PLATE LIGHT LIT	a. Go to Step 25.
	b. RCS subcooling - GREATER THAN 75°F (105°F FOR ADVERSE CONTAINMENT)	b. Establish minimum ECCS flow to remove decay heat. Perform the following: 1) Determine minimum ECCS flow required from Table in Attachment 5. 2) Establish minimum ECCS flow. 3) Go to Step 25.
21	Establish Instrument Air And Nitrogen To Containment:	
	a. Establish Instrument air:	
	1) Verify air compressor running.	1) Manually start one air compressor and align valve as appropriate.
	2) Manually establish instrument air to containment.	
	b. Establish Nitrogen:	
	1) Verify ACCUM 1•4 VENT CTRL, 1-HC-943 - CLOSED	1) Manually close valve.
	2) Open SI/PORV ACCUM N2 ISOL VLV, 1/1-8880.	
22	Stop ECCS Pumps And Place In Standby: • SI pumps • RHR pumps • All but one CCP	

Comments / Reference: From ECA-1.2A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 4 OF 6

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
3	Check If Break Is Isolated:	
	a. RCS pressure - INCREASING	a. Go to ECA-1.1A, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.
	b. Go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.	

-END-

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	001 AK1.12	
Importance Rating	2.8	

Continuous Rod Withdrawal: Knowledge of the operational implications of the following concepts as they apply to the Continuous Rod Withdrawal: Long-range effects of core quadrant power tilt

Proposed Question: Common 57

Given the following conditions:

- Unit 1 was at 28% power.
- A continuous withdrawal of Control Bank D, Rod D4 occurred.
- The Reactor Protection System failed to automatically trip the Reactor.
- FRS-0.1A, Response to Nuclear Power Generation / ATWT, is in progress.
- The Reactor was taken subcritical by manually inserting control rods.

Which of the following explains the possible effect of the flux tilt created by the rod withdrawal?

A. High local power density in the affected quadrant resulting in fuel damage.

B. High local power density in the opposite quadrant resulting in fuel damage.

C. Overheated fuel in the affected quadrant due to elevated fission product concentration.

D. Overheated fuel in the opposite quadrant due to elevated fission product concentration.

Proposed Answer: A

Explanation:

A. Correct. Power density at the withdrawn rod could cause Linear Heat Rate (LHR) to exceed its limit. The fuel may overheat and cause fuel damage.
B. Incorrect. Plausible because high power density will occur, however, power will peak in the affected quadrant vice the opposite quadrant.
C. Incorrect. Plausible because fuel overheating may occur, however, overheating is caused by high Linear Heat Rate (LHR) not fission product concentration buildup.
D. Incorrect. Plausible because fuel overheating may occur, however, overheating is caused by high Linear Heat Rate (LHR) not fission product concentration buildup. Additionally, fuel overheating will occur in the affected quadrant vice the opposite quadrant.

Technical Reference(s) Technical Specification 2.1.1 Bases Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRS-0.1A/B, Response To Nuclear Generation/ATWT.
APPLY the administrative requirements of the Reactor Protection and Engineered Safeguard Actuation Systems including Technical Specifications, TRM and ODCM.

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, 10
55.43 ________

Comments / Reference: From Technical Specification 2.1.1Bases	Revision # 67

Reactor Core SLs
B 2.1.1

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (the 95/95 DNB criterion) that DNB will not occur and by requiring that fuel centerline temperature stays below the melting temperature.

The restrictions of this SL prevent overheating of the fuel and cladding, as well as possible cladding perforation, that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	028 AA1.06	
Importance Rating	3.3	

Pressurizer Level Control Malfunction: Ability to operate and/or monitor the following as they apply to Pressurizer Level Control Malfunctions: Checking of RCS leaks

Proposed Question: Common 58

Given the following conditions:

- Unit 1 is at 100%.
- Pressurizer level is stable.
- 1-TI-126, Regenerative Heat Exchanger Letdown Outlet temperature is 250ºF.
- 1-TI-127, Regenerative Heat Exchanger Charging Return temperature is 500ºF.
- 1-PI-131, Letdown Heat Exchanger Outlet pressure is 310 psig.

A Chemical and Volume Control System malfunction occurs resulting in the following conditions:

- Pressurizer level is lowering.
- 1-TI-126, Regenerative Heat Exchanger Letdown Outlet temperature is 390ºF.
- 1-TI-127, Regenerative Heat Exchanger Charging Return temperature is 540ºF.
- 1-PI-131, Letdown Heat Exchanger Outlet pressure is 310 psig.

Which of the following malfunctions would explain the observed change in indications?

A. A Charging Header leak upstream of the Regenerative Heat Exchanger.

B. A Letdown leak upstream of the Regenerative Heat Exchanger.

C. Loss of air to 1-FCV-0121, Charging Header Flow Control Valve.

D. Loss of air to 1-PCV-131, Letdown Pressure Control Valve.

Proposed Answer: A

Explanation:

A. Correct. A leak in this location would divert cooling flow (Charging) from the Regenerative Heat Exchanger resulting in higher Letdown temperatures out of the Regenerative Heat Exchanger and the Charging temperature returning to the loop would also be higher than normal and also result in lowering Pressurizer level.

B. Incorrect. Plausible because it could be thought that this leak location would result in these indications, however, a leak in Letdown upstream of the Regenerative Heat Exchanger would reduce the flow though the Heat Exchanger and result in a lower rather than higher temperature.

C. Incorrect. Plausible if thought that this failure reduced the flow of Charging and caused higher Letdown temperatures from the Regenerative Heat Exchanger and lowering Pressurizer level, however, the valve fails OPEN which would yield lowering Letdown temperatures from the Regenerative Heat Exchanger and rising Pressurizer level.

D. Incorrect. Plausible because this failure would result in the higher Letdown temperatures from the Regenerative Heat Exchanger and lowering Pressurizer level, however, on a loss of air to the Letdown Pressure Control Valve the Letdown pressure would be much lower than normal instead of at or slightly above normal pressure.

Technical Reference(s): LO21.SYS.CS1.LN, Pages 12, 18 & 40; ALM-0061A, 1-ALB-6A, Window 1.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Chemical and Volume Control System.

Question Source:
Bank # X
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, 7
55.43 ______

Comments / Reference: From LO21.SYS.CS1.LN, Page 12	Revision # 04/28/11

REGENERATIVE HEAT EXCHANGER

The regenerative heat exchanger is a stainless steel heat exchanger, located in containment in a shielded room on the 832' elevation. It is a counter flow design with charging return flow through the tube side and letdown flow on the shell side. Letdown enters the heat exchanger at reactor coolant system cold leg temperature and pressure and passes through baffles, which create crossflow to increase the contact between the hot letdown and the relatively cold charging flow. This allows more heat transfer between the two fluids. By utilizing a regenerative heat exchanger, the amount of heat lost from the primary system and the thermal stresses that exist at the loop charging penetrations are reduced.

During normal power operations the regenerative heat exchanger reduces letdown temperature from 560°F to approximately 260°F and raises charging water temperature to approximately that of the reactor coolant system cold loop.

Temperature of the charging flow leaving the regenerative heat exchanger is provided by thermowell-mounted RTD and indicated on CB-06 (u-TI-0126, 100-600°F) and on the plant computer.

Temperature of the letdown flow leaving the regenerative heat exchanger is provided by thermowell-mounted RTD and indicated on CB-06 (u-TI-0127, 100-600°F) and on the plant computer. This device also actuates an alarm (REGEN HX LTDN OUT TEMP HI) at ≥400°F.

Comments / Reference: From LO21.SYS.CS1.LN, Page 18	Revision # 04/28/11

LETDOWN HEAT EXCHANGER OUTLET PRESSURE CONTROL VALVE

Letdown Heat Exchanger Outlet Pressure Control Valve, u-PCV-0131 is located downstream of the letdown heat exchanger, in the letdown heat exchanger valve room. During normal power operations the valve functions to automatically maintain upstream pressure at the established setpoint, which is normally 310 psig. By maintaining a back-pressure on the system, the hot letdown fluid between the letdown orifices and the letdown heat exchanger is maintained in a subcooled condition.

The saturation temperature of water at 310 psig is approximately 425°F. The temperature of letdown leaving the regenerative heat exchanger is normally 260°F, which is approximately 165°F below saturation temperature. An alarm (REGEN HX LTDN OUT TEMP HI, set at ≥400°F) will warn of temperature approaching saturation temperature for this portion of the letdown piping.

Letdown Heat Exchanger Outlet Pressure Control Valve, u-PCV-0131, is air operated and fails fully open on a loss of air or control power.

Comments / Reference: From LO21.SYS.CS1.LN, Page 40	Revision # 04/28/11

CENTRIFUGAL CHARGING PUMP FLOW CONTROL

Charging Flow Element u-FE-0121 is located on the combined charging pump discharge header in the charging pump valve room. It provides the differential pressure which is related to charging flow to Charging Pump Discharge Flow Transmitter u-FT-0121 (located, for Unit 1, in the boric acid storage tank room and, for Unit 2, in the Unit 2 CVCS valve operating room on the 822 foot elevation of the auxiliary building. The flow transmitter functions to generate a current signal which is proportional to charging header flow (from 0 to 270 gpm) for indication and control of charging flow. Charging Flow Indicators u-FI-0121A and u-FI-0121B (0 to 270 gpm) are located on CB-06 and at the remote shutdown panel, respectively. Charging flow is provided as an input to the plant computer. The CHG FLO HI/LO alarm on CB-06 also receives its inputs from the charging flow transmitter and is set to actuate at ≥ 150 gpm and at ≤ 55gpm.

The discharge flow from the centrifugal charging pumps to the normal charging header and to the reactor coolant pump seal injection lines is controlled by regulating the position of CCP u-01/u-02 Charging Flow Control Valve, u-FCV-0121, located in the charging pump valve room. This valve is air operated, and fails open on loss of air or control power.

Comments / Reference: From ALM-0061A, 1-ALB-6A, Window 1.4	Revision # 7

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0061A
ALARM PROCEDURE 1-ALB-6A	REVISION NO. 7	PAGE 15 OF 79

ANNUNCIATOR NOM./NO.: **REGEN HX LTDN OUT TEMP HI** **1.4**

PROBABLE CAUSE:

Inadequate charging flow
Charging pump malfunction

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	060 G 2.4.45	
	Importance Rating	4.1	

Accidental Gaseous Radwaste Release: Emergency Procedures/Plan: Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question: Common 59

Given the following conditions:

- Unit 2 is at 100% power.
- A gaseous release is in progress.
- The Digital Radiation Monitoring System (PC-11) receives a high radiation (RED) signal on X-RE-5701, Auxiliary Building Ventilation Duct Radiation Monitor.

Which of the following describes the expected automatic action initiated due to the high radiation alarm?

A. Any Containment Vent is terminated.

B. Any Gas Decay Tank release is terminated.

C. Containment Ventilation Isolation is actuated.

D. Control Room Emergency Recirculation is actuated.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that the Containment Vent release was sampled by the Auxiliary Building Ventilation Duct Radiation Monitor.
B. Correct. A high radiation signal from X-RE-5701 automatically closes HCV-014 which terminates any waste gas release that may be in progress.
C. Incorrect. Plausible because Containment Ventilation Isolation is caused by a high radiation signal, but the signal is from 2-RE-5502 or 2-RE-5503, the Containment Air Particulate and Gaseous channels.
D. Incorrect. Plausible because Control Room Emergency Recirculation is caused by a high radiation signal, but the signal is from X-RE-5895A/B or X-RE-5896A/B, the Control Room Air Supply gas channel.

Technical Reference(s) ABN-902, Section 2.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operations of the Gaseous Waste Systems.

EXPLAIN the instrumentation and controls of the Digital Radiation Monitoring System and **PREDICT** the system response.

Question Source: Bank # X

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 ABN-902, Sections 2.1 & 2.2	Revision # 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ABN-902
RELEASE OF RADIOACTIVE/TOXIC GAS	REVISION NO. 7	PAGE 3 OF 23

2.0 RELEASE OF RADIOACTIVE GAS

2.1 Symptoms

a. Annunciator Alarms

- None

b. Plant Indications

- An unexpected increase on any of the following Process Radiation Monitors:
  - u-RE-5502, CNTMT AIR PIG PART (CAPu98)
  - u-RE-5566, CONTMT AIR PIG IODINE (CAIu99)
  - u-RE-5503, CNTMT AIR PIG GAS (CAGu97)
  - X-RE-5570A, S WRGM EFFLUENT (PVF684)
  - X-RE-5570B, N WRGM EFFLUENT (PVF685)
  - X-RE-5701, AUX BLDG VENT DUCT (ABV089)
  - X-RE-5702, HVAC ROOM VENT DUCT (HVV090)

Comments / Reference: From ABN-902, Sections 2.1 & 2.2	Revision # 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ABN-902
RELEASE OF RADIOACTIVE/TOXIC GAS	REVISION NO. 7	PAGE 4 OF 23

2.1 b.

- u-RE-2325A, MSL #1 LEAK RATE MONITOR (N16u74)
- u-RE-2326A, MSL #2 LEAK RATE MONITOR (N16u75)
- u-RE-2327A, MSL #3 LEAK RATE MONITOR (N16u76)
- u-RE-2328A, MSL #4 LEAK RATE MONITOR (N16u77)
- X-RE-5567A, S VENT STACK DISCHARGE NOBLE GAS (PVG384)
- X-RE-5567B, N VENT STACK DISCHARGE NOBLE GAS (PVG385)
- X-RE-5895A, CR HVAC, N VENT (CRV053)
- X-RE-5895B, CR HVAC, N VENT (CRV054)
- X-RE-5896A, CR HVAC, S VENT INTK (CRV091)
- X-RE-5896B, CR HVAC, S VENT (CRV092)

2.2 Automatic Actions

- "HIGH RADIATION" alarm on the Containment Particulate OR Gaseous Monitor initiates a Containment Ventilation Isolation Signal.
- "HIGH RADIATION" alarm on Plant Vent Stack Wide Range Noble Gas Monitor will close X-HCV-0014 in the Gaseous Waste Processing System.
- "HIGH RADIATION" alarm on any Control Room Duct Monitor will shift Control Room Ventilation to the Emergency Recirculation Mode.
- "HIGH RADIATION" alarm on Auxiliary Building Vent Duct Monitor will close X-HCV-0014 in the Gaseous Waste Processing System.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	067 AA2.08	
	Importance Rating	2.9	

Plant Fire on Site: Ability to determine and interpret the following as they apply to the Plant Fire on Site: Limits of affected area

Proposed Question: Common 60

Given the following conditions:

- A fire has been reported in the Nuclear Operations Support Facility (NOSF).
- The Fire Brigade has been dispatched due to the Plant Simulator being housed in the NOSF.

Which of the following describes the actions required following dispatch of the Fire Brigade to the NOSF?

The [1] shall request assistance from [2] to restore the Fire Brigade compliment inside the Protected Area as soon as possible.

	[1]	[2]
A.	Shift Manager	Glen Rose Fire Department
B.	Fire Brigade Leader	Glen Rose Fire Department
C.	Shift Manager	Hood County Fire Department
D.	Fire Brigade Leader	Hood County Fire Department

Proposed Answer: A

Explanation:

A. Correct. The Shift Manager is responsible to ensure that the Fire Brigade inside the Protected Area is restored to acceptable strength as soon as possible and Glen Rose is the CPNPP designated backup Fire Department.

B. Incorrect. Plausible because Glen Rose is the CPNPP designated backup Fire Department, however, the Shift Manager is responsible to restore the Fire Brigade.

C. Incorrect. Plausible because the Shift Manager is responsible to restore the Fire Brigade, however, the Glen Rose Fire Department is the CPNPP designated backup Fire Department.

D. Incorrect. Plausible if thought that the Fire Brigade leader requests backup and that Hood County is the CPNPP designated backup Fire Department.

Technical Reference(s) STA-727, Step 6.3.4 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: During unit evolutions, **DIRECT** shift personnel actions and ENSURE proper and effective communications are maintained.

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 STA-727, Step 6.3.4	Revision # 5

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-727
FIRE BRIGADE	REVISION NO. 5 INFORMATION USE	PAGE 16 OF 19

6.3.4 Fire Brigade Response to Fires Outside the Protected Area

CAUTION :

- The importance of these buildings in relation to continued plant operation due to the Simulator, spare parts, and radioactive materials/waste storage is such that Fire Brigade response to these areas is vital.
- The response to these areas must be weighed at the time of the event in relation to plant conditions and available personnel to determine if the Fire Brigade will respond. This determination will be made by the Shift Manager/Unit Supervisor.
- When the decision is made to dispatch the Fire Brigade to any fire outside the Protected Area, the Shift Manager shall request assistance from off-duty Fire Brigade members to restore the standby Fire Brigade compliment inside the Protected Area as soon as possible.

6.3.4.1 NOSF, Warehouse ‘A’, ‘B’, or ‘C’, ISFSI Pad or Heavy Haul Path during TRANSPORT OPERATIONS.

a. If a call is received from security stating that there is a fire alarm, or a message is received from another party stating that there is a fire in the NOSF, Warehouse ‘A’, ‘B’ or ‘C’, ISFSI Pad or Heavy Haul Path during TRANSPORT OPERATIONS, the Shift Manager/Unit Supervisor should send the Perimeter NEO as a Brigade Recon NEO to investigate the situation.

b. Preparations should be made to send the Fire Brigade to the affected area for fire fighting activities if the Recon NEO reports a fire.

c. Contact the Glen Rose, Somervell County Volunteer Fire Department for assistance and designate a resource staging location for the off-site response.

d. Inform Security that off-site assistance has been requested and of the resource staging location for the off-site response.

e. If the Fire Brigade is dispatched outside the Protected Area, restore the Fire Brigade compliment inside the Protected Area as soon as possible.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	068 AA2.09	
	Importance Rating	4.1	

Control Room Evacuation: Ability to determine and interpret the following as they apply to the Control Room Evacuation: Saturation margin

Proposed Question: Common 61

Given the following conditions:

- The Unit 1 Control Room has been evacuated per ABN-905A, Loss of Control Room Habitability.
- A plant cooldown from the Remote Shutdown Panel has been initiated with the following limits:
  - Subcooling greater than 65°F.
  - Actual Pressurizer Level - 25% to 50%.
  - Actual Steam Generator Level - 84% to 92%.

Which of the following describes how the above limits are monitored per ABN-905A, Loss of Control Room Habitability?

Subcooling is determined by __________, Actual Pressurizer Level is determined by __________, and Actual Steam Generator level is determined by __________.

A. reading RCS Saturation meter
calculation using temperature correction of indicated level
calculation using temperature correction of indicated level

B. calculation using indicated temperature and pressure
calculation using temperature correction of indicated level
calculation using temperature correction of indicated level

C. reading RCS Saturation meter
reading Pressurizer level meter
reading Steam Generator level meter

D. calculation using indicated temperature and pressure
reading Pressurizer level meter
reading Steam Generator level meter

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Pressurizer and Steam Generator levels are calculated using temperature correction curves of indicated level per Attachments 17 and 16 respectively, however, there is no RCS Saturation meter at the Remote Shutdown Panel.
B. Correct. Subcooling must be calculated using RCS temperature and pressure at the Remote Shutdown Panel per Attachment 13. Pressurizer and Steam Generator levels are calculated using temperature correction curves of indicated level per Attachments 17 and 16 respectively.
C. Incorrect. Plausible if thought that there is an RCS Saturation meter at the Remote Shutdown Panel and temperature corrections must be applied for Pressurizer and Steam Generator levels.
D. Incorrect. Plausible because Subcooling must be calculated using RCS temperature and pressure at the Remote Shutdown Panel, however, temperature corrections must be applied for Pressurizer and Steam Generator levels.

Technical Reference(s): ABN-905A, Step 2.3.58; ABN-905A, Attachments 13, 16, & 17 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Loss Of Control Room Habitability per ABN-905, Loss Of Control Room Habitability.

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 ABN-905A, Step 2.3.58	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 16 OF 74

2.3 Operator Actions

NOTE:
- IPO-009A may be referred to for general guidance on securing the secondary plant.
- SDM calculations need not all be completed prior to continuing.

☐ 55. Initiate a Shutdown Margin (SDM) Calculation per OPT-301 (uncorrected minimum Boron Concentration) for the following temperature plateaus while continuing:

- 500°F Required boron concentration ________ ppm
- 400°F Required boron concentration ________ ppm
- 300°F Required boron concentration ________ ppm
- 200°F Required boron concentration ________ ppm

☐ 56. WHEN plant cooldown is desired, THEN continue this procedure.

☐ 57. Borate to desired temperature plateau uncorrected minimum boron concentration per OPT-301, using Attachment 12 of this procedure.

CAUTION:
- Low steam line pressure SI may occur at 605 psig if not blocked.
- Low pressurizer pressure SI may occur at 1820 psig if not blocked.

NOTE: When RCS pressure is approximately 1900 psig, SI may be blocked by I&C per Attachment 6.

☐ 58. During cooldown, maintain the following limits:

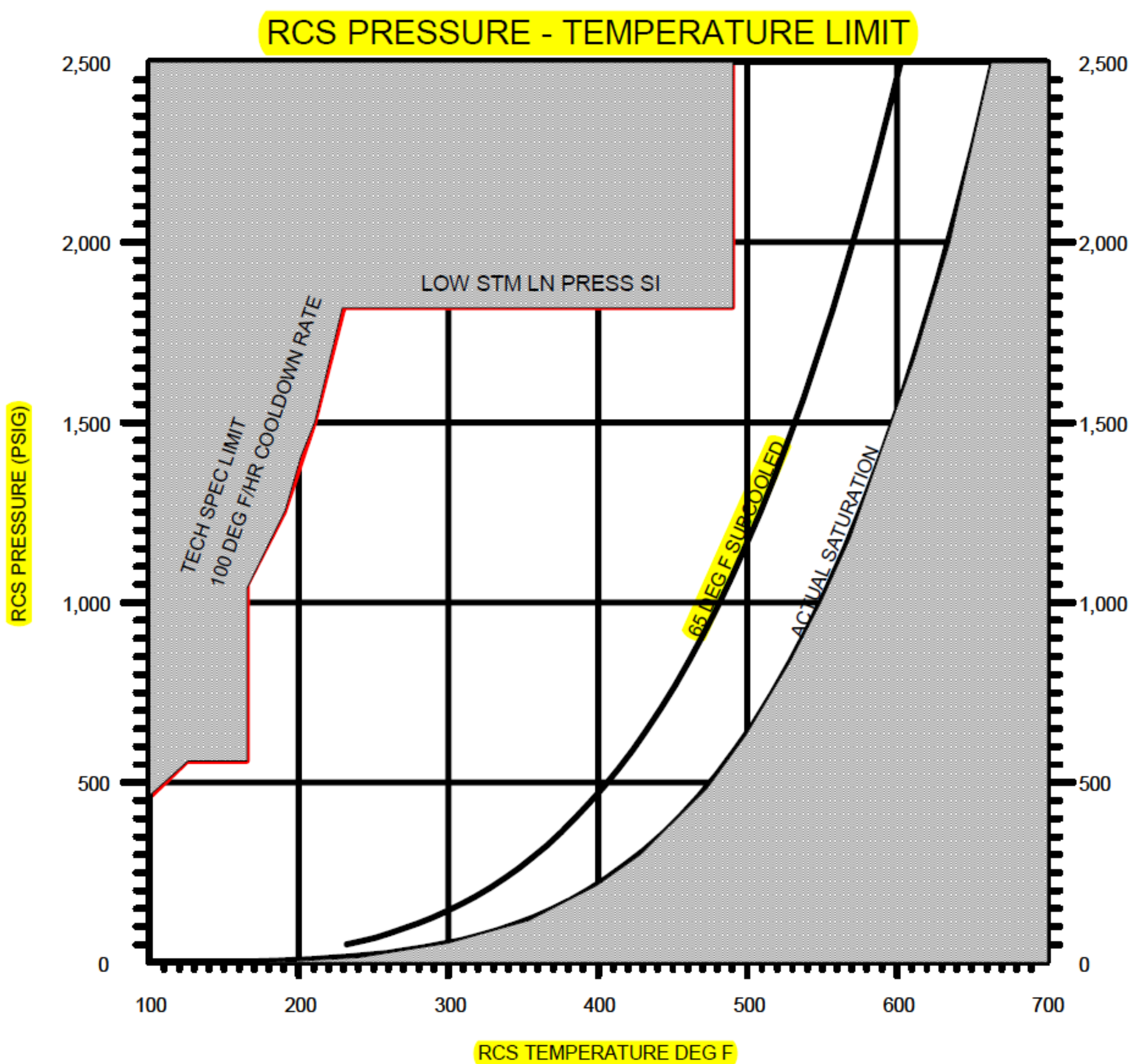
- Subcooling >65°F (Attachment 13)
- Actual PRZR LVL - 25% to 50% (Attachment 17)
- Actual SG LVL - 84% to 92% (Attachment 16)

Comments / Reference: From ABN-905A, Attachment 13	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 56 OF 74

ATTACHMENT 13
PAGE 1 OF 1

RCS PRESSURE - TEMPERATURE LIMIT



Comments / Reference: From ABN-905A, Attachment 16	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 59 OF 74

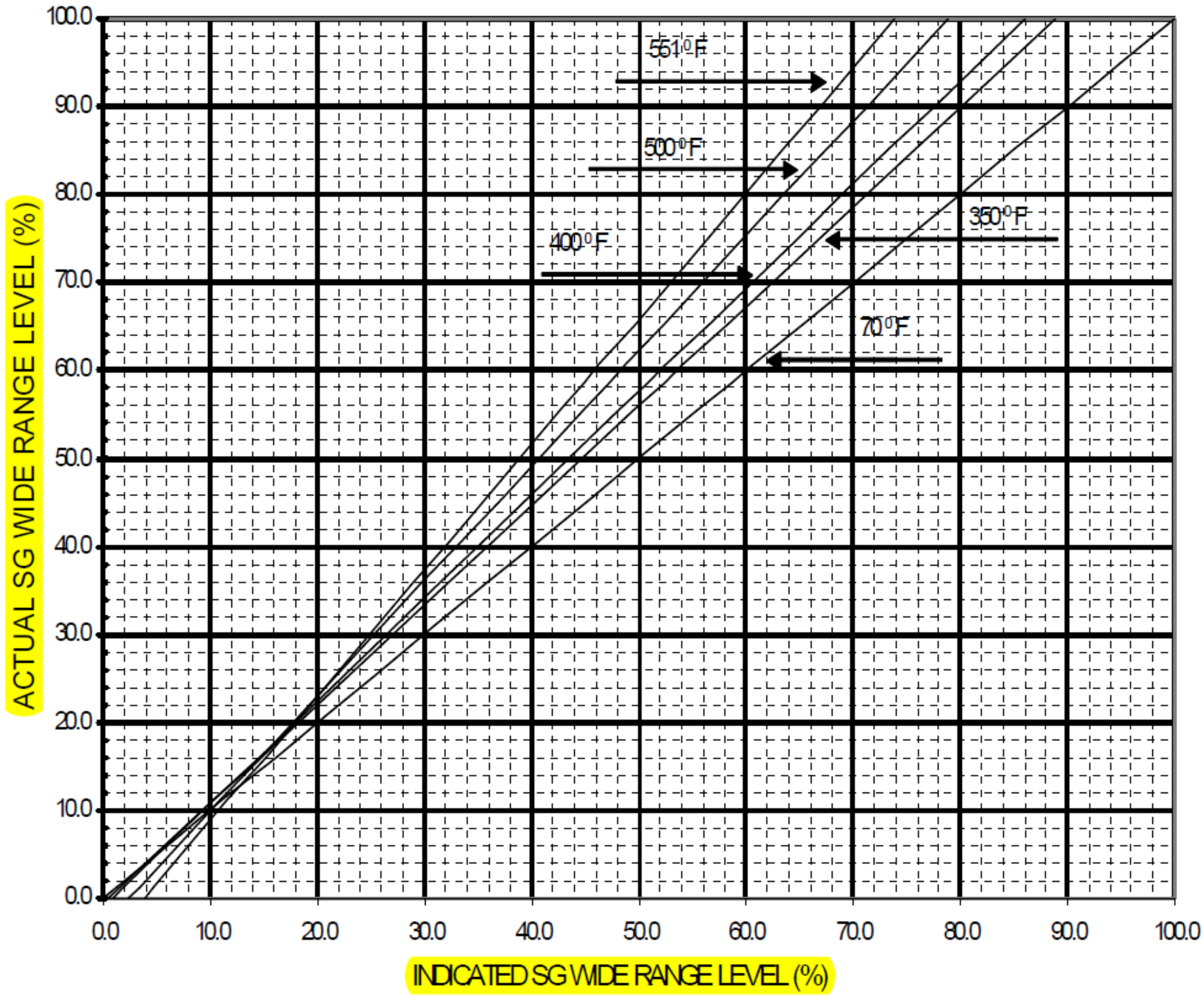
ATTACHMENT 16
PAGE 1 OF 1

SG LEVEL TEMPERATURE CORRECTION

NOTE: Normal SG level for Hot Standby and Cooldown (60 - 75% NR) is between 83% and 90% actual wide range. Operating outside this range could cause uncovering AFW nozzle OR ESF actuation OR moisture carryover. Approximate critical levels (actual wide range) are:

- Lo-Lo (ESF actuation) Unit 1 - 74%
- AFW Nozzle Unit 1 - 83%
- Hi-Hi (moisture carryover) 92%

(L)



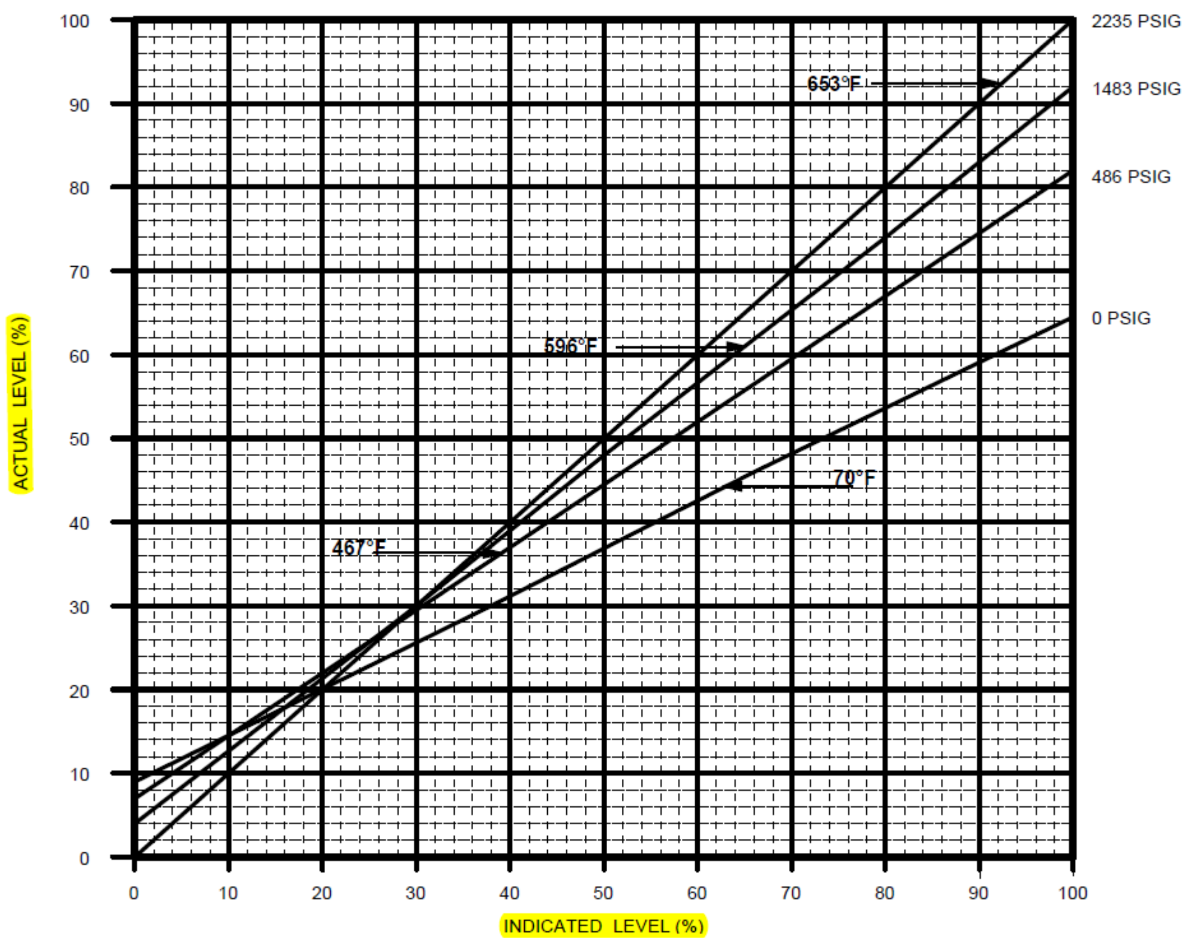
Comments / Reference: From ABN-905A, Attachment 17	Revision # 9

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ABN-905A
LOSS OF CONTROL ROOM HABITABILITY	REVISION NO. 9	PAGE 60 OF 74

ATTACHMENT 17
PAGE 1 OF 1

PRZR LEVEL TEMPERATURE CORRECTION

PRESSURIZER LEVEL CHANNEL
(Hot Calibrated)
(LI-459B, LI-460B)



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076 AK2.01	
	Importance Rating	2.6	

High Reactor Coolant Activity: Knowledge of the interrelations between High Reactor Coolant Activity and the following: Process radiation monitors

Proposed Question: Common 62

Given the following conditions:

- Unit 1 has experienced a problem with the Volume Control Tank (VCT).
- Charging Pump suction has been shifted to the Refueling Water Storage Tank (RWST) per SOP-103A, Chemical and Volume Control System.
- Several hours after shifting Charging Pump suction, 1-RE-0406 (FFL106), Failed Fuel Monitor indication has begun rising at an increasing rate.
- Chemistry has sampled the Reactor Coolant System and determined that Mn-54 and Mn-56 levels are increasing.

Which of the following is the probable cause for the increasing reading on FFL-160, Failed Fuel Monitor?

A. Oxygen induced CRUD burst.

B. Hydraulically induced CRUD burst.

C. Oxygen induced cladding creep.

D. Chemically induced cladding pin-holing.

Proposed Answer: A

Explanation:

A. Correct. An increase in Mn-54 and Mn-56 are the result of oxygen induced CRUD burst from shifting to the Refueling Water Storage Tank.

B. Incorrect. Plausible because a CRUD burst is in progress, however, a hydraulically induced CRUD burst is associated with Rod movement due to contamination of CRDM latch assemblies.

C. Incorrect. Plausible because cladding creep is a phenomenon exhibited on fuel assemblies, however, it is not oxygen induced.

D. Incorrect. Plausible because any increase in RCS fission product isotopes during steady-state operation may indicate fuel damage, however, Mn-54 and Mn-56 are the result of a CRUD burst.

Technical Reference(s) ABN-102, Steps 1, 6 & 7 NOTES Attached w/ Revision # See Comments / Reference
SOP-103A, Step 5.5.15.A CAUTION

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to High Reactor Coolant Activity per ABN-102, High Reactor Coolant Activity.

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, 12
55.43 ______

Comments / Reference: From SOP-103A, Step 5.5.15.A CAUTION	Revision # 17

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-103A
CHEMICAL AND VOLUME CONTROL SYSTEM	REVISION NO. 17	PAGE 88 OF 131

5.5.15 Shifting Charging Pump Suction Between the VCT and RWST

CAUTION: Charging pump suction should normally remain aligned to the VCT due to dissolved oxygen concerns when suction comes from the RWST. When entering a plant outage, suctions should NOT be rolled to the RWST prior to crud burst. When time allows, Chemistry should be notified prior to rolling suction to the RWST.

A. IF desired to shift charging pump suction from the VCT to the RWST, THEN perform the following:

1) OPEN ONE or BOTH of the following valves:

☐ • 1/1-LCV-112D, RWST TO CHRG PMP SUCT VLV

☐ • 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV

Comments / Reference: From ABN-102, Step 7 NOTE	Revision # 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-102
HIGH REACTOR COOLANT ACTIVITY	REVISION NO. 7	PAGE 5 OF 6

NOTE:
- An increase of RCS activated corrosion products may indicate a "CRUD" burst. (e.g., Fe-59, Co-58, Co-60, Mn-54, Mn-56, Cr-51, and Zr-95).
- The stepping or tripping of control or shutdown rods should be kept to a minimum when reactor coolant CRUD levels are high to reduce the potential for CRDM mis-stepping due to CRUD contamination of CRDM latch assemblies (CR 2009-008942).

☐ 7. IF RCS activity increase is believed to be result of RCS transient OR "CRUD" burst, THEN refer to Technical Specification 3.4.16.

Comments / Reference: From ABN-102, Step 1 & 6 NOTES	Revision # 7

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-102
HIGH REACTOR COOLANT ACTIVITY	REVISION NO. 7	PAGE 4 OF 6

2.3 Operator Actions

NOTE:
- Reactor Coolant System transients such as power changes, temperature changes, pressure changes, and starting and stopping RCPs can cause temporary increases in RCS activity.
- Monitor spiking and return to normal is not a real indication of failed fuel and as such does not require sampling. A steady or sustained increase over time would be a real indication of failed fuel/RCS activity problems.

☐ 1. Request additional reactor coolant specific activity samples be taken in accordance with CHM-111 for isotopic content analysis per Technical Specification 3.4.16, SURVEILLANCE REQUIREMENTS.

☐ 2. Notify Chemistry to review chemistry data and Core Performance Engineering to review chemistry data and core follow trends. Chemistry will determine if a "CRUD" burst has occurred. Core Performance Engineering will determine if the source of RCS activity is failed fuel and the extent of failed fuel, if any.

☐ 3. Increase letdown flow to 120-140 gpm as follows:

a) IF PDP is in operation, THEN start up a centrifugal charging pump AND shutdown PDP per SOP-103A/B.

b) Increase letdown flow to 120-140 gpm per SOP-103A/B.

☐ 4. Notify Radiation Protection that radiation levels may increase in Auxiliary and Safeguards Buildings AND on any ARMs.

☐ 5. Make a plant announcement via Gai-Tronics of indication of an increase in RCS Activity AND a possibility of increased radiation in Auxiliary and Safeguards Buildings.

NOTE: A rapid increase of RCS fission product isotopes during steady state operation may indicate fuel cladding damage. (e.g., Xe-133, Kr-85M, Cs-137, Cs-136, Sr-84, Sr-90, Iodine).

☐ 6. IF Core Performance Engineering Review of the chemistry data indicates failed fuel, THEN proceed as follows:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E08 EK3.1	
	Importance Rating	3.4	

RCS Overcooling - PTS: Knowledge of the reasons for the following responses as they apply to the Pressurized Thermal Shock: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics

Proposed Question: Common 63

What is the basis for terminating Safety Injection (SI) flow during the performance of FRP-0.1A, Response to Imminent Pressurized Thermal Shock Conditions?

A. SI flow must be terminated to prevent entry into a condition to the right of Limit A.

B. If SI can be terminated, the event causing entry into the procedure has been corrected.

C. SI flow must be terminated to maintain RWST inventory in the event a LOCA outside Containment.

D. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because to the right of Limit A is a branch on the Integrity Safety Function Status Tree, however, this is a desirable position during a PTS event.
B. Incorrect. Plausible because Safety Injection flow can contribute to a PTS condition due to cooldown, however, this is not the only condition that allows exit from FRP-0.1A.
C. Incorrect. Plausible because FRP-0.1A may be entered after a LOCA if plant cooldown is sufficient to drive RCS temperature low enough, however, FRP-0.1A actions take precedent over LOCA procedures and therefore SI flow is terminated to reduce the probability of a Pressurized Thermal Shock event.
D. Correct. Following SI actuation, RCS conditions may be restored to within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this procedure than those present in the ERGs since, for an imminent PTS condition, ECCS flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

Technical Reference(s) FRP-0.1A, Attachment 4, Step 7 Bases; FRP-0.1A, CSFST — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedural Step, NOTE, or CAUTION, **DISCUSS** the reason or basis for the Step, NOTE, or CAUTION in FRP-0.1, Response to Imminent Pressurized Thermal Shock Conditions.

Question Source: Bank # X
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 FRP-0.1A, Attachment 4, Step 7 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRP-0.1A
RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION	REVISION NO. 8	PAGE 33 OF 53

ATTACHMENT 4
PAGE 3 OF 23

BASES

This action contains the action verb "control" which implies continuous performance; therefore, this step has been identified as a continuous action step.

STEP 3: When CST Level decreases below 10%, an adequate water supply to AFW pumps is no longer available. An alternate suction source should be provided.

This action could apply anytime during response and recovery actions; therefore, the step has been identified as a continuous action step.

STEP 4: To ensure operability of the block valves, it should be verified that power is available to them. In order for the PRZR PORV to perform its function of relieving RCS overpressure conditions, the associated block valve must be open.

CAUTION: Normally, the PRZR PORV is checked to be closed after RCS pressure decreases below its setpoint. Because pressure transients may occur, and because of the range over which this procedure is required to be effective, this CAUTION alerts the operator to check closure. No specific setpoint is included, since the following step extends over the range of applicability of the Low Temperature Overpressure Protection System.

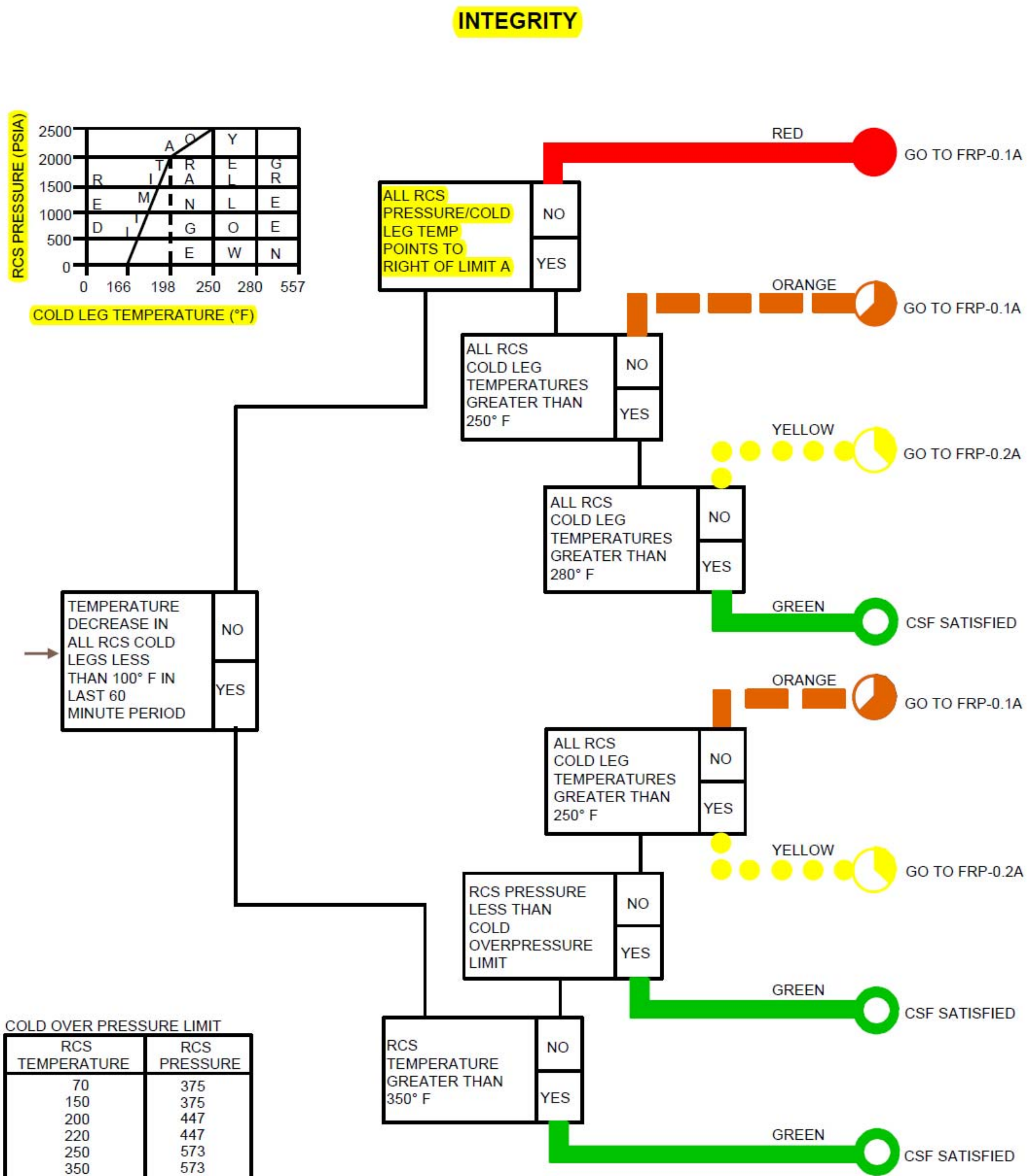
STEP 5: Depending upon the implementation of the Low Temperature Overpressure Protection System (LTOPS), the pressure criterion used for checking PORV operations is either PRZR pressure less than the PORV setpoint (if LTOPS not in service) or RCS pressure less than cold overpressure limit (if LTOPS in service). If the appropriate pressure criterion is met, the PRZR PORVs should be closed.

Based on the preceding caution, this action is applicable during recovery actions; therefore, the step has been designated as a continuous action step.

STEP 6: If ECCS is in service, then the SI termination sequence in Steps 7 through 16, which includes stopping ECCS pumps, establishing charging and isolating the CCP injection line is appropriate. If ECCS is not in service, these steps are bypassed.

STEP 7: Following SI actuation, RCS conditions may be restored to within acceptable limits for SI termination to be allowed. The combination of a minimum subcooling and sufficient liquid level in the vessel to cover the core represents less restrictive SI termination criteria in this procedure than those present in the ERGs since, for an imminent PTS condition, ECCS flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure.

Comments / Reference: From FRP-0.1A, CSFST	Revision # 8



COLD OVER PRESSURE LIMIT

RCS TEMPERATURE	RCS PRESSURE
70	375
150	375
200	447
220	447
250	573
350	573

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	W/E10 EK3.3	
Importance Rating	3.4	

Natural Circulation with Steam Void in Vessel with/without RVLIS: Knowledge of the reasons for the following responses as they apply to the Natural Circulation with Steam Void in Vessel with/without RVLIS: Manipulation of controls required to obtain desired operating results during abnormal and emergency situations

Proposed Question: Common 64

During the performance of the EOS-0.3B, Natural Circulation with Steam Void in Vessel (with RVLIS), when starting the first Reactor Coolant Pump with a void in the upper head, Pressurizer level is maintained…

A. ...below 34% to compensate for the level increase due to the void formation.

B. ...below 34% to compensate for the level increase due to the non-condensable gases being swept into the upper head.

C. ...above 90% to compensate for the level decrease due to the void collapsing.

D. ...above 90% to compensate for the level decrease due to the non-condensable gases being swept from the upper head.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because after the 1st Reactor Coolant Pump is started EOS-0.3B contains guidance to maintain Pressurizer level between 30% and 40%, however, level will decrease due to void collapse.

B. Incorrect. Plausible because after the 1st Reactor Coolant Pump is started EOS-0.3B contains guidance to maintain Pressurizer level between 30% and 40%, however, any non-condensable gases present would have to be reabsorbed by RCS fluid unless the Reactor Vessel Head were vented.

C. Correct. Pressurizer level is maintained above 90% to prevent void collapse when starting the 1st Reactor Coolant Pump.

D. Incorrect. Plausible because the Pressurizer level value is correct, however, void collapse is the reason for maintaining this level. Non-condensable gases will be present but they are not swept from the head region.

Technical Reference(s): EOS-0.3B, Step 1.b RNO & 2; EOS-0.3B, Attachment 4, Step 1 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the recovery technique used and the procedure steps of EOS-0.3, Natural Circulation Cooldown with Steam Void in Vessel (with RVLIS).

Question Source: Bank # X
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 EOS-0.3B, Step 1.b RNO & 2	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.3B
NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)	REVISION NO. 8	PAGE 4 OF 30

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	b. Check RVLIS indication - 49 IN ABOVE FLANGE LIGHT LIT	b. Perform the following: 1) Increase PRZR level to 90% using charging and letdown. 2) Establish subcooling greater than 60°F using steam dumps or SG atmospherics. 3) Use PRZR heaters, as necessary to saturate the pressurizer water.
	c. Start one RCP per Attachment 2.	c. Go to Step 2. OBSERVE NOTE PRIOR TO STEP 2.
	d. Go to IPO-007B, MAINTAINING HOT STANDBY.	

NOTE: Saturated conditions in the PRZR should be established before trying to decrease PRZR level.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Establish PRZR Level To Accommodate Void Growth:	
	a. Check PRZR level - BETWEEN 30% AND 40%	a. Control charging and letdown as necessary.

Comments / Reference: From EOS-0.3B, Attachment 4, Step 1 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.3B
NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)	REVISION NO. 8	PAGE 21 OF 30

ATTACHMENT 4
PAGE 4 OF 13

BASES

Pressurizer level and subcooling requirements for starting an RCP with a void in the upper head are designed to accommodate a collapse of the void. Starting an RCP will preclude the use of a pressurizer PORV during subsequent recovery, however, the operator should anticipate a decrease in pressurizer level and RCS subcooling when the RCP is started with upper head voiding. Charging flow should be increased as necessary to maintain pressurizer level on span and adequate RCS subcooling. It may also be necessary to isolate letdown in order to maintain pressurizer level. If pressurizer level or RCS subcooling is lost, SI actuation will be required per the foldout page.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E15 EA1.1	
	Importance Rating	2.9	

Containment Flooding: Ability to operate and/or monitor the following as they apply to Containment Flooding: Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features

Proposed Question: Common 65

Given the following conditions:

- Unit 1 has experienced a Large Break Loss of Coolant Accident.
- While responding in EOP-1.0A, Loss of Reactor or Secondary Coolant, the Control Room observed Containment Sump level greater than 816'.
- The Control Room is attempting to identify water volumes other than the Refueling Water Storage Tank or Safety Injection Accumulators that may be the source of the additional water in Containment.

In addition to Fire Protection Water, Auxiliary Feedwater, and Ventilation Chilled Water, which of the following systems are likely sources of the additional water in Containment per FRZ-0.2A, Response to Containment Flooding?

1. Main Feedwater
2. Safety Chilled Water
3. Component Cooling Water
4. Reactor Makeup Water
5. Vents and Drains (Floor Drain Tank 1-01)

A. 2, 3, 5

B. 1, 3, 4

C. 1, 4, 5

D. 1, 2, 3

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the sources listed are correct with the exception of Safety Chilled Water and Floor Drain Tank.
B. Correct. Safety Chilled Water and the Floor Drain Tank are the 2 exceptions to sources of water during a Large Break LOCA.
C. Incorrect. Plausible because the sources listed are correct with the exception of the Floor Drain Tank.
D. Incorrect. Plausible because the sources listed are correct with the exception of Safety Chilled Water.

Technical Reference(s) FRZ-0.2A, Step 1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the symptoms, or Entry Conditions for FRZ-0.2, Response to Containment Flooding.

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 9, 10
55.43

Comments / Reference: From FRZ-0.2A, Step 1	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRZ-0.2A
RESPONSE TO CONTAINMENT FLOODING	REVISION NO. 8	PAGE 3 OF 9

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Check The Following Systems For Indication Of Possible Source Of Water To The Containment Sump (i.e., Pressure, Surge Tank Level, Flow, etc.): • RMUW • Demineralized water • CCW • Chemical and Volume Control System • Main Feedwater • AFW • Ventilation Chilled Water • Fire protection water	

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #		
K/A #	G 2.1.2	
Importance Rating	4.1	

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation

Proposed Question: Common 66

Given the following conditions:

- A non-licensed operator successfully completed the Generic Fundamentals Examination, and has returned to Shift until License Training on Plant Systems starts next month.
- Unit 2 is at End-Of-Life.
- BTRS Demineralizers will be placed in service during the Shift to dilute the Reactor Coolant System.
- The trainee requests to perform the dilution for training. The Shift Manager agrees, provided the trainee participates in the Briefing.

Should the trainee be allowed to perform this evolution?

A. Yes.
The trainee has been enrolled in the Replacement License Program per TRA-203.

B. No.
The use of a trainee for this evolution has not been approved by the Shift Operations Manager.

C. Yes.
The Shift Manager has granted permission for the trainee to participate.

D. No.
The trainee has not completed necessary classroom instruction for the evolution.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because this evolution is permissible only if a person is enrolled in the Replacement License Program and has successfully completed classroom instruction. This individual has not completed classroom instruction.
B. Incorrect. Plausible because this individual is not allowed to perform this evolution, however, approval by the Shift Operations Manager is only required for Reactor Startups.
C. Incorrect. Plausible because the Shift Manager can authorize evolutions that affect reactivity, however, appropriate classroom training has not been completed.
D. Correct. This evolution is permissible only if a person is enrolled in the Replacement License Program and has successfully completed classroom instruction

Technical Reference(s) ODA-102, Step 6.24 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source: Bank # X
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 ODA-102, Step 6.24	Revision # 26

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 26 INFORMATION USE	PAGE 30 OF 37

6.24 Conduct of On-The-Job Training

- The SM shall approve the use of a trainee in any evolution which may affect reactivity, plant safety or a control system and ensure the trainee is included in required briefing per ODA-407.

[C]
- The Shift Operations Manager shall approve the use of a trainee to conduct a reactor startup as the RO. [23344]

- Manipulation of controls by a non-licensed person (Trainee), is permissible only if that person is currently enrolled in the replacement license program as described in TRA-203, has successfully completed classroom instructions for the given evolution, is directly supervised by a Licensed Operator, and approval is granted by the SM.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.1.36	
	Importance Rating	3.0	

Conduct of Operations: Knowledge of procedures and limitations involved in core alterations

Proposed Question: Common 67

Given the following conditions:

- Unit 2 is in MODE 6.
- N-31 and N-32 are the OPERABLE Source Range Nuclear Instruments.
- STA-617, High Voltage Switching and Clearance, is about to be performed in the Switchyard.

Which of the following must be performed prior to implementing STA-617, High Voltage Switching and Clearance?

A. Place the High Flux at Shutdown Switch in BLOCK on both N-31 and N-32 to prevent loss of the Source Range Nuclear Instrumentation.

B. Place the High Flux at Shutdown Switch in BLOCK on both N-31 and N-32 to prevent a Containment evacuation.

C. Suspend CORE ALTERATIONS and positive reactivity additions due to the potential for spiking of the Source Range Nuclear Instrumentation.

D. Suspend CORE ALTERATIONS and positive reactivity additions due to the potential for loss of power to Refueling equipment.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because spiking of the Source Range Nuclear Instrumentation will occur, however, placing the High Flux at Shutdown Switch in BLOCK would violate Technical Specifications.
B. Incorrect. Plausible because spiking could activate the Containment Evacuation Alarm, however, the switch would not be placed in BLOCK.
C. Correct. Per the Precaution outlined in RFO-102.
D. Incorrect. Plausible because CORE ALTERATIONS would be suspended, however, not for the reasons listed.

Technical Reference(s) RFO-102, Steps 3.13 & 3.17 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** indication and Control/Trips for Source Range High Flux at Shutdown and Containment Evacuation Alarms.

Question Source: Bank # X
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 6, 10
55.43 ____

Comments / Reference: From RFO-102, Step 3.13	Revision # 13

CPNPP STATION REFUELING MANUAL	UNIT COMMON	PROCEDURE NO. RFO-102
REFUELING OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 7 OF 96

3.9 If the High Flux at Shutdown alarm is actuated, all personnel should evacuate Containment in an orderly manner. If the alarm is actuated during movement of fuel, fuel should be placed in a safe condition before evacuating Containment. The Fuel Handling Supervisor should designate personnel for Containment re-entry, when conditions allow, to investigate the cause of the alarm:

- The Fuel Handling Supervisor has the authority to determine if the High Flux at Shutdown alarm is spurious and whether to continue with evacuation.
- The Fuel Handling Supervisor or an operator designated by the Shift Manager will ensure that Containment is evacuated in a safe and orderly manner.

3.10 In case of malfunction or suspicion of malfunction of any fuel handling equipment, operation of the affected equipment will be terminated and the Fuel Handling Supervisor shall be notified immediately for resolution.

[C]
3.11 The transfer tube gate valve shall not be opened until it has been verified that the boron concentration of the water, if any, in the Fuel Building transfer canal is greater than or equal to the boron concentration of the water in the Refueling Cavity OR boron concentration $\geq$2400 ppm in both areas. (Admin Limit) (TS 3.9.1 limit is specified in the COLR) [4408996]

[C]
3.12 The 4-inch Refueling Cavity drain valves should be closed and the drain strainer removed from the refueling cavity (4-inch drain strainer CP1/CP2-SFSRDS-02 located in the Lower Internals Area). This will prevent contamination of the drain strainer which must be removed to enable cavity drain after refueling. The debris screens over the 6-inch drains (CP1/CP2-SFSRDS-03 and -04) should be rotated and the blind flanges installed over the two 6-inch drain lines from the lower internals storage stand prior to opening the transfer tube gate valve. This will minimize the effects of a transfer canal or SFP gate seal failure.

3.13 Prior to initiation of any switching per STA-617 AND Westinghouse Source Range detectors are the operable SR instruments, all CORE ALTERATIONS and positive reactivity additions should be suspended due to potential spiking on the Source Range nuclear instrumentation (N-31, N-32) and possibility of receiving the Source Range High Flux at Shutdown alarm, if N-31 and N-32 are the operable source range channels.

Comments / Reference: From RFO-102, Step 3.17	Revision # 13

CPNPP STATION REFUELING MANUAL	UNIT COMMON	PROCEDURE NO. RFO-102
REFUELING OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 8 OF 96

3.15 Significant level differences between the Refueling Cavity and Spent Fuel Pool can occur due to atmosphere pressure differences. To avoid Spent Fuel Pool overflow, consideration of this effect should be given to any evolution which would change the pressure differential (manipulation of the Fuel Building or Containment Ventilation System, use of compressed gases in Containment, air displacement due to filling the cavity, temperature changes, etc.).

3.16 When possible, an additional DG or an additional offsite power source should be maintained available to augment the Technical Specification required AC sources during refueling operations.

3.17 Source range nuclear instruments shall be monitored by a Control Room operator at all times when CORE ALTERATIONS are being performed.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.2.18	
	Importance Rating	2.6	

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

Proposed Question: Common 68

Given the following conditions with Unit 1 in Reduced Inventory:

- Spent fuel is still in the Reactor Vessel.
- IPO-010A, Reactor Coolant System Reduced Inventory Operations, is in progress and Attachment 1, Shiftly Checklist, was completed by the previous shift.
- The oncoming crew has been on watch for 30 minutes.
- Maintenance wants to open the Instrument Air to Containment penetration for repairs.
- No other penetrations are opened for maintenance.

Which of the following describes the appropriate response to the request to open the Instrument Air to Containment penetration?

A. The penetration may NOT be opened until Reactor Vessel level raised above 80".

B. The penetration may be opened for maintenance with NO administrative controls.

C. The penetration may be opened for maintenance if tracked per LCO 3.9.4, Containment Penetrations.

D. The penetration may NOT be opened until a new IPO-010A, Attachment 1, Shiftly Checklist is completed.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that no penetrations may be opened in Reduced Inventory conditions.
B. Incorrect. Plausible if thought that since no other penetrations are impaired one does not have any administrative controls.
C. Correct. Up to ten penetrations may be impaired per IPO-010A, Attachment 1 if tracked and can be sealed if needed.
D. Incorrect. Plausible if thought that the Shiftly Checklist from the previous shift is not adequate.

Technical Reference(s) IPO-010A, Attachment 1, 2.0.C Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the Precautions, Limitations and key Attachments of WCI-401, Outreach Safety Function Guide.

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 IPO-010A, Attachment 1, 2.0.C	Revision # 18

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-010A
REACTOR COOLANT SYSTEM REDUCED INVENTORY OPERATIONS	REVISION NO. 18 CONTINUOUS USE	PAGE 127 OF 195

ATTACHMENT 1
PAGE 4 OF 12

[C] SHIFTLY CHECKLIST

C. IF work is in progress on any Containment penetration, THEN PERFORM the following steps:

1) REVIEW penetrations being tracked per LCO 3.9.4 and RECORD or VERIFY the required information is entered on the log.

_____ / _____
Initials Date

[C] 2) REVIEW the work package and VERIFY that any required means for quickly sealing the penetration are available at the work site (i.e. temporary flanges). Temporary seals must be capable of being installed within the time limits of Attachment 7.

_____ / _____
Initials Date

CAUTION: DO NOT enter reduced inventory operations with fuel in the Reactor Vessel if more than ten penetrations are impaired and not sealed.

3) VERIFY that ten or fewer penetrations are impaired.

_____ / _____
Initials Date

Examination Outline Cross-reference:

	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.2.38	
	Importance Rating	3.6	

Equipment Control: Knowledge of conditions and limitations in the facility license

Proposed Question: Common 69

Given the following conditions:

- Unit 2 is in MODE 5.
- RCS temperature is currently 110ºF.
- Maintenance reports that Component Cooling Water (CCW) Pump 2-02 has a failed motor bearing.

Under these conditions, which of the following is the HIGHEST Reactor Coolant System temperature that Unit 2 can be increased to WITHOUT violating Technical Specifications?

A. 135ºF

B. 195ºF

C. 315ºF

D. 345ºF

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because Refueling outages are typically considered to be conducted after the RCS temperature is reduced below 140ºF, but both CCW Trains are only required to be OPERABLE in MODES 1-4.

B. Correct. Both CCW Trains are required to be OPERABLE prior to entry into MODE 4. The RCS temperature defining MODE 4 operations is 200ºF, so this is the highest temperature that can be achieved. There are no provisions of Technical Specification LCO 3.0.4 that would allow entry into MODE 4.

C. Incorrect. Plausible because a temperature of 320ºF appears throughout Tech Specs, primarily associated with LTOP, but both trains of CCW are required to be OPERABLE in MODES 1-4.

D. Incorrect. Plausible because a temperature of 345ºF would prevent entering MODE 3 from MODE 4, however, both trains of CCW are required to be OPERABLE in MODES 1-4.

Technical Reference(s) Technical Specification LCO 3.7.7 Attached w/ Revision # See Comments / Reference
Technical Specification Definitions 1.1

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Component Cooling Water System including Technical Specifications, TRM and ODCM.

Question Source: Bank # X
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 Technical Specification LCO 3.7.7	Amendment # 156

CCW System
3.7.7

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME

Comments / Reference: From Technical Specification Definitions 1.1	Amendment # 150

Definitions
1.1

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.
(b) All reactor vessel head closure bolts fully tensioned.
(c) One or more reactor vessel head closure bolts less than fully tensioned.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.3.4	
	Importance Rating	3.2	

Radiation Control: Knowledge of radiation exposure limits under normal or emergency condition

Proposed Question: Common 70

Given the following condition:

- A 20 year old CPNPP employee has 1000 mrem of TEDE exposure for 2013.

Which of the following describes the MAXIMUM additional amount of TEDE exposure that may be received without additional CPNPP authorization (i.e., Plant Manager approval, Radiation and Industrial Safety Manager, employee supervisor, etc.), and what is the MAXIMUM additional amount of TEDE exposure that may be received prior to exceeding 10CFR20 (NRC) exposure limits?

A. 1000 mrem; 4000 mrem

B. 3000 mrem; 4000 mrem

C. 1000 mrem; 9000 mrem

D. 3000 mrem; 9000 mrem

Proposed Answer: A

Explanation:

A. Correct. Admin limit is 2000 mrem per year; 10CFR20 limit is 5000 mrem per year.
B. Incorrect. Plausible because the Admin limit previously was 4000 mrem and a Planned Special Exposure is also 4000 mrem; 10CFR20 limit is 5000 mrem per year.
C. Incorrect. Plausible because the Admin limit is 2000 mrem, however, the 2nd part is only correct if thought that the 10CFR20 5(N-18) rule was still applicable.
D. Incorrect. Plausible because the Admin limit previously was 4000 mrem and a Planned Special Exposure is also 4000 mrem, however, the 2nd part is only correct if thought that the 10CFR20 5(N-18) rule was still applicable.

Technical Reference(s) STA-655, Attachment 8.A Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 12
55.43 ______

Comments / Reference: From STA-655, Attachment 8.A	Revision # 20

CPNPP STATION ADMINISTRATION		PROCEDURE NO. STA-655
EXPOSURE MONITORING PROGRAM	REVISION NO. 20 INFORMATION USE	Page 22 of 28

ATTACHMENT 8.A
PAGE 1 OF 2

ADMINISTRATIVE EXPOSURE LEVELS
DEEP DOSE

RADIATION WORKERS

PERIOD	CALCULATION	LEVEL
Annual	TEDE (Total Effective Dose Equivalent)	2000 mrem
Annual	TODE - (The SUM of Deep-Dose Equivalent and Committed Dose Equivalent to any individual organ or tissue other than the lens of the eye).	20,000 mrem

PERIOD	EVENT	LEVEL
Annual	Planned Special Exposure (PSE) **NOT TO EXCEED:**	4000 mrem
Lifetime	Planned Special Exposure (PSE)	Five times the annual dose limit.

Comments / Reference: From STA-655, Attachment 8.A	Revision # 20

CPNPP STATION ADMINISTRATION		PROCEDURE NO. STA-655
EXPOSURE MONITORING PROGRAM	REVISION NO. 20 INFORMATION USE	Page 24 of 28

ATTACHMENT 8.B
PAGE 1 OF 2

NRC EXPOSURE LIMITS

RADIATION WORKERS

PERIOD	CALCULATION	LEVEL
Annual	TEDE (Total Effective Dose Equivalent)	5000 mrem
	OR	
Annual	TODE - (The SUM of Deep-Dose Equivalent and Committed Dose Equivalent to any individual organ or tissue other than the lens of the eye).	50,000 mrem

Comments / Reference: From CPNPP Exam Bank	Revision # 03/24//11

Given the following condition:

- A 20 year old CPNPP employee has 500 mrem of TEDE exposure for 2010.

Which of the following describes the MAXIMUM additional amount of TEDE exposure that may be received without additional CPNPP authorization (i.e., Plant Manager approval, Radiation and Industrial Safety Manager, employee supervisor, etc.), and what is the MAXIMUM additional amount of TEDE exposure that may be received prior to exceeding 10CFR20 (NRC) exposure limits?

A. <u>1500 mrem; 4500 mrem</u>
B. 3500 mrem; 4500 mrem
C. 1500 mrem; 9500 mrem
D. 3500 mrem; 9500 mrem

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #		
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 71

Given the following:

- Radiography is in progress in the Unit 2 Safeguards Building.
- 2-RE-6297, Emergency Airlock Area Radiation Monitor (ARM) is reading 10 mrem/hr above background when the source is EXPOSED.
- The exposed source is located 20 feet from the monitor.

The source is then moved closer to the ARM and exposed from 10 feet away.

Which of the following describes the current 2-RE-6297, Emergency Airlock Area Radiation Monitor reading?

A. 20 mrem/hr above background.

B. 40 mrem/hr above background.

C. 200 mrem/hr above background.

D. 400 mrem/hr above background.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that $I_1 \times D_1 = I_2 \times D_2 = 10_1 \times 20_1 = I_2 \times 10_2 = 20$ mrem/hr

B. Correct. $I_1 \times D_1^2 = I_2 \times D_2^2$ therefore, $10_1 \times 20_1^2 = I_2 \times 10_2^2 = 40$ mrem/hr

C. Incorrect. Plausible if math error is made.

D. Incorrect. Plausible if thought that $I_1^2 \times D_1^2 = I_2^2 \times D_2^2 = 10_1^2 \times 20_1^2 = I_2^2 \times 10_2^2 = 400$ mrem/hr

Technical Reference(s) GFE.RR4.LN, Page 31 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the inverse square law.
CALCULATE the dose rate at varying distances from point sources, line sources, plane sources, and tank sources.

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

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 __12__
55.43 ____

Comments / Reference: From GFE.RR4.LN, Page 31	Revision # 0

The simplest type of radiation source emanates from a single point in space. The radiation intensity decreases with the square of the distance. The phenomena is called the inverse square law, and is defined mathematically by:

$$I_1 \text{ x } (D_1)^2 = I_2 \text{ x } (D_2)^2$$

Comments / Reference: From CPNPP Exam Bank	Revision # 04/10/12

Given the following:

- Radiography is in progress in the Unit 2 Safeguards Building.
- 2-RE-6297, Emergency Airlock Area Radiation Monitor (ARM) is reading 5 mrem/hr above background when the source is EXPOSED.
- The exposed source is located 40 feet from the monitor.

The source is then moved closer to the ARM and exposed from 10 feet away.
Which of the following describes the current 2-RE-6297, Emergency Airlock Area Radiation Monitor reading?

A. 20 mrem/hr above background.
B. 40 mrem/hr above background.
C. 60 mrem/hr above background.
D. <u>80 mrem/hr above background.</u>

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Category #		
K/A #	G 2.3.13	
Importance Rating	3.4	

Radiation Control: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: Common 72

Given the following condition with a Refueling in progress:

- Maintenance Services has requested entry into the Incore Instrumentation Room, Elev. 832' to clean up debris around the Seal Table.

Which of the following identifies the condition that must be met prior to allowing access per STA-620, Containment Entry?

The Incore Detectors...

A. …must NOT be inserted in the core.

B. …must be continuously monitored at the Seal Table.

C. …should be stored and tagged out-of-service.

D. …Drive System must NOT be disconnected.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that storing the Incore Detectors in this location provides a hazard to individuals in the room, however, this is a desirable location.
B. Incorrect. Plausible because the Seal Table is one of the locations where movement of the Incore Detectors could take place, however, the system is tagged out to prevent inadvertent movement.
C. Correct. Per STA-620 the Incore Detectors System should be stored and tagged out of service to prevent possible movement and radiation overexposure of personnel.
D. Incorrect. Plausible if thought that disconnecting the drive system could allow inadvertent movement of the Incore Detectors, however, it is the tagout that ultimately protects personnel.

Technical Reference(s) STA-620, Step 6.1.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **LIST** the prerequisites that must be met prior to a Containment Entry.

Question Source: Bank # X
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 STA-620, Step 6.1.2	Revision # 13

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-620
CONTAINMENT ENTRY	REVISION NO. 13	PAGE 15 OF 37
	INFORMATION USE	

6.1.2 During refueling outages and maintenance activities, or when Containment is otherwise occupied for extended periods, the incore detectors should be tagged out of service until work activities are completed and/or arrangements made to preclude entry to the following areas:

- 808'-Incore Instrumentation Room
- 808'-Excess Letdown Heat Exchanger Room
- 808'-Steam Generator Loop Rooms
- 832'-Incore Instrumentation Room
- 832'-Regenerative Heat Exchanger Room
- 849'-Incore Instrumentation Room

IF either of the following is true, THEN Caution Tags may be lifted by the Shift Manager:

- The detectors have been placed in storage and/or are incapable of being withdrawn or moved during performance of maintenance and testing.
- It has been determined by Radiation Protection that operation of the incore detectors will not adversely affect other activities in Containment.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.4.11	
	Importance Rating	4.0	

Emergency Procedures / Plan: Knowledge of abnormal condition procedures

Proposed Question: Common 73

Given the following conditions:

- Unit 1 is at 100% power.
- The following Annunciators have just alarmed:
  - 1-ALB-7B, Window 4.13 – FWPT A/B DIGITAL CNTRL TRBL.
  - 1-ALB-10B, Window 3.19 – 118 VAC INV IV1C5/1C6 TRBL.
- Instrument Bus 1C5 has been lost.

Which of the following describes the actions and reason per ABN-604, Loss of a Non-1E Instrument Bus?

A. Place Rod Control in MANUAL and manually control Seal Injection, Letdown and Charging flows.
Multiple primary instrument and control failures have occurred.

B. Swap to Alternate Power, verify instrument indications and refer to Technical Specifications for REQUIRED ACTIONS.
Loss of safety related indications with no automatic actions has occurred.

C. Direct a Turbine Runback and trip the Reactor if Steam Generator levels decrease uncontrollably.
Loss of forward flow from the Heater Drains System has occurred.

D. Verify all Plant Computer CRT Screens performing normally and BYPASS Non-Safeguards Inverter IV1C5.
Plant Computer FAILOVER to backup mode has occurred.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because these are some of the symptoms and action for a loss of Protection Bus 1PC1.

B. Incorrect. Plausible because these are some of the symptoms and action for a loss of Protection Bus 1EC1.

C. Incorrect. Plausible because procedure entry, actions, and the reason are all correct, however, these apply to a loss of Inverter IV1C3.

D. Correct. These are the correct actions and reason for a loss of Inverter IV1C5.

Technical Reference(s): ABN-604, Sections 4.1 & 4.2; ABN-604, Steps 4.3.1 & 4.3.2; ABN-604, Section 3.1 & 3.2; ABN-604, Steps 3.3.1 & 3.3.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Loss of Instrument Bus C5 per ABN-604, Loss of Non-1E Instrument Bus.

Question Source:
- Bank # X
- 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 ABN-604, Sections 4.1 & 4.2	Revision # 4

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-604
LOSS OF NON-1E INSTRUMENT BUS	REVISION NO. 4	PAGE 8 OF 55

4.0 LOSS OF INSTRUMENT BUS uC5 OR uC6

4.1 Symptoms

a. Annunciator Alarms

- "FWPT A/B DIGITAL CNTRL TRBL (7B-4.13)
- "AMSAC TRBL" (9B-4.7)
- "118V INV IVuC5/IVuC6 TRBL" (10B-3.19)

b. Plant Indications

- The words TIME NOT UPDATING will appear in time and date area on Plant Computer CRT screen.
- The word FAILOVER will appear in active section of Plant Computer CRT screen until backup takes over.

4.2 Automatic Actions

Plant Computer fails over to BACKUP mode.

NOTE:
- Inverters IVuC5/6 supply power to the Mark V FWP controllers.
- Inverter IVuC6 supplies power to Condensate Polishing controls. Loss of IVuC6 will cause u-PV-2242, Uu CNDS POL FILT BYP PRESS CTRL VLV to fail open.

Comments / Reference: From ABN-604, Steps 4.3.1 & 4.3.2	Revision # 4

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-604
LOSS OF NON-1E INSTRUMENT BUS	REVISION NO. 4	PAGE 9 OF 55

4.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: During time Plant Computer is not operating, increased operator vigilance is required due to loss of many important monitoring functions performed by computer (e.g., alarms, thermal power calculations).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify ALL Plant Computer CRT screens performing normally.	GO TO ABN-906, while other operators continue this procedure.
2 Locally perform the following at IVuC5 OR IVuC6:	Energize uC5 or uC6 per SOP-607A/B, if desired.
☐ a. Depress BYPASS SOURCE TO LOAD pushbutton for affected inverter.	
☐ b. Verify green status light is illuminated on affected inverter BYPASS SWITCH panel.	
☐ c. Place affected inverter MAINTENANCE SWITCH in BYPASS position.	

Comments / Reference: From ABN-604, Sections 3.1 & 3.2	Revision # 4

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-604
LOSS OF NON-1E INSTRUMENT BUS	REVISION NO. 4	PAGE 5 OF 55

3.0 LOSS OF INSTRUMENT BUS uC3

3.1 Symptoms

- "118 VAC INV IVuC3 TRBL" (10B-4.14)
- The associated bus instruments alarming or failing.

3.2 Automatic Actions

- Steam Generator Blowdown isolates by closing valves u-HV-2397, u-HV-2398, u-HV-2399, u-HV-2400, u-HV-2397A, u-HV-2398A, u-HV-2399A, u-HV-2400A
- Turbine Trip Test channel 2 is disabled.
- C-9 interlock disabled (steam dumps are unavailable).
- Reactor Coolant make-up control will not function in
  -MANUAL
  -BORATE
  -DILUTE
  -ALT DILUTE
- Extraction Steam to FW Heater valves will not close on Turbine Trip.
- May cause loss of forward flow from Heater Drains. ABN-302 provides guidance on loss of extraction steam.

Comments / Reference: From ABN-604, ABN-604, Steps 3.3.1 & 3.3.2	Revision # 4

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-604
LOSS OF NON-1E INSTRUMENT BUS	REVISION NO. 4	PAGE 6 OF 55

3.3 Operator Actions

CAUTION:
- Reactor power must be established at a value within the capability of available feed water. Auxiliary feedwater pumps can supply approximately 6% reactor power.
- The status of the secondary heat sink and available feedwater must be closely monitored during the performance of this procedure. The Reactor should be manually tripped if secondary heat sink cannot be maintained.

NOTE:
- Diamond step 1 denotes Initial Operator Action.
- Should a Reactor Trip occur at any time during performance of this procedure, immediately proceed to EOP-0.0A/B, Reactor Trip or Safety Injection.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ ◇1 Ensure Turbine Power - LESS THAN OR EQUAL TO 800 MW. • Ensure 1/u-RBSS, CONTROL ROD BANK SELECT in AUTO. • Manually runback Turbine load to 700 MW.	
☐ 2 Verify SG Levels - STABLE OR TRENDING TO NORMAL OPERATING RANGE.	Perform the following: a. IF SG level is decreasing in an uncontrolled manner, THEN trip the Reactor AND GO TO EOP-0.0A/B while continuing this procedure. b. IF Reactor Power is above the capability of available feed flow, THEN reduce power using a combination of rod control, turbine control or boration until steam generator levels can be maintained while continuing this procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.4.31	
	Importance Rating	4.2	

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

Proposed Question: Common 74

Given the following conditions:

- Unit 1 is operating at 100% power.
- 1-ALB-5C-1.1, RV FLANGE LKOFF TEMP HI is alarming.
- 1-TI-5400A, CNTMT AVE TEMP is indicating 95°F
- 1-TI-401, RV FLANGE LKOFF TEMP is indicating 165°F.
- Three Containment Fan Coolers are in service.

Which of the following actions should be performed?

A. Start an additional Containment Fan Cooler per SOP-801A, Containment Ventilation System.

B. Open 1/1-8032, RV SEAL LKOFF VLV and perform OPT-303, Reactor Coolant System Water Inventory.

C. Make a Containment Entry to close 1RC-8069B, RV 1-01 HEAD INNER SL LKOFF ISOL VLV and open 1RC-8069A, RV 1-01 HEAD OUTER SL LKOFF ISOL VLV.

D. Make a Containment Entry to close 1RC-8069A, RV 1-01 HEAD OUTER SL LKOFF ISOL VLV, and open 1RC-8069B, RV 1-01 HEAD INNER SL LKOFF ISOL VLV.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this is the correct action if Containment average temperature is greater than or equal to 110°F.
B. Incorrect. Plausible because OPT-303 should be performed to determine leak rate, however, 1/1-8032 should be closed to isolate leakoff flow.
C. Correct. When conditions permit, Containment entry is made and the valve alignment listed performed.
D. Incorrect. Plausible because both of these valves must be manipulated, however, 1-RC-8069A, Reactor Vessel Outer Seal Leakoff Valve should be <u>opened</u> and 1-RC-8069B, Reactor Vessel Inner Seal Leakoff Valve should be <u>closed</u>.

Technical Reference(s) ALM-0053A, 1-ALB-5C, Window 1.1 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **STATE** the performance and design attributes of the following Reactor Vessel and Internals System components, flowpaths, and features:

- Reactor Vessel Head
- Penetrations and “O” Rings

Question Source: Bank # X
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 ALM-0053A, 1-ALB-5C, Window 1.1	Revision # 7

CPNPP ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0053A
ALARM PROCEDURE 1-ALB-5C	REVISION NO. 7	PAGE 7 OF 71

ANNUNCIATOR NOM./NO.: **RV FLANGE LKOFF TEMP HI** **1.1**

PROBABLE CAUSE:

High Containment temperature
Reactor vessel O-ring failure

AUTOMATIC ACTIONS: None

OPERATOR ACTIONS:

1. VERIFY 1-TI-5400A, CNTMT AVE TEMP is <110°F.
   A. IF temperature is ≥110°F,
      THEN
      START an additional containment fan cooler per SOP-801A.
2. MONITOR 1-TI-401, RV FLANGE LKOFF TEMP.
3. CLOSE 1/1-8032, RV SEAL LKOFF VLV.
4. NOTIFY Chemistry to increase monitoring of containment atmosphere to detect possible outer O-ring failure.
5. PERFORM OPT-303 to determine leakage rate, as applicable.
6. WHEN conditions permit,
   THEN
   PERFORM a containment entry per STA-620 to align outer O-ring seal leakoff to RCDT.
   A. CLOSE 1RC-8069B, RV 1-01 HEAD INNER SL LKOFF ISOL VLV.
   B. OPEN 1RC-8069A, RV 1-01 HEAD OUTER SL LKOFF ISOL VLV.
   C. OPEN 1/1-8032, RV SEAL LKOFF VLV.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Category #		
	K/A #	G 2.4.46	
	Importance Rating	4.2	

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

Proposed Question: Common 75

Given the following conditions:

- Unit 1 is operating at 100% power.
- ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, is being performed due to a trip of Main Feedwater Pump 1-01.

Which of the following annunciators is the Unit Supervisor required to be informed of per ODA-102, Conduct of Operations?

A. 1-ALB-8A, Window 2.12 – SG 2 LEV DEV

B. 1-ALB-6D, Window 1.10 – AVG TAVG – TREF DEV

C. 1-PCIP Window 3.4 – TURB LOAD REJ STM DMP ARMED C-7

D. 1-ALB-6D, Window 4.14 – CONTROL ROD BANK D FULL WTHDRWL

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a level deviation could exist, however, this information need not be reported to the Unit Supervisor.
B. Incorrect. Plausible because a temperature deviation will exist, however, this information need not be reported to the Unit Supervisor.
C. Incorrect. Plausible because a load reject will have occurred, however, this information need not be reported to the Unit Supervisor.
D. Correct. Per Conduct of Operations guidance, any power change should be secured if this alarm is annunciated as it has the potential to impact plant safety.

Technical Reference(s): ODA-102, Step 6.25.1; ALM-0064A, 1-ALB-6D, Window 4.14 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source:	Bank #	X	
	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 ODA-102, Step 6.25.1	Revision # 26

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 26 INFORMATION USE	PAGE 31 OF 37

6.25 Communications

The following guidelines establish standards for maintaining effective, consistent and accurate transfer of information during training, normal and emergency conditions. Additional attributes of communications and expectation may be found in NMG-114.

6.25.1 Control Room Communications

- The RO should always verbalize control board manipulations to ensure the US is cognizant of his actions.
- Control Board annunciators should be verbalized when acknowledged or silenced. If an annunciator is acknowledged or silenced by a Licensed Operator other than the assigned Unit RO, the RO should be notified as soon as possible.
- [C] During transients which result in multiple alarms (e.g., reactor trip), the alarms may be silenced during the performance of the Immediate Actions to allow for better communications between Control Room Operators. Upon completion of the Immediate Actions, the ROs should acknowledge the annunciators on their assigned Control Boards, and notify the US of any annunciator not previously identified which could impact plant safety. [25166]
- [C] During emergencies, a verbal verification between the US and the RO shall be implemented on a step by step basis during the performance of the ERGs. [09536]

Comments / Reference: From ALM-0064A, 1-ALB-06D, Window 4.14	Revision # 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 147 OF 147

ANNUNCIATOR NOM./NO.: **CONTROL ROD BANK D FULL WTHDRWL** 4.14

PROBABLE CAUSE:

Excessive boration
Excessive turbine loading rate
Rod Control System malfunction
Rod Referencing
Axial Flux Difference Control
Physics Test

AUTOMATIC ACTIONS:

Stops control bank D automatic rod withdrawal (C-11)

OPERATOR ACTIONS:

1. Stop all secondary power changes.

2. Verify 1-SC-CBD1, CTRL BANK D GROUP 1 and 1-SC-CBD2, CTRL BANK D GROUP 2 step counters indicates ≤231 steps.
   A. If bank D rods are >231 steps, refer to ABN-712.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	011 G 2.2.4	
Importance Rating		3.6

Large Break LOCA: Emergency Procedures/Plan: (multi-unit) Ability to explain the variations in control board layouts, systems, instrumentation and procedural actions between units at a facility

Proposed Question: SRO 76

Given the following conditions:

- Unit 2 is responding to Large Break Loss of Coolant Accident (LOCA).
- Containment pressure is 30 psig and lowering.
- Reactor Coolant System pressure is 30 psig and lowering.
- Wide range T_{HOT} indications in three loops are 280°F and lowering.
    - The other wide range T_{HOT} indication is 210°F and lowering.
- Refueling Water Storage Tank level is 63% and lowering.
- All safety systems functioned as designed.
- While checking intact Steam Generators (SG) the following levels are observed:
    - SG 2-01 is 45% and stable.
    - SG 2-02 is 8% and stable.
    - SG 2-03 is 15% and stable.
    - SG 2-04 is 35% and stable.

Which of the following actions is required to ensure the Steam Generator level control band is maintained per EOP-1.0B, Loss of Reactor or Secondary Coolant?

Raise the level in ...

A. ...Steam Generator 2-02 above 10%.

B. ...Steam Generators 2-02 & 2-03 above 18%.

C. ...Steam Generators 2-02, 2-03 & 2-04 above 43%.

D. ...Steam Generators 2-01, 2-02, 2-03 & 2-04 above 50%.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the control band on Unit 2 without Adverse Containment is 10% to 50%.
B. Correct. The control band on Unit 2 with Adverse Containment is 18% to 50%.
C. Incorrect. Plausible because the control band without Adverse Containment on Unit 1 is 43% to 60%.
D. Incorrect. Plausible because the control band on Unit 1 with Adverse Containment is 50% to 60%.

Technical Reference(s): EOP-1.0A, Step 3; EOP-1.0B, Step 3 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the operator actions, including all cautions, notes, RNOs, and bases associated with EOP-1.0, Loss of Reactor or Secondary Coolant.

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 5

Comments / Reference: From EOP-1.0A, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-1.0A
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 5 OF 44

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Check If Any SG Is Faulted:	
	a. Check pressures in all SGs • ANY SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER -OR- • ANY SG COMPLETELY DEPRESSURIZED	a. Go to Step 3.
	b. Verify all faulted SG(s) isolated: • Steamlines • Feedlines • Blowdown and sample lines	b. Go to EOP-2.0A, FAULTED STEAM GENERATOR ISOLATION, Step 1.
* 3	Check Intact SG Levels:	
	a. Narrow range level - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)	a. Maintain total AFW flow greater than 460 gpm until narrow range level greater than 43% (50% FOR ADVERSE CONTAINMENT) in at least one intact SG.
	b. Control AFW flow to maintain narrow range level between 43% (50% FOR ADVERSE CONTAINMENT) and 60%	b. IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.

Comments / Reference: From EOP-1.0B, Step 3	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-1.0B
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 5 OF 44

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2	Check If Any SG Is Faulted:	
	a. Check pressures in all SGs • ANY SG PRESSURE DECREASING IN AN UNCONTROLLED MANNER -OR- • ANY SG COMPLETELY DEPRESSURIZED	a. Go to Step 3.
	b. Verify all faulted SG(s) isolated: • Steamlines • Feedlines • Blowdown and sample lines	b. Go to EOP-2.0B, FAULTED STEAM GENERATOR ISOLATION, Step 1.
* 3	Check Intact SG Levels:	
	a. Narrow range level - GREATER THAN 10% (18% FOR ADVERSE CONTAINMENT)	a. Maintain total AFW flow greater than 460 gpm until narrow range level greater than 10% (18% FOR ADVERSE CONTAINMENT) in at least one intact SG.
	b. Control AFW flow to maintain narrow range level between 10% (18% FOR ADVERSE CONTAINMENT) and 50%.	b. IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN go to EOP-3.0B, STEAM GENERATOR TUBE RUPTURE, Step 1.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	022 AA2.02	
Importance Rating		3.7

Loss of Reactor Coolant Makeup: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Charging pump problems

Proposed Question: SRO 77

Given the following conditions:

- Unit 1 is at 100% power.
- Centrifugal Charging Pump (CCP) 1-01 is in service.
- Volume Control Tank (VCT) level is 50%.
- 1-FI-121A, CHRG FLO is stable at 130 gpm.
- 1-FI-132, LTDN FLO is stable at 120 gpm.
- 1/1-LCV-112B, VCT TO CHRG PMP SUCT VLV spuriously closes.

Which of the following actions must be taken?

A. Open 1/1-LCV-112D or 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV and close 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV per SOP-103A, Chemical and Volume Control System.

B. Stop Centrifugal Charging Pump 1-01 and close the Letdown and Orifice Isolation Valves per ABN-105, Chemical and Volume Control System Malfunction.

C. Open 1/1-LCV-112D or 1/1-LCV-112E, RWST TO CHRG PMP SUCT VLV and close the Letdown and Orifice Isolation Valves per SOP-103A, Chemical and Volume Control System.

D. Stop Centrifugal Charging Pump 1-01 and close 1/1-LCV-112C, VCT TO CHRG PMP SUCT VLV per ABN-105, Chemical and Volume Control System Malfunctions.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the precautions of SOP-103 caution against degraded CCP pump performance due to gas intrusion, however, ABN-105 requires stopping the CCP to protect the pump and isolation of Letdown is required anytime Charging is secured.

B. Correct. This is a symptom of pump cavitation due to suction valve being closed. The RNO actions of ABN-105 direct the operator to stop the CCP and isolate Letdown.

C. Incorrect. Plausible because Letdown and Orifice Isolation Valves should be closed, however, this would be performed per ABN-105 and the Charging Pump must be secured.

D. Incorrect. Plausible because the CCP is stopped to protect the pump and isolation of Letdown is required anytime charging is secured, however, closing 1/1-LCV-112C would only be required if 1/1-LCV-112D or 1/1-LCV-112E were opened.

Technical Reference(s): ABN-105, Step 7.3.1 RNO; SOP-103A, Section 4.1, Limitations — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EVALUATE** plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrumentation while responding to a Chemical and Volume Control System malfunction.

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 ABN-105, Step 7.3.1 RNO	Revision # 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-105
CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 28 OF 41

7.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Operating a CCP with symptoms of cavitation or gas binding may cause rapid pump failure.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Verify VCT conditions - NORMAL	
☐ a. VCT TO CHRG PMP SUCT VLVs - OPEN: • 1/u-LCV-112B • 1/u-LCV-112C	a. Perform the following: 1) Ensure ALL charging pumps STOPPED. 2) Ensure letdown isolated. [C] 3) OPEN BOTH VCT TO CHRG PMP SUCT VLVs. IF either valve does NOT remain OPEN, AND charging pump is needed immediately, THEN perform the following: A) OPEN RWST TO CHRG PMP SUCT VLVs: • 1/u-LCV-112D • 1/u-LCV-112E B) CLOSE 1/u-LCV-112B AND 1/u-LCV-112C

Comments / Reference: From SOP-103A, Section 4.1, Limitations	Revision # 17

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-103A
CHEMICAL AND VOLUME CONTROL SYSTEM	REVISION NO. 17	PAGE 9 OF 131

4.1 Limitations (continued)

- Letdown flow is limited to 170 gpm (when RCS temp is < 500 degrees) with 1 Mixed Bed Demineralizer in service. (Reference EVAL-2005-001409-01-00)

- Letdown flow is limited to 195 gpm (when RCS temp is < 500 degrees) when 2 demineralizers are in service. (Reference FDA-2007-001435-01-00)

[C]
- Seal injection to a coupled RCP may be secured if RCS level is above or below the seal package provided that the following actions are implemented to minimize the exposure to risk associated with this configuration:
  - Seal injection should be in service any time RCS level is moving through the seal package.
  - The #1 seal leak off isolation valves should be closed.
  - No pump should be rotated while seal injection is secured; this will prevent cycling water through the shaft alley and seal package.
  - The time with seal injection secured should be limited to the time required to perform maintenance on the Chemical and Volume Control System (CVCS) and testing / surveillances that require seal injection to be secured.
  - The RCP Oil Lift system should remain secured during the time that seal injection is isolated to prevent movement of the shaft and possibly cycling water through the shaft and seal package.
  - The pumps shall be hand rotated with the RCS at Low Pressure and Seal Injection in service to assist in dislodging any debris/deposits, prior to pump operation.
  - A flush of the seals at a higher seal injection flow rate may be used to purge any debris or unfiltered water from the seal package and shaft alley, if necessary.

  Although additional risk is incurred by securing seal injection to a coupled pump, this added risk to the seals may be mitigated by implementing the above actions. (Reference EVAL-2007-002946-01-0)

- During certain conditions, it may be necessary to start the CCP before an operator can be dispatched to locally start the Aux Lube Oil Pump. The start of a CCP without starting the Aux Lube Oil Pump is classified as an emergency start and the following limitations apply:
  - Any emergency start of a CCP should be recorded in the Unit Log.
  - WHEN the Aux Lube Oil Pump has been operated within the last 30-day period THEN CCP bearings retain sufficient lubrication for a CCP start without prior start of the Aux Lube Oil Pump.
  - ABNs and ERGs have been evaluated to determine which instructions for a CCP start are considered to be an "emergency" start of the CCP. WHEN ABN instructions reference that a CCP start be performed per SOP-103A, THEN the Aux Lube Oil Pump is expected to be started prior to starting the CCP. ABN instructions that initiate start of a CCP WITHOUT reference to SOP-103A can be performed as an emergency start. Any CCP start within the ERGs is considered an emergency start.
- Degraded CVCS Pump performance may be caused by gas instrusion into the CVCS System. Gas intrusion into the CVCS System may cause fluctuations in OR a reduction in CVCS Pump discharge pressure OR flow, OR increased pump vibration (Reference STA-698, "Gas Instrusion Program").

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	029 G 2.2.40	
Importance Rating		4.7

ATWS: Equipment Control: Ability to apply Technical Specifications for a system

Proposed Question: SRO 78

Given the following conditions:

- Unit 1 is at 100% power.
- A valid First Out alarm was received and the Reactor did not automatically trip.
- Attempts to trip the Reactor from the Control Room were NOT successful.
- The crew responded to the ATWS per FRS-0.1A, Response to Nuclear Power Generation / ATWT.
- Efforts to open the Reactor Trip Breakers in the field were NOT successful due to mechanical binding.
- Rod Drive Motor Generator Sets 1 and 2 are stopped.
- The plant is in MODE 3 with all Control Banks and Shutdown Banks fully inserted.

Which of the following describes the Technical Specification inoperability and Applicability?

A. Reactor Trip System Instrumentation Function 19, Reactor Trip Breakers are inoperable but NOT Applicable in the current MODE;
Engineered Safety Feature Actuation System Instrumentation Function 8.a, Reactor Trip P-4 is inoperable and Applicable in the current MODE.

B. Reactor Trip System Instrumentation Function 19, Reactor Trip Breakers are inoperable and Applicable in the current MODE;
Engineered Safety Feature Actuation System Instrumentation Function 8.a, Reactor Trip P-4 is inoperable and Applicable in the current MODE.

C. Reactor Trip System Instrumentation Function 19, Reactor Trip Breakers are OPERABLE and Applicable in the current MODE;
Engineered Safety Feature Actuation System Instrumentation Function 8.a, Reactor Trip P-4 is inoperable but NOT applicable in the current MODE.

D. Reactor Trip System Instrumentation Function 19, Reactor Trip Breakers are OPERABLE but NOT Applicable in the current MODE;
Engineered Safety Feature Actuation System Instrumentation Function 8.a, Reactor Trip P-4 is OPERABLE but NOT applicable in the current MODE.

Proposed Answer: A

Explanation:

A. Correct. Both the Reactor Trip Breakers and the P-4 ESFAS signal are inoperable with the Reactor Trip Breakers unable to be opened, however, with the MG sets stopped and all rods inserted, the Reactor Trip Breaker function is not applicable in MODE 3 with rod control incapable of rod withdrawal. P-4 is applicable in MODE 3.
B. Incorrect. Plausible as the conditions necessary to make the Reactor Trip Breaker function not applicable must be recognized, as the Rod Control System must not be capable of rod withdrawal, with no rods withdrawn.
C. Incorrect. Plausible because Reactor Trip P-4 is inoperable, however, the Reactor Trip Breakers are not OPERABLE. Further, the applicability of both specifications is incorrect for the current plant conditions.
D. Incorrect. Plausible as the applicability for the Reactor Trip Breakers is correct; however, the applicability for P-4 is incorrect. The P-4 OPERABILITY is also incorrect.

Technical Reference(s): Technical Specification Table 3.3.1-1, #19; Technical Specification Table 3.3.2-1, #8.a; FRS-0.1A, Step 6; IPO-001A, Attachment 5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the applicability of FRS-0.1, Response to Nuclear Generation/ATWT, regarding MODES and the ACTIONS taken if the procedure is otherwise used.

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 Table 3.3.1-1, #19	Amendment # 156

RTS Instrumentation
3.3.1

Table 3.3.1-1 (page 4 of 6)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE(a)
16. Turbine Trip					
a. Low Fluid Oil Pressure	1(j)	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 46.6 psig
b. Turbine Stop Valve Closure	1(j)	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(e)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 50.7% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 52.7% RTP
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 7.3% RTP and ≤ 12.7% RTP
f. Turbine First Stage Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 12.7% turbine power
19. Reactor Trip Breakers(RTBs)(k)	1,2	2 trains	R	SR 3.3.1.4	NA
	3(b), 4(b), 5(b)	2 trains	C	SR 3.3.1.4	NA

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 2a, 2b, 6, 7, and 14 (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
(b) With Rod Contol System capable of rod withdrawal or one or more rods not fully inserted.
(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(j) Above the P-9 (Power Range Neutron Flux) interlock.
(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Comments / Reference: From Technical Specification Table 3.3.2-1, #8.a	Amendment # 156

ESFAS Instrumentation
3.3.2

Table 3.3.2-1 (page 6 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1, 2, 3, 4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 31.9% instrument span
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1, 2, 3	1 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1, 2, 3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1975.2 psig (Unit 1) ≤ 1976.4 psig (Unit 2)

(a) The Allowable Value defines the limiting safety system except for functions 5b and 6c (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

Comments / Reference: From FRS-0.1A, Step 6	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 8	PAGE 6 OF 30

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Check PRZR pressure - LESS THAN 2335 PSIG	Verify PRZR PORVs and block valves open. IF NOT, THEN open PRZR PORVs and block valves as necessary until PRZR pressure less than 2185 psig.
6	Check If The Following Trips Have Occurred:	
	a. Reactor - TRIPPED	a. Dispatch operators to locally trip reactor: At reactor switchgear: • Trip reactor trip and bypass breakers A and B. • Stop Rod drive MG sets 1 and 2 At normal switchgear: • Trip Rod drive MG sets 1 and 2 motor breakers on 1B3/8C/BKR and 1B4/8C/BKR.

Comments / Reference: From IPO-001A, Attachment 5	Revision # 21

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-001A
PLANT HEATUP FROM COLD SHUTDOWN TO HOT STANDBY	REVISION NO. 21	PAGE 69 OF 93

ATTACHMENT 5
PAGE 1 OF 2

CHECKLIST REQUIRED PRIOR TO CLOSING REACTOR TRIP BREAKERS

NOTE: Per TE-93-1911, the following methods are considered to make the Rod Control System NOT capable of rod withdrawal, even with the Reactor Trip Breakers CLOSED:

- Shutdown both Rod Drive MG Sets or,
- Open both MG Set Generator Breakers or,
- Disconnect all CRDM cables or,
- Remove all CRDM fuses or,
- Place ALL Lift Coil Disconnect Switches in ROD DISCONNECT (OPEN) position

A. To CLOSE the Reactor Trip or Bypass breakers with the Rod Control System NOT capable of rod withdrawal

1. Verify the following:

- ALL Control AND Shutdown Rods are known to be fully inserted ____/____ Initials Date
- IF in MODE 3, THEN ensure at least two RCPs are in operation (TS 3.4.5 only requires two RCPs to be operable and one RCP in operation):

  RCP 1 ______ RCP 2 ______ RCP 3 ______ RCP 4 ______ ____/____ Initials Date
- The Rod Control System is NOT capable of rod withdrawal ____/____ Initials Date

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	026 AA2.04	
Importance Rating		2.9

Loss of Component Cooling Water: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW

Proposed Question: SRO 79

Given the following conditions on Unit 1:

- High temperature alarms are received on all 4 Reactor Coolant Pumps (RCP).
- All 4 RCP Thermal Barrier Return temperatures indicate 185ºF and rising at 5ºF per minute.
- Lower Seal Bearing temperature indicates 200ºF and rising at 5ºF per minute.

Which of the following describes the status of RCP thermal barrier cooling, and the action required for this condition?

A. 1-HV-4696, THBR CLR CCW RET ISOL VLV (IRC), is closed.
RCP temperatures exceed the operating limits and must be immediately tripped per ABN-502, Component Cooling Water System Malfunctions.

B. 1-HV-4709, THBR CLR CCW RET ISOL VLV (ORC), is closed.
RCPs must be tripped within 5 minutes due to high temperature per ABN-101, Reactor Coolant Pump Trip/Malfunction.

C. 1-HV-4709, THBR CLR CCW RET ISOL VLV (ORC), is closed.
RCP temperatures exceed the operating limits and must be immediately tripped per ABN-502, Component Cooling Water System Malfunctions.

D. 1-HV-4696, THBR CLR CCW RET ISOL VLV (IRC), is closed.
RCPs must be tripped within 5 minutes due to high temperature per ABN-101, Reactor Coolant Pump Trip/Malfunction.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because closure of 1-HV-4696 could result in a rise in temperatures as shown, however, this condition is addressed by ABN-101 and RCP temperature limits have not yet been exceeded.

B. Correct. 1-HV-4709 closes when temperatures exceed 182.5°F. Per ABN-101, RCPs must be tripped when operating limits are exceeded and this will occur within 5 min. per the conditions in the Stem.

C. Incorrect. Plausible because 1-HV-4709 is closed, however, RCP temperatures have not yet exceeded the limits and the procedure reference is incorrect.

D. Incorrect. Plausible because the RCPs must be tripped within 5 min. and the procedure reference is correct, however, 1-HV-4696 closes when flow is greater than or equal to 64 gpm and this condition has not been identified.

Technical Reference(s): ABN-101, Section 8.1 & 8.2; ABN-101, Step 8.3.2; ABN-101, Attachment 1; ABN-502, Section 6.1 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to an RCP High Temperature or Loss of CCW to any RCP in accordance with ABN-101, Reactor Coolant Pump Trip/Malfunction.

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 ABN-101, Section 8.1 & 8.2	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 35 OF 48

8.0 RCP HIGH TEMPERATURE OR LOSS OF CCW TO ANY RCP

8.1 Symptoms

a. Annunciator Alarms

- ANY RCP THBR CLR CCW RET TEMP HI (3B-2.11)
- ANY RCP THBR CLR CCW RET FLO LO (3B-3.11)
- ANY RCP MOTOR CLR CCW RET FLO LO (3B-2.12)
- ANY RCP UP BRG L\O CLR CCW RET FLO LO (3B-3.12)
- ANY RCP LOW BRG L\O CLR CCW RET FLO LO (3B-4.12)

b. Plant Indications

- Computer alarms on RCP bearing temperatures
- Computer alarm on RCP motor winding temperatures

8.2 Automatic Actions

NOTE: Closure of u-HS-4709 or u-HS-4696 isolates CCW return from ALL RCPs.

a. High thermal barrier CCW return temperature (182.5°F) will cause the following:

1) Auto closure of Thermal Barrier Cooler CCW Return Valve for affected pumps(s)

- u-HS-4691 RCP 1 THBR CLR CCW RET VLV
- u-HS-4692 RCP 2 THBR CLR CCW RET VLV
- u-HS-4693 RCP 3 THBR CLR CCW RET VLV
- u-HS-4694 RCP 4 THBR CLR CCW RET VLV

2) Auto closure of u-HS-4709, THBR CLR CCW RET ISOL VLV (ORC)

b. High thermal barrier return flow will cause auto closure of u-HS-4696, THBR CLR CCW RET ISOL VLV (IRC)

Comments / Reference: From ABN-101, Step 8.3.2	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 36 OF 48

8.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify seal injection flow to ALL RCP(s) - GREATER THAN OR EQUAL TO 6 GPM per pump.	GO TO Section 9.0 of this procedure.
☐ [C] 2 Verify RCP parameters within OPERATING LIMITS per Attachment 1.	Perform the following: a. Trip Reactor AND GO TO EOP-0.0A/B while other operators continue this procedure. b. Stop affected RCP(s) c. Increase seal injection flow to affected RCP(s) NOT to exceed 13 gpm, as necessary. d. GO TO Section 2.0 of this procedure.

Comments / Reference: From ABN-101, Attachment 1	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-101
REACTOR COOLANT PUMP TRIP/MALFUNCTION	REVISION NO. 10	PAGE 46 OF 48

ATTACHMENT 1
PAGE 1 OF 1
RCP PARAMETERS

NOTE: The following list may aid determination of the validity of a temperature alarm or indication change:

- Local RTD (stator) monitoring (System Engineering/I&C) outside bioshield
  U1- RTD terminals: TBX-RCDARK-01[RCP 1, 2]; TBX-RCDARK-02 [RCP 3, 4]
  U2- RTD terminals: TCX-RCDARK-01[RCP 1, 2]; TCX-RCDARK-02 [RCP 3, 4]
- Thermographic performance comparison between pumps (System Engineering/Predictive Maintenance)
- Local evidence of restricted air flow
- Vibration change
- RCP motor amps high or changing
- Affected RCP loop flow or temperature change
- Bus voltage high or low, phase imbalance
- RCP motor air cooler air outlet temperature change
- Affected cooler CCW inlet/outlet temperature change
- Loose Parts Monitoring System alarm
- RCP seal leakoff or injection, flow or temperature change

Monitor the parameters below, as determined by Unit Supervisor:

IF motor bearing temperature is greater than or equal to 190°F, THEN perform Section 3.0 for RCP High or Low Lube Oil Level, while continuing.

IF motor bearing temperature increases by approximately 2°F from previous reading AND NO significant change in L\O Cooler CCW temperatures is observed, THEN notify System Engineering and Duty Manager.

IF any RCP bearing oil reservoir alarm LIT, THEN perform Section 3.0 while continuing section in effect.

RCP OPERATING LIMITS					
PARAMETER LIMIT		RCP 1	RCP 2	RCP 3	RCP 4
MOT STAT WNDG TEMP	300°F	T0412A	T0432A	T0452A	T0472A
MOT UP RDL BRG TEMP	195°F	T0413A	T0433A	T0453A	T0473A
MOT UP THR BRG TEMP	195°F	T0414A	T0434A	T0454A	T0474A
MOT LOW RDL BRG TEMP	195°F	T0415A	T0435A	T0455A	T0475A
MOT LOW THR BRG TEMP	195°F	T0416A	T0436A	T0456A	T0476A
LOW SEAL WTR BEARING TEMP (Pump Bearing)	225°F	T0417A	T0437A	T0457A	T0477A
SEAL WTR IN TEMP	235°F	T0181A	T0182A	T0183A	T0184A

Comments / Reference: From ABN-502, Section 6.1	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-502
COMPONENT COOLING WATER SYSTEM MALFUNCTIONS	REVISION NO. 6	PAGE 34 OF 75

6.0 LOSS OF ALL CCW FLOW

6.1 Symptoms

a. Annunciator Alarms

- ANY NEUT DET WELL OUT TEMP HI (3B-1.2)
- SFP HX 1 CCW RET FLO LO (3B-1.8)
- ANY RCP L/O/MOTOR CLR CCW RET TEMP HI (3B-1.12)
  - u-TI-4720, RCP 1 MOTOR AIR CLR CCW RET TEMP
  - u-TI-4721, RCP 2 MOTOR AIR CLR CCW RET TEMP
  - u-TI-4722, RCP 3 MOTOR AIR CLR CCW RET TEMP
  - u-TI-4723, RCP 4 MOTOR AIR CLR CCW RET TEMP
- RCDT HX CCW RET FLO LO (3B-1.14)
- WSTE GAS COMPR 2 SEAL WTR CLR CCW RET FLO LO (3B-1.15)
- SEAL WTR HX CCW RET FLO LO (3B-1.16)
- CCW SRG TK TRN A/B EMPTY (3B-2.2)
- CCWP 1/2 OVRLOAD/TRIP (3B-2.3)
- SFP HX 2 CCW RET FLO LO (3B-2.8)
- ANY RCP THBR CLR CCW RET TEMP HI (3B-2.11)
- ANY RCP MOTOR CLR CCW RET FLO LO (3B-2.12)
- H_2 RCMB 2 CCW RET FLO LO (3B-2.15)
- ANY CNTMT FN CLR FN DISCH TEMP HI (3B-3.2)
- CCW HX 1/2 OUT & RECIRC FLO LO (3B-3.5)
- ANY RCP THBR CLR CCW RET FLO LO (3B-3.11)
- ANY RCP UP BRG L/O CLR CCW RET FLO LO (3B-3.12)
- PDP L/O CLR CCW RET FLO LO (3B-3.14)
- H_2 RCMB 1 CCW RET FLO LO (3B-3.15)
- WSTE EVAP VENT CNDSR CCW RET FLO LO (3B-3.16)
- CRDM ANY VENT FN DISCH TEMP HI (3B-4.2)
- CCW HX 1/2 SPLY FLO LO (3B-4.5)
- ANY RCP LOW BRG L/O CLR CCW RET FLO LO (3B-4.12)
- XS LTDN HX CCW RET FLO LO (3B-4.13)
- WSTE GAS COMPR 1 SEAL WTR CLR CCW RET LO (3B-4.14)
- WSTE EVAP DISTL CLR CCW RET FLO LO (3B-4.16)
- FLR DRN EVAP VENT CNDSR CCW RET FLO LO (4A-1.2)
- FLR DRN EVAP DISTL CLR CCW RET FLO LO (4A-2.2)
- BRS EVAP DISTL CLR CCW RET FLO LO (4A-3.1)
- LTDN HX OUT TEMP HI (6A-1.3)

b. Plant Indications

- Temperatures increasing on component(s) supplied by CCW.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	040 G 2.4.41	
	Importance Rating		4.6

Steam Line Rupture: Emergency Procedures/Plan: Knowledge of the emergency action level thresholds and classifications

Proposed Question: SRO 80

Given the following conditions:

- Unit 2 has experienced a steam line fault in the Main Steam Header.
- Security has reported significant damage to the Safeguards Building.
- Two of the four Main Steam Line Isolation Valves have failed to close.
- Personnel cannot enter the Main Steam Header room due to steam and building damage.

Which of the following Emergency Action Level category applies?

A. Fission Product Barriers, CSFST, Reactor Coolant System Barrier

B. Fission Product Barriers, Inventory, Containment Barrier

C. Hazards, Natural or Destructive Phenomena

D. Hazards, Fire or Explosion

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because with a Main Steam Line Break resulting in significant damage to the piping penetration room, the plant's ability to protect the Containment, RCS and CSFST Fission Product Barriers are all potentially challenged, however, with the given conditions a Loss or Potential Loss does not actually exist making this EAL call incorrect.

B. Incorrect. Plausible because with a Main Steam Line Break resulting in significant damage to the piping penetration room, the plant's ability to protect the Containment, RCS and CSFST Fission Product Barriers are all potentially challenged, however, with the given conditions a Loss or Potential Loss does not actually exist making this EAL call incorrect.

C. Incorrect. Plausible because the Safeguards Building is identified as a critical structure in the Hazards category, however, the cause would be either earthquake, tornado, flooding, turbine failure generated projectile, or vehicle crash.

D. Correct. Explosion is defined as a rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures. This scenario is identified in the conditions of the Stem.

Technical Reference(s): EPP-201, Common & Hot EALs Table; EPP-201, EAL Technical Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Emergency Action Level Charts; Emergency Action Level Technical Bases

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

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 1

Comments / Reference: From EPP-201, Common EALs Table	Revision # 12

H

Hazards

2

Fire or Explosion

Note 9: Explosion is defined as a rapid, violent, unconfined combustion, or catastrophic failure of pressurized/energized equipment that imparts energy of sufficient force to potentially damage permanent structures, systems, or components. A steam line break or steam explosion that damages surrounding permanent structures or equipment would be classified under this EAL.

Comments / Reference: From EPP-201, Common EALs Table	Revision # 12

Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown (1 2 3 4 5 6 DEF)	Fire within the Protected Area not extinguished within 15 min. of detection or explosion within the Protected Area (1 2 3 4 5 6 DEF)
HA2.1 (Bases Page 173) Fire or explosion resulting in **EITHER**: • Visible damage to any Table H-1 structures • Control Room indication of degraded performance of systems required to establish or maintain safe shutdown (Note 9)	**HU2.1 (Bases Page 169)** Fire **not** extinguished within 15 min. of Control Room notification or verification of a Control Room fire alarm in **any** Table H-1 area (Note 4) **HU2.2 (Bases Page 171)** Explosion of sufficient force to damage permanent structures or equipment within the Protected Area (Note 9)

Comments / Reference: From EPP-201, Common EALs Table	Revision # 12

Table H-1 Structures Containing Systems Needed for Safe Shutdown

- u-Containment
- u-Safeguards Building
- X-Auxiliary Building
- X-Electrical & Control Building
- X-Fuel Building
- X-Service Water Intake Structure
- u-Diesel Generator Building
- u-Normal switchgear rooms
- u-CST
- u-RWST

Natural or destructive phenomena affecting Vital Areas

1	2	3	4	5	6	DEF

HA1.1 (Bases Page 154)
Seismic event > OBE as indicated by annunciator 2A-3.1, OBE EXCEEDED, or yellow OBE light on Seismic Monitoring system panel
AND
Earthquake confirmed by **any** of the following:
- Earthquake felt in plant
- National Earthquake Information Center (Note 8)
- Control Room indication of degraded performance of systems required for the safe shutdown of the plant

HA1.2 (Bases Page 157)
Tornado striking or sustained high winds > 80 mph resulting in **EITHER**:
- Visible damage to **any** Table H-1 structures
- Control Room indication of degraded performance of systems required to establish or maintain safe shutdown

HA1.3 (Bases Page 160)
Internal flooding in the Safeguards Building or Turbine Building resulting in **EITHER**:
- An electrical shock hazard that precludes access to operate or monitor systems required to establish or maintain safe shutdown
- Control Room indication of degraded performance of systems required to establish or maintain safe shutdown

HA1.4 (Bases Page 162)
Turbine failure-generated projectiles resulting in **EITHER**:
- Visible damage to or penetration of **any** Table H-1 structures
- Control Room indication of degraded performance of systems required to establish or maintain safe shutdown

Natural or destructive phenomena affecting the Protected Area

1	2	3	4	5	6	DEF

HU1.1 (Bases Page 143)
Seismic event identified by **any two** of the following:
- Annunciator 2A- 2.1, SEISMIC MONITORING SYSTEM ACTIVATION, received
- Earthquake felt in plant
- National Earthquake Information Center (Note 8)

HU1.2 (Bases Page 145)
Tornado striking within the Protected Area boundary
OR
Sustained high winds > 80 mph

HU1.3 (Bases Page 147)
Internal flooding that has the potential to affect safety-related equipment required by Technical Specifications for the current operating mode in the Safeguards Building or Turbine Building

HU1.4 (Bases Page 149)
Turbine failure resulting in casing penetration or damage to turbine or generator seals

Comments / Reference: From EPP-201, Hot Condition EALs Table	Revision # 12

1. Containment pressure rise followed by a rapid unexplained drop in Containment pressure (Bases Page 308)
2. Containment pressure or sump level response **not** consistent with LOCA conditions (Bases Page 309)
3. Ruptured SG is also faulted outside of Containment (Bases Page 311)
4. Primary-to-secondary leakrate > 10 gpm
   AND
   Unisolable steam release from affected SG to the environment (Bases Page 313)
5. Containment pressure 50 psig and rising (Bases Page 315)
6. Containment hydrogen concentration > 4% (Bases Page 316)
7. Containment pressure > 18 psig with **neither** Containment Spray system train operating (Bases Page 318)

Comments / Reference: From EPP-201, Hot Condition EALs Table	Revision # 12

Reactor Coolant System Barrier

Loss	Potential Loss
None	1. CSFST RCS Integrity - RED entry conditions met **OR** CSFST Heat Sink - RED entry conditions met and heat sink required (Bases Page 280)

Comments / Reference: From EPP-201, EAL Technical Bases	Revision # 12

Category: H – Hazards

Subcategory: 2 – Fire or Explosion

Initiating Condition: Fire or explosion affecting the operability of plant safety systems required to establish or maintain safe shutdown

EAL:

HA2.1 Alert

Fire or explosion resulting in **EITHER**:

- Visible damage to **any** Table H-1 structures
- Control Room indication of degraded performance of systems required to establish or maintain safe shutdown

Table H-1 Structures Containing Systems Needed for Safe Shutdown
- u-Containment
- u-Safeguards Building
- X-Auxiliary Building
- X-Electrical & Control Building
- X-Fuel Building
- X-Service Water Intake Structure
- u-Diesel Generator Building
- u-Normal switchgear rooms
- u-CST
- u-RWST

Mode Applicability:

All

Basis:

Generic

VISIBLE DAMAGE is used to identify the magnitude of the FIRE or EXPLOSION and to discriminate against minor FIRES and EXPLOSIONS.

The reference to structures containing safety systems or components is included to discriminate against FIRES or EXPLOSIONS in areas having a low probability of affecting safe operation. The significance here is not that a safety system was degraded but the fact that the FIRE or EXPLOSION was large enough to cause damage to these systems.

The use of VISIBLE DAMAGE should not be interpreted as mandating a lengthy damage assessment prior to classification. The declaration of an Alert and the activation of the Technical Support Center will provide the Emergency Coordinator with the resources needed to perform detailed damage assessments.

The Emergency Coordinator also needs to consider any security aspects of the EXPLOSION.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	W/E12 EA2.2	
Importance Rating		3.9

Uncontrolled Depressurization of All Steam Generators: Ability to determine and interpret the following as they apply to the Uncontrolled Depressurization of All Steam Generators: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

Proposed Question: SRO 81

Given the following conditions with the Unit in MODE 1:

- The Reactor tripped and Safety Injection occurred due to low Pressurizer pressure.
- All Steam Generator pressures were decreasing in an uncontrolled manner.
- Main Steam Isolation Valves would NOT close from the Control Room.
- After entering ECA-2.1A, Uncontrolled Depressurization of All Steam Generators, local operator actions restored the pressure boundary for all Steam Generators.
- Steam Generator pressures are stabilizing but an increase in pressure has not been seen in any Steam Generator.
- After Steam Generator 1-01 pressure stabilizes, level continues to increase in an uncontrolled manner.
- The Pressurizer is empty at this time.

Which of the following procedures should be implemented?

A. Transition to FRI-0.2A, Response to Low Pressurizer Level.

B. Continue with ECA-2.1A, Uncontrolled Depressurization of All Steam Generators.

C. Transition to EOP-3.0A, Steam Generator Tube Rupture.

D. Transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because Pressurizer level is low and FRGs could be implemented, however, it is a YELLOW path and the increasing Steam Generator level is the concern.
B. Incorrect. Plausible because this is the procedure in effect, however, the ECA-2.1A Foldout Page requires a transition to EOP-3.0A on any Steam Generator with an increasing level.
C. Correct. ECA-2.1A Foldout Page requires a transition to EOP-3.0A on any Steam Generator with an increasing level.
D. Incorrect. Plausible because with the Pressurizer empty it could be thought that a transition to EOP-1.0A is appropriate due to SI Reinitiation Criteria, however, EOP-3.0A entry is required.

Technical Reference(s): ECA-2.1A, Attachment 1A, Foldout Page; ECA-2.1A, Step 5 RNO; ECA-2.1A, Flowchart; FRI-0.2A, CSFST — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** the proper transitions out of ECA-2.1, Uncontrolled Depressurization of All Steam Generators.

Question Source:
Bank # X
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 5

Comments / Reference: From ECA-2.1A, Attachment 1A, Foldout Page	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-2.1A
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 28 OF 71

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT PAGE FOR ECA-2.1A - UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS

1. SI REINITIATION CRITERIA

   Manually start ECCS pumps as necessary if EITHER condition listed below occurs:

   - RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)

2. EOP-2.0A TRANSITION CRITERIA

   IF any SG pressure increases at any time, except while performing ECCS Termination in Step 10 to 24, THEN go to EOP-2.0A, FAULTED STEAM GENERATOR ISOLATION, Step 1.

3. EOP-3.0A TRANSITION CRITERIA

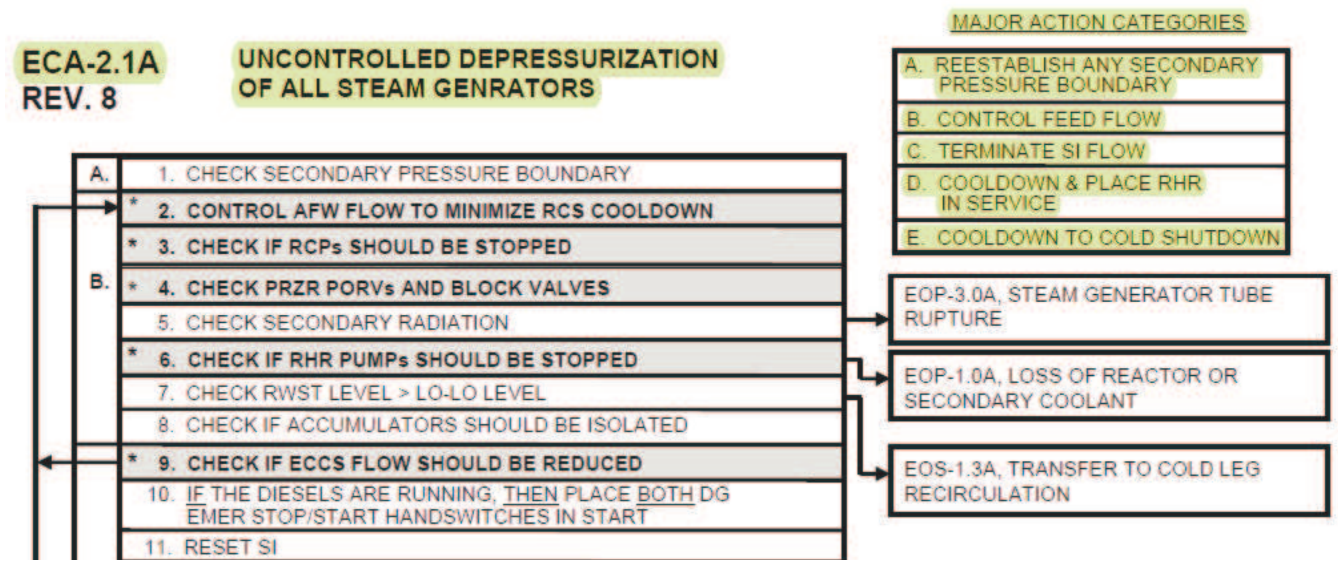
   IF any SG level increases in an uncontrolled manner or any SG has abnormal radiation, THEN manually start ECCS pumps as necessary AND go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.

Comments / Reference: From ECA-2.1A, Step 5 RNO	Revision # 8

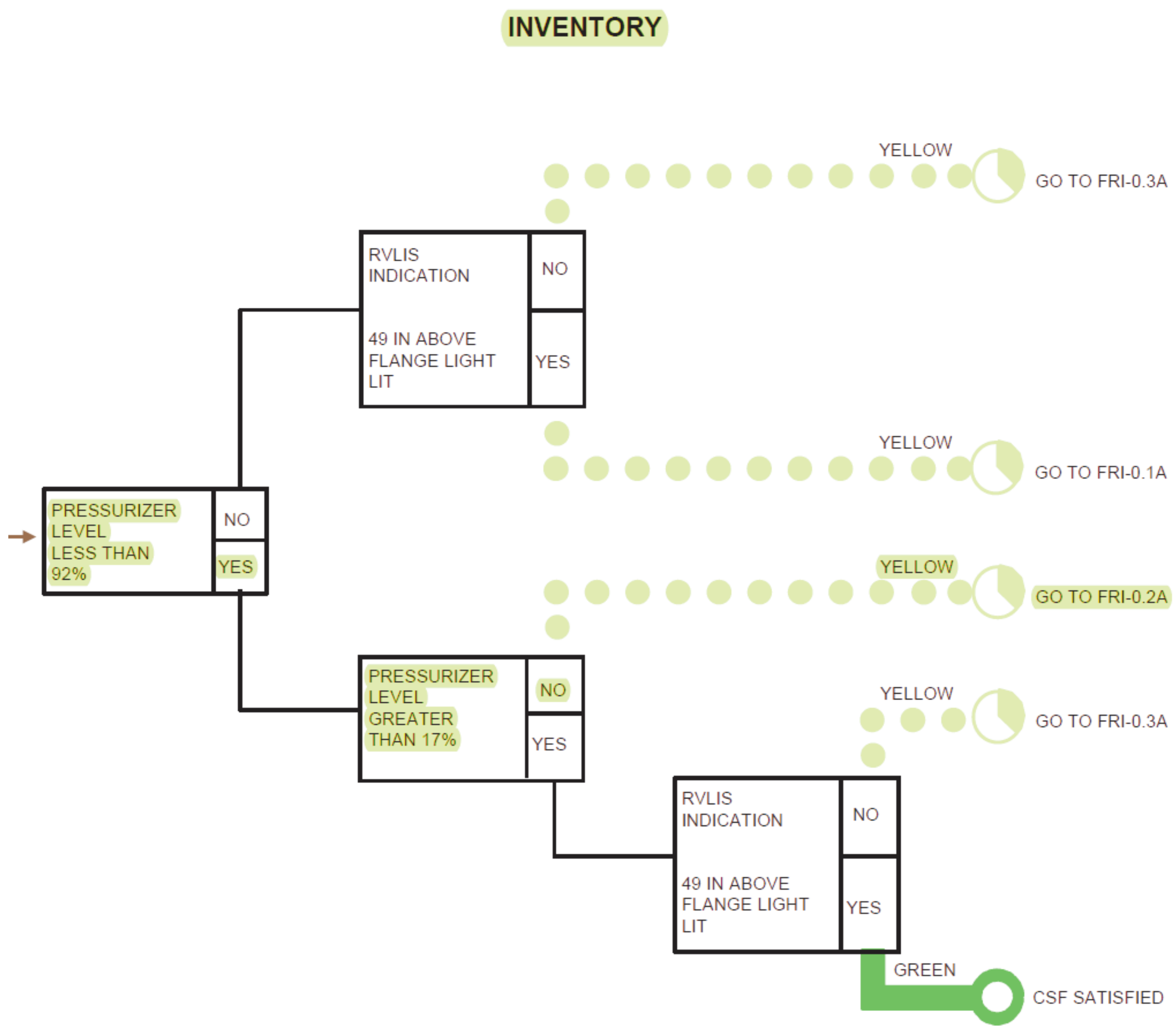
CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-2.1A
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 6 OF 71

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Check Secondary Radiation:	
	a. Request periodic activity samples of all SGs.	
	b. Check any secondary radiation monitor that is <u>NOT</u> isolated - NORMAL: • Condenser off gas radiation (COG-182, 1RE-2959) • Main steamline radiation (MSL-178 through 181, 1RE-2325 through 2328) • SG blowdown sample radiation (SGS-164, 1RE-4200)	b. Go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.

Comments / Reference: From ECA-2.1A, Flowchart	Revision # 8



Comments / Reference: From FRI-0.2A, CSFST	Revision # 8



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	037 AA2.02	
	Importance Rating		3.9

Steam Generator Tube Leak: Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Agreement/disagreement among redundant radiation monitors

Proposed Question: SRO 82

Given the following conditions:

- Unit 1 is at 100% power.
- Chemistry reports an abnormal increase in Condenser Off-Gas specific activity.
- Chemistry suspects that a Steam Generator tube leak exists.
- Chemistry calculates the leak rate is 10 gpd on Steam Generator 1-02.

Which of the following lists the two independent radiation monitors that should be used to confirm the Steam Generator tube leakage?

A. COG-182 Condenser Off-Gas Radiation Monitor, and
N16-175 # 2 Main Steam Line N16 Radiation Monitor

B. COG-182 Condenser Off-Gas Radiation Monitor, and
MSL-179 # 2 Main Steam Line Radiation Monitor

C. SGS-164 Steam Generator Blowdown Sample Radiation Monitor, and
N16-175 # 2 Main Steam Line N16 Radiation Monitor

D. SGS-164 Steam Generator Blowdown Sample Radiation Monitor, and
MSL-179 # 2 Main Steam Line Radiation Monitor

Proposed Answer: A

Explanation:

A. Correct. Per the conditions in the Stem and the guidance provided in ABN-106, these are the correct Radiation Monitors to confirm the Steam Generator Tube Leakage.

B. Incorrect. Plausible because COG-182, Condenser Off-Gas Radiation Monitor is correct, however, MSL-179, #2 Main Steam Line Radiation Monitor does not meet the minimum sensitivity setpoint required per ABN-106.

C. Incorrect. Plausible because the N16 Main Steam Line Radiation Monitor is correct and Steam Generator Blowdown will isolate on high radiation, however, this is a symptom associated with leakage greater than or equal to 75 gpd.

D. Incorrect. Plausible because both of these instruments can be used, however, MSL-179, #2 Main Steam Line Radiation Monitor does not meet the minimum sensitivity setpoint for the leak rate involved (10 gpd) and neither does the Steam Generator Blowdown Radiation Monitor.

Technical Reference(s): ABN-106, Section 2.0
ABN-106, Step 2.3.1 & 2.3.2 NOTES
ABN-106, Step 3.3.1 NOTE
ABN-106, Section 3.0

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a Steam Generator Tube Leakage less than 75 gpd in accordance with ABN-106, High Secondary Activity.

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 ABN-106, Section 2.0	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 3 OF 31

2.0 STEAM GENERATOR TUBE LEAKAGE LESS THAN 75 GPD (.052 GPM)

2.1 Symptoms

a. Annunciator Alarms

None

b. Plant Indications

- An abnormal increase in steam generator specific activity as reported by Chemistry.
- An abnormal increase in steam generator sampling radiation as indicated by u-RE-4200 (SGS-u64).
- An abnormal increase in condenser off-gas radiation as indicated by u-RE-2959 (COG-u82).
- An abnormal increase in main steamline leak rate as indicated on u-RE-2325A (N16-u74), u-RE-2326A (N16-u75), u-RE-2327A (N16-u76), and u-RE-2328A (N16-u77). Computer points R7749A(R7753A) thru R7752A(R7756A).

2.2 Automatic Actions

- Steam Generator Blowdown will isolate on high radiation as indicated on u-RE-4200 (SGS-u64).

Comments / Reference: From ABN-106, Step 2.3.1 & 2.3.2 NOTES	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 5 OF 31

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Due to the minimum sensitivity of the MSL radiation monitors, a valid alarm indicates a leak rate of at least 3600 gpd (2.5 gpm).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify main steamline radiation alarms - CLEAR • u-RE-2325 (MSL-u78) • u-RE-2326 (MSL-u79) • u-RE-2327 (MSL-u80) • u-RE-2328 (MSL-u81)	GO TO Section 3.0.

NOTE: Leakage is qualitatively confirmed when two independent radiation monitors trend in the same direction with the same order of magnitude.

☐ 2 Correlate monitor readings to leak rate, as necessary:

- (≥40% power) N16 leak rate indication.
- Posted COG correlation graphs by PC-11(≥28% power).
- CPINET, Chemistry Department, Pri-Sec leakage tab.

Comments / Reference: From ABN-106, Step 3.3.1 NOTE	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 15 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Due to the minimum sensitivity of the MSL radiation monitors, a valid alarm indicates a leak rate of at least 3600 gpd (2.5 gpm).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify main steamline radiation alarms - CLEAR • u-RE-2325 (MSL-u78) • u-RE-2326 (MSL-u79) • u-RE-2327 (MSL-u80) • u-RE-2328 (MSL-u81)	a. Initiate power reduction to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours. b. GO TO Step 4.b.

Comments / Reference: From ABN-106, Section 3.0	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 13 OF 31

3.0 STEAM GENERATOR TUBE LEAKAGE GREATER THAN OR EQUAL TO 75 GPD (.052 GPM)

3.1 Symptoms

a. Annunciator Alarms

None

b. Plant Indications

- Steam Generator leakage in excess of 75 gpd (.052 gpm) as reported by Chemistry. The reported leak rate should be verified with a second independent radiation monitor or grab sample.
- Unidentified leakage in excess of TS Limits as determined by OPT-303 which is suspected to be Steam Generator tube leakage.
- An abnormal increase in main steamline radiation as indicated on u-RE-2325 (MSL-u78), u-RE-2326 (MSL-u79), u-RE-2327 (MSL-u80), and u-RE-2328 (MSL-u81) or leak rate indication on u-RE-2325A (N16-u74), u-RE-2326A (N16-u75), u-RE-2327A (N16-u76), and u-RE-2328A (N16-u77). Computer points R7749A(R7753A) thru R7752A(R7756A).

3.2 Automatic Actions

- Steam Generator blowdown will isolate on high radiation as indicated on u-RE-4200 (SGS-u64).

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	051 G 2.2.44	
Importance Rating		4.4

Loss of Condenser Vacuum: Equipment Control: Ability to interpret control room indications to verify the status and operation of the system, and understand how operator actions and directives affect plant and system conditions

Proposed Question: SRO 83

Given the following conditions:

- Unit 1 is at 9% power.
- The Main Turbine is latched at 1800 RPM in preparation for synchronization.
- Main Condenser Vacuum is 21" and lowering.
- All Condenser Vacuum Pumps are running.
- Operators have been dispatched to determine the cause for the loss of vacuum.

Which of the following should be performed per ABN-304, Main Condenser and Circulating Water System Malfunction?

Trip the…

A. …Reactor, enter EOP-0.0A, Reactor Trip or Safety Injection, and continue actions of ABN-304, Main Condenser and Circulating Water System Malfunction.

B. …Main Turbine, enter ABN-403, Turbine Trip Response, and continue actions of ABN-304, Main Condenser and Circulating Water System Malfunction.

C. …Reactor, enter EOP-0.0A, Reactor Trip or Safety Injection, and exit ABN-304, Main Condenser and Circulating Water System Malfunction.

D. …Main Turbine, enter ABN-403, Turbine Trip Response, and exit ABN-304, Main Condenser and Circulating Water System Malfunction.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this answer is correct if Reactor power was greater than or equal to 10%, however, at this power level only a Main Turbine trip is required.
B. Correct. As outlined in ABN-304, Step 3.3.3 RNO.
C. Incorrect. Plausible because the Reactor would be tripped if Reactor power was greater than or equal to 10%, however, the loss of vacuum must still be addressed because other plant systems such as Main Feedwater Pumps and Steam Dump System would be affected.
D. Incorrect. Plausible because a Main Turbine trip is required and ABN-403 would be entered, however, the loss of vacuum must still be addressed because other plant systems such as Main Feedwater Pumps would be affected.

Technical Reference(s): ABN-304, Section 3.2; ABN-304, Step 3.3.3 RNO — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to a lowering Main or Auxiliary Condenser Vacuum per ABN-304, Main Condenser and Circulating Water System Malfunction.

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 ABN-304, Section 3.2	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-304
MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 9 OF 28

3.2 Automatic actions

- The Standby Condenser Vacuum Pump automatically starts with decreasing condenser vacuum (24" Hg).
- Condenser Available (C-9) is lost with vacuum less than 12.3" Hg.
- Main Turbine trips with vacuum less than 21" Hg.
- Main Feedwater Pump Turbine trips when both (21" Hg AND 17.5" Hg) exhaust hood low vacuum contacts for associated pump are closed.

Comments / Reference: From ABN-304, Step 3.3.3 RNO	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-304
MAIN CONDENSER AND CIRCULATING WATER SYSTEM MALFUNCTION	REVISION NO. 8	PAGE 10 OF 28

3.3 Operator Actions

	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1	Start ALL Available Condenser Vacuum Pumps. • u-HS-2956, CNDSR VAC PMP 1 • u-HS-2957, CNDSR VAC PMP 2 • u-HS-2958, CNDSR VAC PMP 3	
☐ 2	Dispatch an operator to verify CEV seal water tank - LEVEL INDICATED	Perform the following: a. Stop affected CEV b. Bypass seal water solenoid valve to refill tank. c. Restart CEV pump, if required, per SOP-309A/B.
☐ 3 [C]	Verify Main Condenser Vacuum - GREATER THAN 21" HG • Main Cond. Vacuum on TG Control Display • u-PI-2042-1, CNDSR A PRESS • u-PI-2042-2, CNDSR B PRESS	IF Reactor Power is greater than or equal to 10%, THEN trip Reactor AND GO TO EOP-0.0A/B while others continue this procedure. IF Reactor Power is less than 10%, THEN trip Turbine AND perform ABN-403 while continuing this procedure.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	W/E03 EA2.1	
	Importance Rating		4.2

LOCA Cooldown - Depressurization: Ability to determine and interpret the following as they apply to the LOCA Cooldown and Depressurization: Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question: SRO 84

Given the following conditions:

- A Steam Line Break has occurred.
- Normal Charging is being established per EOS-1.1A, Safety Injection Termination.
- The faulted Steam Generator has stopped depressurizing.
- While attempting to control Pressurizer level with normal Charging, Pressurizer level continues to lower.

Which of the following describes the action required?

A. Actuate Safety Injection and transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

B. Actuate Safety Injection and transition to EOP-0.0A, Reactor Trip or Safety Injection.

C. Realign the Centrifugal Charging Pump Injection flowpath and transition to EOS-1.2A, Post LOCA Cooldown and Depressurization.

D. Realign the Centrifugal Charging Pump Injection flowpath and remain in EOS-1.1A, Safety Injection Termination.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because once SI is terminated the Foldout Page requires transition to EOP-1.0 after starting ECCS Pumps.
B. Incorrect. Plausible because this would be correct if events occurred in EOS-0.1A, Reactor Trip Response. EOS-1.1A requires manual operation and transition to correct procedure for LOCA.
C. Correct. Per EOS-1.1A, Step 11 RNO the Safety Injection Pumps are not stopped, Letdown is not established, and crew is either in the wrong procedure or another event has occurred. Equipment alignment at this point would require transition to EOS-1.2A for more appropriate recovery.
D. Incorrect. Plausible because the Charging Pump injection flowpath should be realigned, however, the RNO action of Step 11 requires entry into EOS-1.2A, Post LOCA Cooldown and Depressurization.

Technical Reference(s): EOS-1.1A, Step 11 RNO; EOS-1.1A, Steps 2 & 15 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOS-1.1, Safety Injection Termination.

Question Source: Bank # X
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 EOS-1.1A, Step 11 RNO	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.1A
SAFETY INJECTION TERMINATION	REVISION NO. 8	PAGE 6 OF 48

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*11	Control Charging Flow To Maintain PRZR Level	IF any SG is faulted, THEN do NOT proceed until faulted SG depressurization stops or PRZR level can be maintained. IF no SG faulted OR PRZR level continues to decrease after faulted SG depressurization stops, THEN realign CCP injection flow path. Go to EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION, Step 1.

Comments / Reference: From EOS-1.1A, Step 2	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.1A
SAFETY INJECTION TERMINATION	REVISION NO. 8	PAGE 3 OF 48

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

CAUTION: If offsite power is lost after SI reset, manual action may be required to restart safeguards equipment.

[1D] 1 IF The Diesels Are Running, THEN Place Both DG EMER STOP/START Handswitches In START.

[1D] 2 Reset SI.

[1D] 3 Reset SI Sequencers.

[1D] 4 Reset Containment Isolation Phase A And Phase B.

[1D] 5 Reset Containment Spray Signal.

Comments / Reference: From EOS-1.1A, Step 15	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.1A
SAFETY INJECTION TERMINATION	REVISION NO. 8	PAGE 8 OF 48

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
[R] *15	Check If Letdown Can Be Established:	
	a. PRZR level - GREATER THAN 30% (50% FOR ADVERSE CONTAINMENT)	a. Continue with Step 16. WHEN PRZR level increases to greater then 30% (50% FOR ADVERSE CONTAINMENT) THEN do Step 15b.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		1
Group #		2
K/A #	W/E14 G 2.4.49	
Importance Rating		4.4

High Containment Pressure: Emergency Procedures/Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls

Proposed Question: SRO 85

Given the following conditions:

- Unit 2 entered EOP-0.0B, Reactor Trip or Safety Injection due to a Reactor trip and Safety Injection with Containment Spray actuation.
- The actions of EOP-0.0B have been completed and a diagnosis has been made that a loss of Reactor Coolant inside Containment exists.
- At the time of entry into EOP-1.0B, Loss of Reactor or Secondary Coolant, an ORANGE Path condition is noted on the Containment Status tree when 2-ALB-4B Window 1.8 – RWST 2 OF 4 LVL LO-LO annunciates and level is confirmed to be at 33%.

Which of the following is the proper course of actions to implement?

A. Transition to FRZ-0.1B, Response to High Containment Pressure and complete the actions without delay so that EOS-1.3B, Transfer to Cold Leg Recirculation can be implemented to realign Emergency Core Cooling System injection.

B. Complete Steps 1, 2 and 3 of EOS-1.3B, Transfer to Cold Leg Recirculation and EOP-1.0B, Loss of Reactor or Secondary Coolant. Performance of FRZ-0.1B, Response to High Containment Pressure is not required as EOP-0.0B verified conditions for Containment Spray.

C. Transition to FRZ-0.1B, Response to High Containment Pressure and concurrently implement actions of EOS-1.3B, Transfer to Cold Leg Recirculation. When completed, return to EOP-1.0B, Loss of Reactor or Secondary Coolant.

D. Complete Steps 1, 2 and 3 of EOS-1.3B, Transfer to Cold Leg Recirculation then review Critical Safety Function Status Trees. If ORANGE condition on Containment still exists, then transition to FRZ-0.1B, Response to High Containment Pressure.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Critical Safety Function ORANGE Path is immediately addressed, however, entry and completion of EOS-1.3B takes priority.

B. Incorrect. Plausible because completing the actions of EOS-1.3B is correct, however, the Critical Safety Function ORANGE Path cannot be ignored.

C. Incorrect. Plausible because the last two actions listed are correct, however, EOP-1.0B is not implemented at this time.

D. Correct. Given the conditions listed, this is the correct set of actions to complete.

Technical Reference(s): FRZ-0.1B, CSFST Flowchart
EOP-1.0B, Attachment 1.A, Foldout Page
ODA-407 Attachment 8A
EOS-1.3B, Step 1 CAUTION

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a set of plant conditions, **IDENTIFY** the proper transitions through/out of EOS-1.3, Transfer to Cold Leg Recirculation.

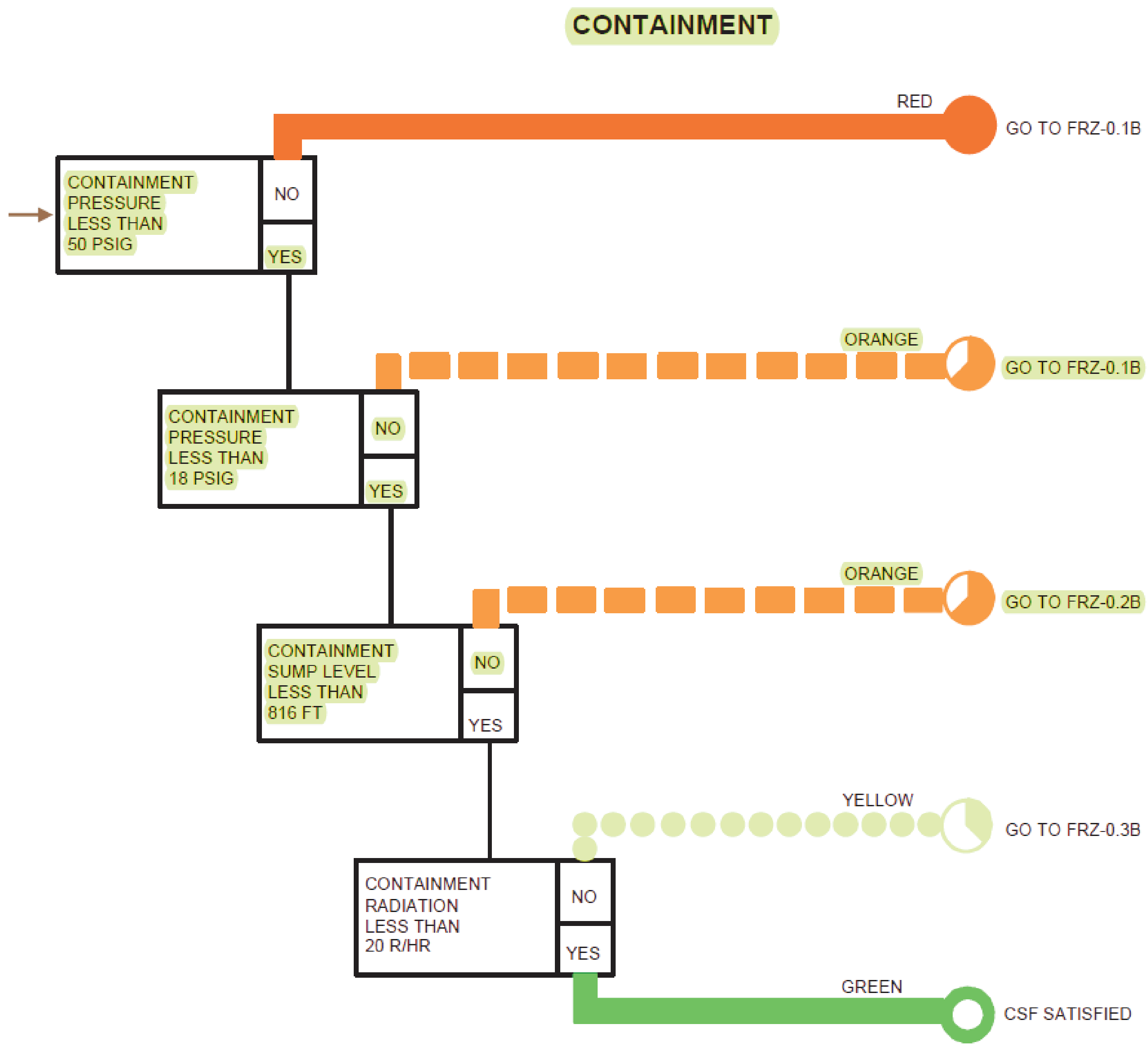
Question Source:
Bank # X
Modified Bank # ____ (Note changes or attach parent)
New ____

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 FRZ-0.1B, CSFST Flowchart	Revision # 8



Comments / Reference: From EOP-1.0B, Attachment 1.A, Foldout Page	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-1.0B
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 18 OF 44

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOP-1.0B, LOSS OF REACTOR OR SECONDARY COOLANT

1. RCP TRIP CRITERIA

   Trip all RCPs if BOTH conditions listed below occur:

   a. RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)

   b. CCP or SI pump - AT LEAST ONE RUNNING

2. SI REINITIATION CRITERIA

   Manually start ECCS pumps as necessary if EITHER condition listed below occurs:

   - RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
   - PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (15% FOR ADVERSE CONTAINMENT)

3. SECONDARY INTEGRITY CRITERIA

   IF any SG pressure is decreasing in an uncontrolled manner or has completely depressurized and has NOT been isolated, THEN go to EOP-2.0B FAULTED STEAM GENERATOR ISOLATION, Step 1.

4. EOP-3.0A TRANSITION CRITERIA

   IF any SG level increases in an uncontrolled manner or any SG has abnormal radiation, THEN manually start ECCS pumps as necessary AND go to EOP-3.0B, STEAM GENERATOR TUBE RUPTURE, Step 1.

5. COLD LEG RECIRCULATION SWITCHOVER CRITERION

   IF RWST level decreases to less than LO-LO LEVEL, THEN go to EOS-1.3B, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

Comments / Reference: From ODA-407 Attachment 8.A	Revision # 14

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 14 INFORMATION USE	PAGE 23 OF 46

ATTACHMENT 8.A
PAGE 8 OF 22
ERG RULES OF USAGE

10. • In general, performance of the FRGs is dependent on current plant parameters. If a RED or ORANGE priority condition comes in and clears before FRG implementation is initiated, the FRG need not be performed. If conditions degrade, the safety function status will become a continuous RED or ORANGE condition; at which time, the operator will be directed to the appropriate FRG.

An exception to this rule is made for entry into FRZ-0.1A/B after transition out of EOP-0.0A/B. The corresponding containment pressure for an ORANGE priority condition of FRZ-0.1A/B is also the Containment Spray initiation setpoint; thus, the containment pressure value impacts FRG status and implementation. The following provides a summary of requirements for FRZ-0.1A/B.

Scenarios Affecting FRZ-0.1A/B Status	**Requirements for Implementing FRZ-0.1A/B**
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition HAS CLEARED when FRG implementation is initiated.	IF FRZ ORANGE condition has CLEARED when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN performance of FRZ-0.1A/B is NOT required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions AND there is NOT currently a challenge to the Containment barrier.
Containment Spray initiates during EOP-0.0A/B performance as Containment Pressure reaches 18 psig (FRZ-0.1A/B ORANGE priority condition exists). Step 7 of EOP-0.0A/B is performed to verify proper Containment Spray alignment. The FRZ ORANGE condition STILL EXISTS when FRG implementation is initiated.	IF an FRZ ORANGE condition exists when FRG implementation is initiated (transition out of EOP-0.0A/B OR EOP-0.0A/B step initiates CSF monitoring AND automatic action verification complete), THEN FRZ-0.1A/B performance is required. Proper response for Containment Spray actuation has been verified with EOP-0.0A/B actions BUT a challenge to the Containment barrier may exist.
EOP-0.0A/B is performed without Containment Spray actuation (Step 7 of EOP-0.0A/B identifies Containment Spray NOT required/NOT verified). The appropriate recovery actions are initiated with transition out of EOP-0.0A/B. The FRZ ORANGE condition COMES IN AND remains in during implementation of recovery actions (after FRG implementation initiated).	All CSFSTs are monitored and FRGs are implemented per the rules of usage. FRG implementation is initiated based on the highest CSF priority. IF an FRZ ORANGE condition exists, THEN FRZ-0.1A/B performance is required. A challenge to the Containment barrier exists AND proper response for Containment Spray actuation is verified to minimize challenges to the Containment barrier.

Comments / Reference: From ODA-407 Attachment 8.A	Revision # 14

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 14 INFORMATION USE	PAGE 27 OF 46

ATTACHMENT 8.A
PAGE 12 OF 22
ERG RULES OF USAGE

13. Certain contingency procedures take precedence over the FRGs due to specific initiating events. These procedures are identified by a note at the beginning of the EOP. Examples: ECA-0.0A/B, ECA-0.1A/B, ECA-0.2A/B and EOS-1.3A/B. EOS-1.3A/B is a specific example of a procedure which takes precedence over the FRGs because not performing EOS-1.3A/B could cause a loss of core cooling and inventory. EOS-1.3A/B should be performed as soon as possible after the RWST LO-LO level is reached.

Comments / Reference: From EOS-1.3B Step 1 CAUTION	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-1.3B
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 3 OF 53

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: Steps 1 through 3 should be performed without delay. FRGs should not be implemented prior to completion of these steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1	Reset SI.	
2	Verify CCW Flow As Required: • From RHR heat exchangers • From Containment Spray heat exchangers	Establish CCW flow to RHR or Containment Spray heat exchanger(s) as required.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	012 A2.01	
Importance Rating		3.6

Reactor Protection System: Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Faulty bistable operation

Proposed Question: SRO 86

Given the following conditions:

- Annunciator 1-ALB-6D, Window 1.3 – 1 OF 4 HI SETPT PR FLUX HI, is in alarm.
- All Power Range Nuclear Instruments read approximately 100%.
- OVERPOWER TRIP HIGH RANGE light for Power Range Nuclear Instrument N-42 is LIT on the NIS Panel due to failure of the High Setpoint High Flux Trip Bistable.
- ABN-703, Power Range Instrument Malfunction, is being performed to place the channel out-of-service for repairs.

Which of the following identifies the impact on the Reactor Protection System and what action should be taken to mitigate the situation?

A. An OP HI FLUX ROD STOP C-2 is generated and cannot be bypassed.
Reactor Trip bistables for Loop 2 must be placed in TRIP within one hour.

B. A Power Range High Flux Trip will be generated but can be blocked.
If the Reactor is to remain at 100% RTP, the QUADRANT POWER TILT RATIO must be determined using Core Power Distribution Measurement information.

C. A Power Range High Flux Trip will be generated but cannot be blocked.
Reactor Trip bistables for Loop 2 must be placed in TRIP within one hour.

D. An OP HI FLUX ROD STOP C-2 is generated and can be bypassed.
If the Reactor is to remain at 100% RTP, the QUADRANT POWER TILT RATIO must be determined using Core Power Distribution Measurement information.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because C-2 Rod Stop is generated, however, N16 Power Channel Defeat Switch must be aligned to Loop 2 but the bistables need not be placed in trip and the Rod Stop can be bypassed.
B. Incorrect. Plausible because a Power Range High Flux Trip will be generated and the performance action is correct, however, the trip cannot be blocked.
C. Incorrect. Plausible because the Power Range High Flux Trip will be generated and cannot be blocked, however, the bistables need not be placed in trip.
D. Correct. The C-2 Rod Stop is generated and QPTR must be monitored once every 12 hours.

Technical Reference(s): ABN-703, Sections 2.1, 2.2, & Step 2.3.2; Technical Specification SR 3.2.4.2 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the normal, abnormal and emergency operation of the Nuclear Instrumentation System.
APPLY the administrative requirements of the Nuclear Instrumentation System including Technical Specifications, TRM and ODCM.

Question Source:
- Bank # X
- 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 2, 5

Comments / Reference: From ABN-703, Section 2.1	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 3 OF 23

2.0 POWER RANGE INSTRUMENTATION MALFUNCTION

2.1 Symptoms

a. Annunciator Alarms

- 1 OF 4 OT N-16 HI (5C-2.5)
- 1 OF 4 HI SETPT PR FLUX HI (6D-1.3)
- 1 OF 4 LO SETPT PR FLUX HI (6D-2.3)
- 1 OF 4 PR FLUX RATE HI (6D-3.3)
- PR HI VOLT FAIL (6D-4.3)
- RX ≥50% PWR UP PR DET FLUX DEV HI (6D-1.4)
- RX ≥50% PWR LOW PR DET FLUX DEV HI (6D-2.4)
- PR CHAN DEV (6D-3.4)
- QUADRANT PWR TILT (6D-4.10)
- OP HI FLUX ROD STOP C-2 (6D-2.14)
- 1 OF 4 OT N-16 ROD STOP & TURB RUNBACK (6D-3.14)

b. Plant Indications

- Loss of the INSTRUMENT POWER ON or CONTROL POWER ON lights on the nuclear instrumentation cabinet drawers for the failed channel.
- Lighting of the LOSS OF DETECTOR VOLT, OVERPOWER TRIP HIGH RANGE, OVERPOWER ROD STOP, lights on the nuclear instrumentation cabinet drawer for the failed channel.

Comments / Reference: From ABN-703, Section 2.2	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 4 OF 23

2.1 b.
- Upscale, downscale, or erratic indication of the PERCENT FULL POWER or the upper or lower MICROAMPERES DETECTOR CURRENT meters on the nuclear instrumentation cabinet drawers for the failed channel.
- Lighting of the POSITIVE RATE TRIP lights on the nuclear instrumentation cabinet drawer for the failed channel, if the failure caused a rate of change of greater than or equal to 5% within 2 seconds.
- Lighting of the CHANNEL DEVIATION light on the comparator and rate drawer.

2.2 Automatic Actions

NOTE: The power range channels are designed with coincidence requirements for operational reliability. For that reason, an individual channel failure will cause an annunciator alarm and the OP HI FLUX ROD STOP C-2 with 1/4 channels at 103% of full power. No other safety system actuations will occur due to coincidence requirements.

- IF a power range channel fails HIGH while the rod control system is in automatic, THEN control rods will be rapidly inserted.
- A power range channel failure LOW will cause no control response.

Comments / Reference: From ABN-703, Step 2.3.2	Revision # 8

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-703
POWER RANGE INSTRUMENTATION MALFUNCTION	REVISION NO. 8	PAGE 5 OF 23

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify rapid control rod insertion - NOT REQUIRED a. Reactor and Turbine Power - MATCHED -AND- Tave less than 3°F above Tref. b. Place Rod Control in MANUAL	Perform the following: 1. Monitor rod motion AND Tave. 2. Ensure Tave restored to programmed temperature. 3. Investigate cause of system upset. 4. IF NO instrument failure/malfunction is indicated, THEN return to procedure and step in effect.
☐ 2 Verify Reactor Power LESS THAN 75% rated thermal power (RTP).	Initiate actions to comply with Technical Specification SR 3.2.4.2.

Comments / Reference: From Tech Spec SR 3.2.4.2	Amendment # 156

QPTR
3.2.4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	-----------------------------------NOTES----------------------------------- 1. With input from one Power Range Neutron Flux channel inoperable and THERMAL POWER ≤ 75% RTP, the remaining three power range channels can be used for calculating QPTR. 2. SR 3.2.4.2 may be performed in lieu of this Surveillance. --- Verify QPTR is within limit by calculation.	In accordance with the Surveillance Frequency Control Program.
SR 3.2.4.2	-----------------------------------NOTE----------------------------------- Not required to be performed until 12 hours after input from one or more Power Range Neutron Flux channels are inoperable with THERMAL POWER > 75% RTP. --- Verify QPTR is within limit using the core power distribution measurement information.	In accordance with the Surveillance Frequency Control Program.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	039 A2.03	
	Importance Rating		3.7

Main and Reheat Steam System: Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Indications and alarms for main steam and area radiation monitors (during SGTR)

Proposed Question: SRO 87

Given the following conditions:

- Unit 2 is at 100% power.
- A small Steam Generator Tube Leak (120 gpd) on Steam Generator 2-01 caused 2-RE-2325A (N16-274), Main Steam Line #1 N16 Radiation Monitor to go into RED alarm.
- Turbine load is being lowered per ABN-106, High Secondary Activity.

Which of the following describes the impact on 2-RE-2325A (N16-274), Main Steam Line #1 N16 Radiation Monitor indication when Turbine load is reduce and what action is required per ABN-106, High Secondary Activity?

2-RE-2325A (N16-274), Main Steam Line #1 N16 Radiation Monitor indication should decrease due to the decrease in...

A. ...N16 production.
Continue with a normal shutdown to be in MODE 3 in ≤ 24 hours.

B. ...Iodine production.
Continue with a normal shutdown to be in MODE 3 in ≤ 24 hours.

C. ... N16 production.
Reduce power to ≤ 50% in 1 hour <u>and</u> be in MODE 3 in the next 2 hours.

D. ... Iodine production.
Reduce power to ≤ 50% in 1 hour <u>and</u> be in MODE 3 in the next 2 hours.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because a leakrate ≥ 75 gpd sustained for ≥ 1 hour, a normal shutdown is performed with MODE 3 entry in ≤ 24 hours. The Radiation Monitor reading on 2-RE-2325A will lower due to a decrease in N16 production.

B. Incorrect. Plausible because the actions are correct for > 75 gpd but iodine production does not cause the change in N16 indication.

C. Correct. N16 production will lower and primary to secondary leakage ≥ 100 gpd requires shutdown to < 50% within 1 hour and MODE 3 in the next 2 hours.

D. Incorrect. Plausible because actions are correct but not due to iodine production.

Technical Reference(s) ABN-106, Steps 3.3.1 to 3.3.3 and NOTES Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DEMONSTRATE** an understanding of the components of the Main Steam System including interrelations with other systems to include interlocks and control loops.

ANALYZE the response to a Steam Generator Tube Leakage greater than or equal to 75 gpd per ABN-106, High Secondary Activity.

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 4, 5

Comments / Reference: From ABN-106, Step 3.3.1	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 15 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Due to the minimum sensitivity of the MSL radiation monitors, a valid alarm indicates a leak rate of at least 3600 gpd (2.5 gpm).

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 1 Verify main steamline radiation alarms - CLEAR • u-RE-2325 (MSL-u78) • u-RE-2326 (MSL-u79) • u-RE-2327 (MSL-u80) • u-RE-2328 (MSL-u81)	a. Initiate power reduction to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours. b. GO TO Step 4.b.

(STA-732, Attachment 8.B) INFORMATION ONLY - ACTIONS DIRECTED BY THIS PROCEDURE

NOTE:
- Leakage is qualitatively confirmed when two independent radiation monitors trend in the same direction with the same order of magnitude. With two confirmed indications (i.e. N-16 and COG monitors or other combination of monitor indication and sample analyses):
- CPNPP uses the CONSTANT LEAKAGE METHOD.

LEAKAGE/LEAK RATE	ACTION
Primary to secondary leakage ≥75 gpd (.052 gpm) sustained for ≥1 hour	Normal shutdown to be in MODE 3 in ≤24 hours
Primary to secondary leakage ≥100 gpd (.07 gpm) OR Primary to secondary leakage ≥75 gpd (.052 gpm) sustained for ≥1 hour AND NO condenser off-gas radiation monitor available AND main steam line leak rate radiation monitor on affected SG(s) - NOT OPERABLE	Reduce power to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours

Comments / Reference: From ABN-106, Steps 3.3.2 & 3.3.3	Revision # 10

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE NO. ABN-106
HIGH SECONDARY ACTIVITY	REVISION NO. 10	PAGE 16 OF 31

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 2 Correlate monitor readings to leak rate • (≥40% power) N16 leak rate indication. • (>28% power)Posted COG correlation graphs by PC-11. • CPINET, Chemistry Department, Pri-Sec leakage tab.	

CAUTION: The RNO action of Step 3 should be initiated WHEN leak rate increases to 100 gpd (.07 gpm) (i.e. from 75 gpd (.052 gpm) to 100 gpd) during the 24 hour shutdown window of step 3.a.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify leak rate <100 gpd (.07 gpm):	Perform the following:
a. Be in Mode 3 in ≤24 hours.	a. Reduce power to ≤50% in 1 hour AND Be in MODE 3 in the next 2 hours.
b. Continue monitoring leak rate and leak rate, rate of change.	b. GO TO Step 4.b.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	004 G 2.2.25	
Importance Rating		4.2

Chemical and Volume Control System: Equipment Control: Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits

Proposed Question: SRO 88

The limitation for a maximum of two Charging Pumps to be OPERABLE when less than 350°F is based on...

A. ...the reduced ECCS flow requirements when less than 350°F.

B. ...a mass addition pressure transient being relieved by a single PORV.

C. ...not exceeding the maximum flow rate to the RCS when less than 350°F.

D. ...preventing excessive cooldown of the Reactor Vessel Cold Leg nozzle.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because there is a reduced ECCS flow requirement in MODE 4 when only one ECCS train shall be OPERABLE per Technical Specification LCO 3.5.3, however, limitations for the Charging Pumps are based on a single PORV in service.
B. Correct. As outlined in Technical Specification Surveillance Requirement (SR) 3.4.12.1.
C. Incorrect. Plausible because there are Low Temperature Overpressure Protection System requirements while in MODE 4, however, limitations in the LCO are not applicable when >320°F with one RCP and operation, Pressurizer level ≤ 92%, and heat up rate limited to 60°F per hour.
D. Incorrect. Plausible because The Reactor Vessel Cold Leg Nozzles are referenced in Technical Specification LCO 3.5.3 Bases, however, limitations for the Charging Pumps are based on a single PORV in service.

Technical Reference(s): Technical Specification SR 3.4.12.1; Technical Specification SR 3.4.12.1 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Chemical and Volume Control System including Technical Specifications, TRM and ODCM.

Question Source:
Bank # X
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 2

Comments / Reference: From Technical Specification SR 3.4.12.1	Amendment # 156

LTOP System
3.4.12

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One required RCS relief valve inoperable in MODE 5 or 6.	F.1 Restore required RCS relief valve to OPERABLE status.	24 hours
G. Two required RCS relief valves inoperable. OR Required Action and associated Completion Time of Condition A, B, D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ 2.98 square inches.	8 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of zero safety injection pumps are capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program.
SR 3.4.12.2	Verify a maximum of two charging pumps are capable of injecting into the RCS.	In accordance with the Surveillance Frequency Control Program.

Comments / Reference: From Technical Specification SR 3.4.12.1 Bases	Revision # 67

LTOP System
B 3.4.12

BASES

ACTIONS

G.1 (continued)

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1, SR 3.4.12.2, and SR 3.4.12.3

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of zero safety injection pumps and a maximum of two charging pumps are verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out. Verification that each accumulator is isolated is only required when accumulator isolation is required as stated in Note 1 to the Applicability.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	059 A2.01	
Importance Rating		3.6

Main Feedwater System: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system

Proposed Question: SRO 89

Given the following conditions:

- Unit 2 is at 35% power.
- Main Feedwater Pump 2A is in operation.
- Main Feedwater Pump 2B windmilling is in progress because the associated Turning Gear is inoperable.
- Motor Driven Auxiliary Feedwater (MDAFW) Pump 2-02 is tagged out.

Which of the following is taken to ensure automatic Auxiliary Feedwater actuation due to trip of both Main Feedwater Pumps and which Technical Specification is met by the action?

Main Feedwater Pump [1] trip oil pressure switches are isolated to comply with Technical Specification LCO 3.3.2, ESFAS Instrumentation, Function 6, Auxiliary Feedwater, [2].

	[1]	[2]
A.	2A	Automatic Actuation Logic and Actuation Relays
B.	2B	Automatic Actuation Logic and Actuation Relays
C.	2A	Trip of all Main Feedwater Pumps
D.	2B	Trip of all Main Feedwater Pumps

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that it was the auto actuation logic for Main Feedwater Pump 2A.
B. Incorrect. Plausible because Main Feedwater Pump 2B is the correct pump, however, the wrong Technical Specification Function.
C. Incorrect. Plausible because the trip oil pressure switches on a Main Feedwater Pump 2B must be isolated, however, Main Feedwater Pump 2B is the correct pump.
D. Correct. The trip oil pressure switches on a Main Feedwater Pump 2B must be isolated to comply with Technical Specification LCO 3.3.2, Function 6.g.

Technical Reference(s): Technical Specification LCO Table 3.3.2-1 / SOP-302B, 5.1.2 / Technical Specification LCO 3.3.2.G & J — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Chemical and Volume Control System including Technical Specifications, TRM and ODCM.

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 Table 3.3.2-1	Amendment # 156

ESFAS Instrumentation
3.3.2

Table 3.3.2-1 (page 5 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater					
a. Automatic Actuation Logic and Actuation Relays (Solid State Protection System)	1, 2, 3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
b. Not Used.					
c. SG Water Level Low-Low	1, 2, 3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥37.5% of narrow range span (Unit 1) [q][r] ≥34.9% of narrow range span (Unit 2) [q][r]
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
e. Loss of Offsite Power	1, 2, 3	1 per train	F	SR 3.3.2.7 SR 3.3.2.9 SR 3.3.2.10	NA
f. Not Used.					
g. Trip of all Main Feedwater Pumps	1, 2	2 per AFW pump	J	SR 3.3.2.8	NA
h. Not Used.					

(a) The Allowable Value defines the limiting safety system except for functions 5b and 6c (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

Comments / Reference: From Technical Specification LCO 3.3.2.G	Amendment # 156

ESFAS Instrumentation
3.3.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. One train inoperable.	-----------------------NOTE----------------------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. ---	
	G.1 Restore train to OPERABLE status.	24 hours
	OR	
	G.2.1 Be in MODE 3.	30 hours
	AND	
	G.2.2 Be in MODE 4.	36 hours

Comments / Reference: From Technical Specification LCO 3.3.2.J	Amendment # 156

ESFAS Instrumentation
3.3.2

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One channel inoperable.	----------NOTE---------- One channel may be bypassed for up to 12 hours for surveillance testing. ------------------------------	
	I.1 Place channel in trip.	72 hours
	<u>OR</u>	
	I.2 Be in MODE 3.	78 hours
J. One Main Feedwater Pump trip channel inoperable.	J.1 Place channel in trip.	6 hours
	<u>OR</u>	
	J.2 Be in MODE 3.	12 hours

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		1
K/A #	103 G 2.4.4	
Importance Rating		4.7

Containment System: Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures

Proposed Question: SRO 90

Given the following conditions:

- Unit 1 experienced a Design Basis Accident 20 minutes ago.
- All systems responded as expected with the following indications:
  - Containment pressure is 16 psig and lowering.
  - Neutron Source Range flux is on scale and stable.
  - Reactor Coolant System (RCS) Subcooling is 0°F.
  - Reactor Vessel Level Indication System lights from 11" through 49" above flange are LIT.
  - All Steam Generator narrow range levels are 50% to 60%.
  - All RCS Cold Leg temperatures are 270°F to 275°F.
  - Containment Sump level is 817 ft.
  - Pressurizer level is 0%.
- EOP-0.0A, Reactor Trip or Safety Injection, has been exited and the Shift Technical Advisor (STA) is reviewing Critical Safety Function Status Trees.

Which of the following Critical Safety Function Status Trees should the STA recommend entering?

A. FRS-0.2A, Response to Loss of Core Shutdown.

B. FRC-0.3A, Response to Saturated Core Cooling.

C. FRP-0.2A, Response to Anticipated Pressurized Thermal Shock.

D. FRZ-0.2A, Response to Containment Flooding.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because a review of the CSFST indicates that FRS-0.2A conditions are met, however, the ORANGE path for CONTAINMENT overrides the YELLOW path for SUBCRITICALITY.
B. Incorrect. Plausible because a review of the CSFST indicates that FRC-0.2A conditions are met, however, the ORANGE path for CONTAINMENT overrides the YELLOW path for CORE COOLING.
C. Incorrect. Plausible because a review of the CSFST indicates that FRP-0.2A conditions are met, however, the ORANGE path for CONTAINMENT overrides the YELLOW path for INTEGRITY.
D. Correct. Given the conditions listed, entry into FRZ-0.2A is the correct Functional Restoration Procedure.

Technical Reference(s): FRZ-0.2A, CSFST; FRP-0.2A, CSFST; FRC-0.3A, CSFST; FRS-0.2A, CSFST — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the symptoms or Entry Conditions for FRZ-0.2, Response to Containment Flooding.

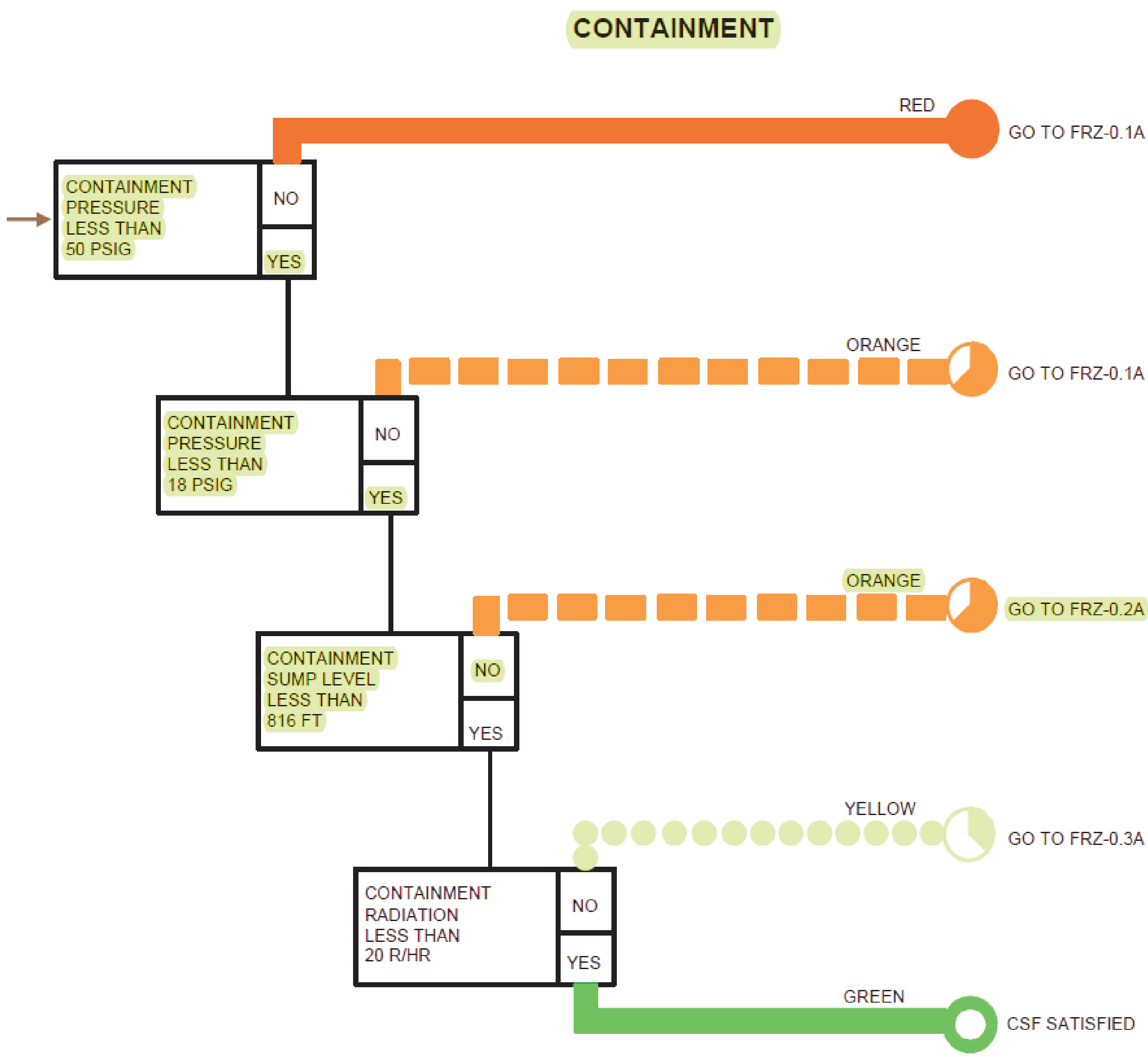
Question Source:
Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History: Last NRC Exam

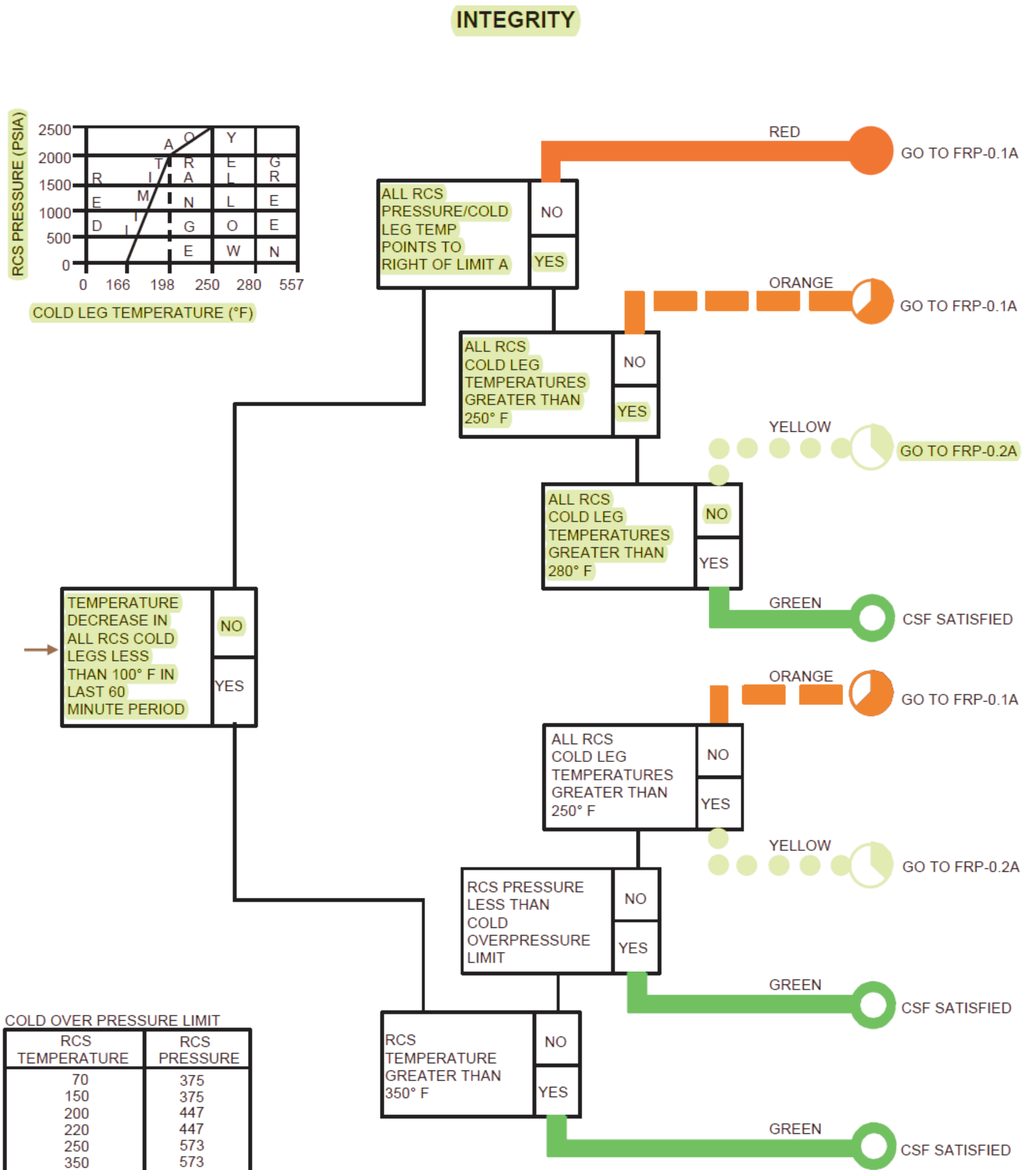
Question Cognitive Level:
Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content:
55.41
55.43 5

Comments / Reference: From FRZ-0.2A, CSFST	Revision # 8



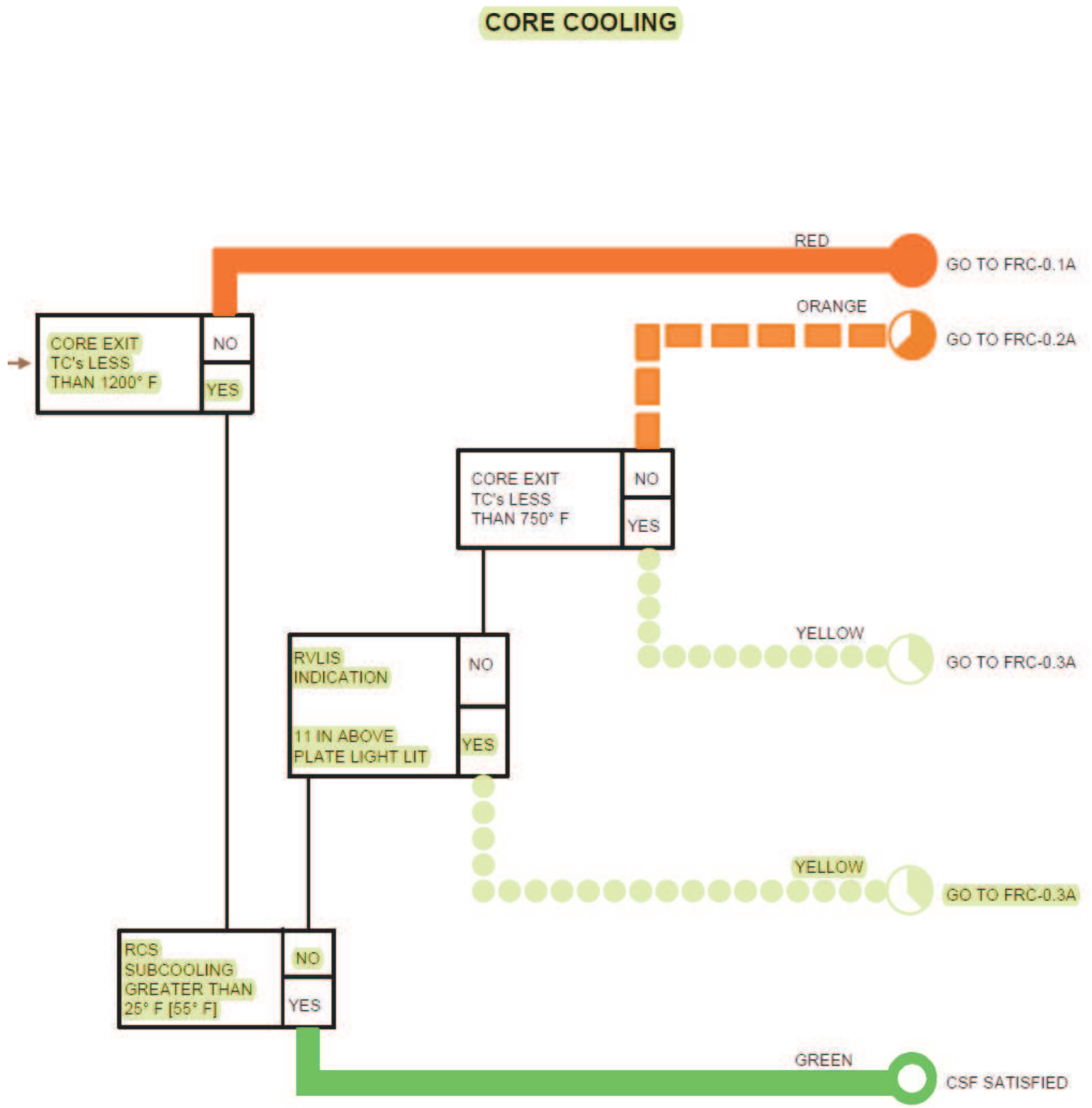
Comments / Reference: From FRP-0.2A, CSFST	Revision # 8



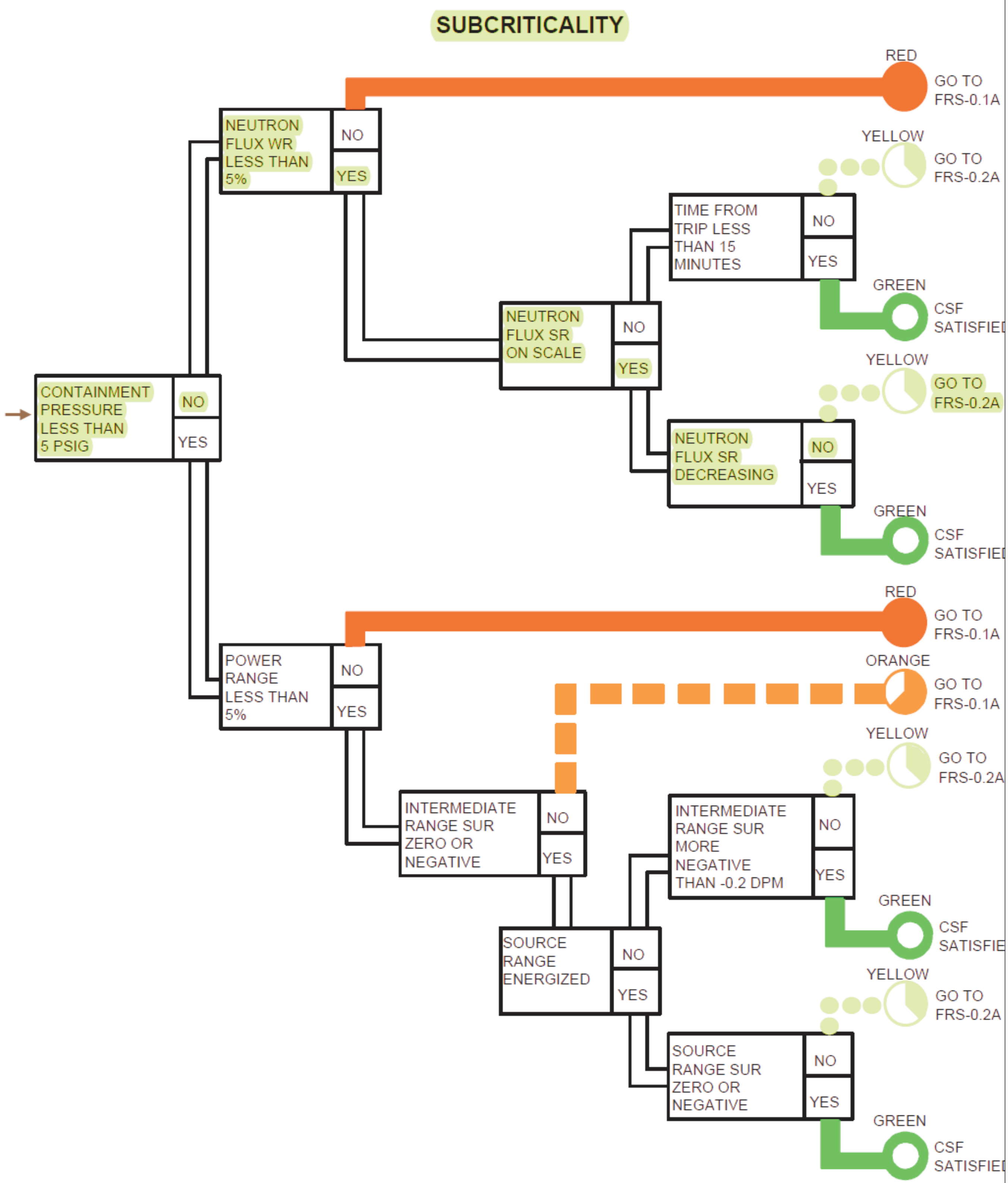
COLD OVER PRESSURE LIMIT

RCS TEMPERATURE	RCS PRESSURE
70	375
150	375
200	447
220	447
250	573
350	573

Comments / Reference: From FRC-0.3A, CSFST	Revision # 8



Comments / Reference: From FRS-0.2A, CSFST	Revision # 8



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	028 A2.03	
	Importance Rating		4.0

Hydrogen Recombiner and Purge Control System: Ability to (a) predict the impacts of the following malfunctions or operations on the HRPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: The hydrogen air concentration in excess of limit flame propagation or detonation with resulting equipment damage in containment

Proposed Question: SRO 91

Given the following conditions:

- A Loss of Coolant Accident has occurred on Unit 2.
- The crew is performing actions of EOP-1.0B, Loss of Reactor or Secondary Coolant, after transition from FRC-0.1B, Response to Inadequate Core Cooling.
- Containment Hydrogen concentration is 2.8% and rising slowly.
- Containment pressure is 4 psig and rising.

Which of the following describes the adverse impact to Containment and the procedural actions implemented to mitigate the impact?

Potential breach of Containment fission barrier due to…

A. …hydrogen combustion.
Reduce Containment hydrogen concentration per SOP-205, Hydrogen Purge Supply and Exhaust System.

B. …internal pressure.
Reduce Containment internal pressure per SOP-801B, Containment Ventilation System.

C. …hydrogen combustion.
Reduce Containment internal pressure per SOP-801B, Containment Ventilation System.

D. …internal pressure.
Reduce Containment hydrogen concentration per SOP-205, Hydrogen Purge Supply and Exhaust System.

Proposed Answer: A

Explanation:

A. Correct. Hydrogen combustion is the problem and use of the Hydrogen Purge Supply and Exhaust System will be directed to lower H2 concentration.
B. Incorrect. Plausible because pressure could impact Containment barrier.
C. Incorrect. Plausible due to hydrogen combustion but the Containment Purge Supply and Exhaust System would not be used.
D. Incorrect. Plausible because pressure could impact Containment barrier.

Technical Reference(s): EOP-1.0B, Step 17 Bases — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** the operator actions, including all cautions, notes, RNOs and bases associated with EOP-1.0, Loss of Reactor or Secondary Coolant.

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

Question History: Last NRC Exam

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

10 CFR Part 55 Content:
55.41
55.43 1, 5

Comments / Reference: From EOP-1.0B, Step 17 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-1.0B
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 36 OF 44

ATTACHMENT 4
PAGE 13 OF 21

BASES

Since SI is on, procedure FRI-0.3B is not applicable at this time, but some of the considerations for head venting may be (e.g., hydrogen concentration in containment).

STEP 17: This step instructs the operator to obtain a current hydrogen concentration measurement. Following a design basis accident, hydrogen gas may be generated inside the Containment by reactions such as zirconium metal with water, corrosion of materials of construction, exposure of the organic cable materials to radiation and radiolysis of aqueous solution in the core and sump. A combustible mixture can be formed when hydrogen gas concentration in the Containment atmosphere is greater than 4.0% volume.

If inadequate core cooling has occurred, the containment hydrogen concentration may be as much as 10 to 12 volume percent, depending on the amount of metal-water reaction (to produce hydrogen) that has occurred in the core. The operator is instructed to obtain a current containment hydrogen concentration measurement at this point in order to determine the concentration of combustible gases in the containment. Note that in order to have the potential for flammable hydrogen concentrations, an inadequate core cooling situation must have already existed. Without an inadequate core cooling situation, sufficient hydrogen would not be expected to have been produced to cause potentially flammable mixtures. If the containment atmosphere hydrogen mixture is less than 0.5% volume, a hydrogen burn is not possible. If the containment atmosphere hydrogen mixture is greater than or equal to 0.5% volume, the Plant Staff is notified to evaluate Unit conditions to determine if additional recovery action is necessary, while the operator proceeds with this procedure.

CPSES containment design includes a large volume, dry containment with a design pressure of 50 psig. Analyses performed to determine the pressure in a dry containment resulting from the combustion of hydrogen corresponding to a 75% metal-water reaction, following onset of a degraded core accident and while the containment was still near its peak pressure, indicated that the peak total containment pressure was below the failure pressure. Furthermore, analyses indicated that essential equipment would function during and after a large deflagration in a dry containment. This conclusion is supported by the TMI-2 event.

Comments / Reference: From EOP-1.0B, Step 17 Bases	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOP-1.0B
LOSS OF REACTOR OR SECONDARY COOLANT	REVISION NO. 8	PAGE 37 OF 44

ATTACHMENT 4
PAGE 14 OF 21

BASES

The Main Control Board hydrogen indicators provide percent hydrogen in wet air; however, all hydrogen measurements are referenced to concentrations in dry air even though the actual containment environment may contain significant steam concentrations. The reason for this is twofold: 1) the hydrogen microprocessor removes moisture from the sample thus approximating a dry air condition and 2) the indication of the potential of hydrogen flammability is conservative when based upon using hydrogen concentration in dry air.

If a hydrogen concentration value can not be obtained from the hydrogen monitoring system, a grab sample from the containment PIG radiation monitor may be used to determine the hydrogen concentration value required for this step. The ability to obtain the sample depends on containment atmosphere activity conditions, and the resulting radiation exposure that may be incurred while collecting and analyzing the sample. If radiological conditions prohibit obtaining a sample, Chemistry will notify control room personnel of the inability to obtain the sample and the plant staff should be notified to determine the appropriate action to support the accident recovery sequence. An evaluation of the projected hydrogen concentration may be performed by engineering to determine the appropriate action for this step. If it is determined that a sample is required to support recovery actions, then the Contingency Sampling Plan per CHM-111, Primary Chemistry Accident Assessment Sampling Program may be initiated with plant staff approval. Use of the Contingency Sampling Plan should be evaluated against the requirements of 10CFR50.54x prior to use.

Comments / Reference: From CPNPP Exam Bank	Revision # 03/12/07

Given the following conditions:

- A Loss of Coolant Accident has occurred on Unit 2.
- The crew is performing actions of EOP-1.0B, Loss of Reactor or Secondary Coolant, after transition from FRC-0.2B, Response to Degraded Core Cooling.
- Containment Hydrogen concentration is 2.8% and rising slowly.
- Use of the Electric Hydrogen Recombiners is being considered.

Which of the following describes the requirement for operation of the Hydrogen Recombiners for this condition?

A. Recombiners are required to maintain Containment hydrogen concentration below design basis. Place at least ONE (1) recombiner in service in accordance with OP-206A, Electric Hydrogen Recombiner System.

B. Recombiners are required to maintain Containment hydrogen concentration below design basis. Notify Plant Engineering to determine if it is safe to place a recombiner in service at the current containment hydrogen concentration. Continue in EOP-1.0B

C. Recombiners are NOT required to maintain Containment hydrogen concentration below design basis.
Use of a recombiner is prohibited at the current containment hydrogen concentration level. Continue in EOP-1.0B.

D. Recombiners are NOT required to maintain Containment hydrogen concentration below design basis.
Notify Plant Engineering to determine if it is safe to place a recombiner in service at the current containment hydrogen concentration, while continuing in EOP-1.0B.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	056 A2.04	
	Importance Rating		2.8

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

Proposed Question: SRO 92

Given the following conditions:

- A plant startup is in progress on Unit 1 with the Reactor currently at 8% power.
- Main Feed Water Pump 1-01 and Condensate Pumps 1-01 and 1-02 are operating.
- Main Feed Water Pump 1-02 is on the turning gear.
- 1-LV-2211/12, CNDS REJ LVL CTRL VLV, is open due to a high Condenser Hotwell level when Condensate Pump 1-01 trips.

Which of the following identifies the effect of the Condensate Pump trip on the Condensate System and what procedural actions should be taken to mitigate the situation?

A. 1.) 1-PV-2286, LP FW HTR BYP VLV automatically opens and 1-LV-2211/12, CNDS REJ LVL CTRL VLV automatically closes.
2.) Perform actions of ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, Section 3, Condensate Pump Trip.

B. 1.) 1-PV-2286, LP FW HTR BYP VLV automatically opens and 1-LV-2211/12, CNDS REJ LVL CTRL VLV remains open.
2.) Perform actions of ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, Section 7, LP Heater Bypass Valve Opening At Power.

C. 1.) 1-PV-2286, LP FW HTR BYP VLV remains closed and 1-LV-2211/12, CNDS REJ LVL CTRL VLV automatically closes.
2.) Perform actions of ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, Section 3, Condensate Pump Trip.

D. 1.) 1-PV-2286, LP FW HTR BYP VLV remains closed and 1-LV-2211/12, CNDS REJ LVL CTRL VLV remains open.
2.) Perform actions of ABN-302, Feedwater, Condensate, Heater Drain System Malfunction, Section 3, Condensate Pump Trip.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because normally PV-2286 would open and LV-2211/12 would close, but with generator output below 15%, PV-2286 will not open. Response is per ABN-302, Section 3, Condensate Pump Trip.

B. Incorrect. Plausible since normally PV-2286 would open and LV-2211/12 does close, but with generator output below 15%, PV-2286 does not open. Section 7 of ABN-302 addresses PV-2286 opening at power, but the valve will not open and only Section 3 need be addressed.

C. Incorrect. Plausible since on a loss of both Condensate Pumps, PV-2286 automatically opens but ONLY if power level is above 15%. Since power is below 15%, the valve remains closed. LV-2211/12 will automatically close if either PV-2286 opens or BOTH Condensate Pumps trip, however, as PV-2286 remains closed and only one Condensate Pump trips, the valve should remain open.

D. Correct. As power is below 15%, PV-2286 will not automatically open. As PV-2286 remains closed and only one Condensate Pump trips, LV-2211/12 should remain open. Response is per ABN-302, Section 3, Condensate Pump Trip.

Technical Reference(s) ABN-302, Steps 3.2 & 7.0 | ALM-0082A, 1-ALB-8B, Window 3.8 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the response to HS-2286, LP HTR BYPASS VLV opening at power per ABN-302, Feedwater, Condensate, Heater Drains System Malfunction**.**

Question Source:
- Bank # X
- 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 ABN-302, Steps 3.2 & 7.0	Revision # 14

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-302
FEEDWATER, CONDENSATE, HEATER DRAIN SYSTEM MALFUNCTION	REVISION NO. 14	PAGE 10 OF 78

3.2 Automatic Actions

a. Turbine runback at 35% per minute to 60% power.

> NOTE: Opening the LP FW HTR BYP VLV will reduce unit efficiency resulting in an increased mismatch between reactor and turbine power.

b. A low feedwater pump suction pressure 250 psig and greater than 15% turbine power will open u-PV-2286 (to bypass the low pressure heater strings) and close the condensate reject and recirc valves (to provide maximum condensate pressure at the feedwater pump suction).

- u-HS-2286, LP FW HTR BYP VLV
- u-HS-2211/12, CNDS REJECT VLV
- u-ZL-2239, CNDS PMP RECIRC VLV

c. A low feedwater pump suction pressure ≈280 psig will open u-PV-2242, Uu CNDS POL FILT BYP PRESS CTRL VLV.

d. A trip of a condensate pump at high power has a high probability of losing both Main Feed Water Pumps resulting in a Plant Trip.

Comments / Reference: From ALM-0082A, 1-ALB-8B, Window 3.8	Revision # 8

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0082A
ALARM PROCEDURE 1-ALB-8B	REVISION NO. 8	PAGE 105 OF 157

ANNUNCIATOR NOM./NO.: **CNDS LP HTR BYP VLV OPEN PV-2286** **3.8**

PROBABLE CAUSE:

Blown control power fuse
Two of three low feed pump suction pressure (250 psig) AND turbine power $\geq$ 15%

NOTE: 1-HS-2286, LP FW HTR BYP VLV fails open on loss of air or power. If handswitch light indication is off, a blown control power fuse is probable cause of valve opening.

AUTOMATIC ACTIONS:

1-HS-2211/12, CNDS REJECT VLV and 1-ZL-2239, CNDS PMP RECIRC VLV close.

NOTE:

- The Main Feedwater Pumps trip on low suction pressure at approximately 190 psig with a 30 second delay for pump 1-01 and a 45 second delay for pump 1-02 and 170 psig with a 4 second delay.
- 1-PV-2242, U1 CNDS POL FILT BYP PRESS CTRL VLV will open if 2 of 3 FWP suction pressures are $\leq$ 280 psig.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	071 G 2.4.35	
	Importance Rating		4.0

Waste Gas Disposal System: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects

Proposed Question: SRO 93

Given the following conditions:

- Waste Gas System Decay Tank #1 release is in progress.
- The following alarms are received simultaneously:
  - PC-11 HIGH alarm for X-RE-5701 AUX BLDG VENT DUCT (ABV089).
  - 1-ALB-6B, Window 3.7 – GWPS PNL TRBL.
- The Radwaste Operator reports the following alarm:
  - Gaseous Waste Panel, Window 1.8 – AUX BLDG VENT EXHAUST MONITOR HIGH RAD.

Which of the following is the most likely cause and what action is required?

A. Release permit setpoints for Waste Gas System Decay Tank #1 have been exceeded.
Enter ABN-902, Accidental Release of Radioactive Gas, and direct the Rad Waste Operator to ensure X-HCV-0014, Waste Gas Discharge Control Valve, is closed.

B. The in-service Waste Gas Decay Tank Relief Valve is lifting.
Enter RWS-201, Gaseous Waste Processing System, and isolate the in-service Waste Gas Decay Tank.

C. The in-service Waste Gas Decay Tank Relief Valve is lifting.
Enter ABN-902, Accidental Release of Radioactive Gas, and ensure Emergency Recirculation Initiation has occurred.

D. Release permit setpoints for Waste Gas System Decay Tank #1 have been exceeded.
Enter RWS-201, Gaseous Waste Processing System, and isolate Waste Gas System Decay Tank #1.

Proposed Answer: A

Explanation:

A. Correct. Given the conditions listed, these are the correct actions and procedure entry required.
B. Incorrect. Plausible because the Waste Gas Decay Tank Relief Valve could be lifting, however, this discharge is directed to the Waste Gas Holdup Tank and annunciator ALM-0401, Window 1.8 would not be an alarm.
C. Incorrect. Plausible because the procedure entry is correct and the Waste Gas Decay Tank Relief Valve could be lifting, however, this action would not be required for the conditions listed.
D. Incorrect. Plausible because release setpoints have been exceeded, however, ABN entry is required prior to performing actions to isolate the Waste Gas System Decay Tank.

Technical Reference(s) ALM-0062A, 1-ALB-6B, Window 3.7
ABN-902, Step 2.3.1.b
ALM-0401, Window 1.8

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** how the Gaseous Waste Processing System Main Control Board/Plant Computer controls, alarms and indications are used to predict, monitor and control changes in the system.

Question Source: Bank # X
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 4, 5

Comments / Reference: From ALM-0062A, 1-ALB-6B, Window 3.7	Revision # 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0062A
ALARM PROCEDURE 1-ALB-6B	REVISION NO.6	PAGE 52 OF 70

ANNUNCIATOR NOM./NO.: **GWPS PNL TRBL** **3.7**

PROBABLE CAUSE:

Any alarm on the gaseous waste panel

AUTOMATIC ACTIONS: None

NOTE: Several automatic actions may be initiated at the individual local alarm setpoints. The operator responding to the local panel alarm will initiate appropriate response to these conditions.

OPERATOR ACTIONS:

1. Coordinate with Unit 2 and dispatch a Radwaste operator to the Gaseous Waste Panel to determine and correct cause of alarm condition per ALM-0401.
   A. If a high radiation alarm occurs on X-RE-5250 (WSG083) WASTE GAS, place a standby gas decay tank in service per RWS-201.

2. Correct the condition or initiate a work request per STA-606.

Comments / Reference: From ABN-902, Step 2.3.1.b	Revision # 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. ABN-902
RELEASE OF RADIOACTIVE/TOXIC GAS	REVISION NO. 7	PAGE 5 OF 23

2.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE: Unit 1 typically handles response for common alarms (e.g. vent stack). However, Unit 2 should be informed to check the ABN to ensure applicable steps addressed for that unit.

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1 Verify applicable Automatic Action has occurred with related alarm:	
☐ a. Verify Containment air radiation alarms - CLEAR: • CAPu98 (u-RE-5502), CNTMT AIR PIG PART • CAGu97 (u-RE-5503), CNTMT AIR PIG GAS	a. Manually ensure Containment Ventilation Isolation per Attachment 1.
☐ b. Verify the following radiation alarms - CLEAR: • PVF684 (X-RE-5570A), S. WRGM EFFLUENT • PVF685 (X-RE-5570B), N. WRGM EFFLUENT • ABV089 (X-RE-5701), AUX BLDG VENT DUCT	b. AT X-GP-01, GWPS WASTE GAS PROCESS CONTROL PANEL (AB 862 Rm X-243) ensure X-HS-0014, WASTE GAS DISCHARGE CONTROL VALVE - CLOSED.

Comments / Reference: From ALM-0401, Window 1.8	Revision # 5

CPNPP ALARM PROCEDURES MANUAL	COMMON	PROCEDURE NO. ALM-0401
ALARM PROCEDURE GASEOUS WASTE PANEL	REVISION NO. 5	PAGE 23 OF 75

ANNUNCIATOR NOM./NO.: **AUX BLDG VENT EXHAUST MONITOR HIGH RAD** **1.8**

PROBABLE CAUSE:

Excessive flow rate during release
X-RE-5701 Operating Failure

AUTOMATIC ACTIONS:

X-HCV-0014, GWPS DISCH TO PLT EXH PLNM ISOL VLV closes (WHITE TRIP LIGHT ENERGIZED)

OPERATOR ACTIONS:

1. IF GWPS discharge is in progress,
   THEN
   PERFORM the following:
   A. ENSURE X-HS-0014, WASTE GAS DISCH CONTROL VALVE is CLOSED.
   B. CLOSE XGH-7898-RO, GWPS H2/N2 TO PLT VENT EXH PLNM SPLY DNSTRM ISOL VLV.
   C. NOTIFY the Radwaste Supervisor, the Control Room AND Radiation Protection of a possible Discharge Permit violation
      AND
      REFER to ABN-902.
   D. SECURE the discharge per RWS-201.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		
K/A #	G 2.1.45	
Importance Rating		4.3

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indication

Proposed Question: SRO 94

Given the following conditions:

- Unit 1 is at 100% power
- The following annunciators are in alarm:
  - 1-ALB-8A, Window 2.7 – MSL 2 1 OF 3 PRESS LO.
  - 1-ALB-8A, Window 2.8 – SG 2 STM & FW FLO MISMATCH.
  - 1-ALB-8A, Window 2.12 – SG 2 LVL DEV.
- Plant indications are as follows:
  - 1-FI-522A, SG 2 STM FLO, is 3.8×10^6 lbm/hr and steady.
  - 1-FI-523A, SG 2 STM FLO, is 0 lbm/hr and stable.
  - 1-PI-524A, MSL 2 PRESS CHAN I, is 1000 psig and stable.
  - 1-PI-525A, MSL 2 PRESS CHAN II, is 0 psig and stable.
  - 1-LT-529A, SG 2 LVL (NR) CHAN I, is 64% and lowering.
  - 1-FI-520A, SG 2 FW FLO, is 1.5×10^6 lbm/hr and stable.

Which of the following identifies the procedure entry necessary to mitigate the situation?

A. ABN-707, Steam Flow Instrument Malfunction.

B. ABN-709, Steam Line Pressure, Steam Header Pressure, Turbine 1st Stage Pressure, And Feed Header Pressure Instrument Malfunction.

C. ABN-708, Feedwater Flow Instrument Malfunction.

D. ABN-710, Steam Generator Level Instrument Malfunction.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because listed indications would seem to verify that a Steam Flow Instrument has failed with subsequent ABN-707 entry, however, the failure of a Steam Line Pressure Instrument will affect the associated Steam Flow Pressure Compensation and provide the indications listed.
B. Correct. Given the indications, a steam line pressure instrument has failed and entry into ABN-709 is required.
C. Incorrect. Plausible because there is a Steam Flow/Feed Flow mismatch, however, it is the Steam Pressure Instrument that has failed.
D. Incorrect. Plausible because Steam Generator level is lowering and ABN-710 entry would be required, however, it is due to the system response of a failed Steam Line Pressure Instrument.

Technical Reference(s): ABN-709, Steps 2.1 & 2.2; ABN-708, Steps 2.1 & 2.2; ABN-707, Steps 2.1 & 2.2; ABN-710, Steps 2.1 & 2.2; IPO-001A, Attachment 4

Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **EXPLAIN** the instrumentation and controls of the Main Steam System and **PREDICT** the system response.

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

Question History: Last NRC Exam CPNPP 2012

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

10 CFR Part 55 Content:
55.41
55.43 5

Comments / Reference: From ABN-709, Step 2.1	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-709
STM LINE, STM HDR & TURB 1st STAGE PRESS. & FEED HDR PRESS. INSTR MALFUNCTION	REVISION NO. 8	PAGE 3 OF 33

2.0 STEAM LINE PRESSURE INSTRUMENT MALFUNCTION

2.1 Symptoms

a. Annunciator Alarms

- MSL 1 1 OF 3 PRESS LO (8A-1.7)
- MSL 2 1 OF 3 PRESS LO (8A-2.7)
- MSL 3 1 OF 3 PRESS LO (8A-3.7)
- MSL 4 1 OF 3 PRESS LO (8A-4.7)

- SG 1 STM & FW FLO MISMATCH (8A-1.8)
- SG 2 STM & FW FLO MISMATCH (8A-2.8)
- SG 3 STM & FW FLO MISMATCH (8A-3.8)
- SG 4 STM & FW FLO MISMATCH (8A-4.8)

- ANY SG PRESS RATE HI (8A-4.10)

- SG 1 1 OF 3 PRESS RATE HI (8A-1.16)
- SG 2 1 OF 3 PRESS RATE HI (8A-2.16)
- SG 3 1 OF 3 PRESS RATE HI (8A-3.16)
- SG 4 1 OF 3 PRESS RATE HI (8A-4.16)

b. Plant Indications

- Steam line pressure channel indicating higher or lower than the other three channels in the same steam line.

1) u-PI-515A, MSL 1 PRESS CHAN II
u-PI-516A, MSL 1 PRESS CHAN IV
u-PI-514A, MSL 1 PRESS CHAN I
u-PI-2325, MSL 1 PRESS

2) u-PI-524A, MSL 2 PRESS CHAN I
u-PI-526A, MSL 2 PRESS CHAN III
u-PI-525A, MSL 2 PRESS CHAN II
u-PI-2326, MSL 2 PRESS

3) u-PI-535A, MSL 3 PRESS CHAN II
u-PI-534A, MSL 3 PRESS CHAN I
u-PI-536A, MSL 3 PRESS CHAN III
u-PI-2327, MSL 3 PRESS

4) u-PI-544A, MSL 4 PRESS CHAN I
u-PI-545A, MSL 4 PRESS CHAN II
u-PI-546A, MSL 4 PRESS CHAN IV
u-PI-2328, MSL 4 PRESS

Comments / Reference: From ABN-709, Step 2.2	Revision # 8

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-709
STM LINE, STM HDR & TURB 1st STAGE PRESS. & FEED HDR PRESS. INSTR MALFUNCTION	REVISION NO. 8	PAGE 4 OF 33

2.2 Automatic Actions

NOTE:
- Control responses will only occur if failure is in channel selected for control.
- A Steam Generator Automatic Relief Valve will fail open only if associated pressure channel (u-PT-2325, 2326, 2327, or 2328) fails high.

a. Steam line channel failure high will cause feedwater flow to increase, feedwater pump speed to increase or Steam Generator Atmospheric Relief Valves to open.

b. Steam line channel failure low will cause feedwater flow to decrease and feedwater pumps to decrease in speed.

Comments / Reference: From ABN-708, Steps 2.1 & 2.2	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-708
FEEDWATER FLOW INSTRUMENT MALFUNCTION	REVISION NO. 6	PAGE 3 OF 6

2.0 Feedwater Flow Instrument Malfunction

2.1 Symptoms

a. Annunciator Alarms

- SG 1 STM & FW FLO MISMATCH (8A-1.8)
- SG 2 STM & FW FLO MISMATCH (8A-2.8)
- SG 3 STM & FW FLO MISMATCH (8A-3.8)
- SG 4 STM & FW FLO MISMATCH (8A-4.8)
- SG 1 LVL DEV (8A-1.12)
- SG 2 LVL DEV (8A-2.12)
- SG 3 LVL DEV (8A-3.12)
- SG 4 LVL DEV (8A-4.12)

b. Plant Indications

1) One feedwater line flow channel indicating higher or lower than the other.

- u-FI-510A, SG 1 FW FLO
- u-FI-511A, SG 1 FW FLO
- u-FI-520A, SG 2 FW FLO
- u-FI-521A, SG 2 FW FLO
- u-FI-530A, SG 3 FW FLO
- u-FI-531A, SG 3 FW FLO
- u-FI-540A, SG 4 FW FLO
- u-FI-541A, SG 4 FW FLO

2.2 Automatic Actions

NOTE: Control responses will only occur if failure is in channel selected for control.

a. Feedwater flow channel failing HIGH will cause feedwater flow to decrease by closing feedwater control valve. Without operator action, reactor will trip at 38%(35.4%) SG level.

b. Feedwater flow channel failing LOW will cause feedwater flow to increase by opening feedwater control valve. Without operator action, turbine will trip at 84%(81.5%) SG level.

Comments / Reference: From ABN-707, Step 2.1	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-707
STEAM FLOW INSTRUMENT MALFUNCTION	REVISION NO. 6	PAGE 3 OF 9

2.0 <u>Steam Flow Instrument Malfunction</u>

2.1 Symptoms

a. Annunciator Alarms

- MSL 1 1 OF 3 PRESS LO (8A-1.7)
- MSL 2 1 OF 3 PRESS LO (8A-2.7)
- MSL 3 1 OF 3 PRESS LO (8A-3.7)
- MSL 4 1 OF 3 PRESS LO (8A-4.7)

- SG 1 STM & FW FLO MISMATCH (8A-1.8)
- SG 2 STM & FW FLO MISMATCH (8A-2.8)
- SG 3 STM & FW FLO MISMATCH (8A-3.8)
- SG 4 STM & FW FLO MISMATCH (8A-4.8)

- SG 1 LVL DEV (8A-1.12)
- SG 2 LVL DEV (8A-2.12)
- SG 3 LVL DEV (8A-3.12)
- SG 4 LVL DEV (8A-4.12)

b. Plant Indications

- One steam line flow higher or lower than the others
  - <u>u</u>-FI-512A, SG 1 STM FLO
  - <u>u</u>-FI-513A, SG 1 STM FLO

  - <u>u</u>-FI-522A, SG 2 STM FLO
  - <u>u</u>-FI-523A, SG 2 STM FLO

Comments / Reference: From ABN-707, Step 2.2	Revision # 6

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-707
STEAM FLOW INSTRUMENT MALFUNCTION	REVISION NO. 6	PAGE 4 OF 9

2.1 b. • One steam line pressure higher or lower than the others.

- u-PI-514A, MSL 1 PRESS CHAN I
- u-PI-515A, MSL 1 PRESS CHAN II
- u-PI-516A, MSL 1 PRESS CHAN IV
- u-PI-2325, MSL 1 PRESS

- u-PI-524A, MSL 2 PRESS CHAN I
- u-PI-525A, MSL 2 PRESS CHAN II
- u-PI-526A, MSL 2 PRESS CHAN III
- u-PI-2326, MSL 2 PRESS

- u-PI-534A, MSL 3 PRESS CHAN I
- u-PI-535A, MSL 3 PRESS CHAN II
- u-PI-536A, MSL 3 PRESS CHAN III
- u-PI-2327, MSL 3 PRESS

- u-PI-544A, MSL 4 PRESS CHAN I
- u-PI-545A, MSL 4 PRESS CHAN II
- u-PI-546A, MSL 4 PRESS CHAN IV
- u-PI-2328, MSL 4 PRESS

• Annunciator alarms on steam and feedwater flow mismatch and narrow range steam generator level at the same time could indicate a common instrument line failure. (See Attachment 2).

2.2 Automatic Actions

a. Failed channel selected for control

- Steam flow channel failing HIGH (steam flow failed high OR pressure compensation failed high) will cause feedwater flow to increase and feedwater pump speed to increase due to larger programmed Δp between feedwater pressure and steam pressure. Without operator action, turbine will trip at 84%(81.5%) SG level.

- Steam flow channel failing LOW (steam flow failed low OR pressure compensation failed low) will cause feedwater flow to decrease and feedwater pump speed to decrease to achieve a smaller Δp between feedwater pressure and steam pressure. Without operator action, reactor will trip at 38%(35.4%) steam generator level.

Comments / Reference: From ABN-710, Step 2.1	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-710
STEAM GENERATOR LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 3 OF 29

2.0 STEAM GENERATOR LEVEL INSTRUMENTATION MALFUNCTION

2.1 Symptoms

a. Annunciator Alarms

- SG 1 LVL LO (8A-1.6)
- SG 2 LVL LO (8A-2.6)
- SG 3 LVL LO (8A-3.6)
- SG 4 LVL LO (8A-4.6)

- SG 1 LVL DEV (8A-1.12)
- SG 2 LVL DEV (8A-2.12)
- SG 3 LVL DEV (8A-3.12)
- SG 4 LVL DEV (8A-4.12)

Comments / Reference: From ABN-710, Step 2.2	Revision # 10

CPSES ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-710
STEAM GENERATOR LEVEL INSTRUMENTATION MALFUNCTION	REVISION NO. 10	PAGE 4 OF 29

2.2 Automatic Actions

NOTE: Control responses will occur only if the failed channel is selected for control.

- Steam Generator Level Channel failing HIGH will cause the feedwater control valve to CLOSE, thereby decreasing steam generator level.
  (Unit 1 LO LEVEL REACTOR TRIP at 38%)
  (Unit 2 LO LEVEL REACTOR TRIP at 35.4%)

- Steam Generator Level Channel failing LOW will cause the feedwater control valve to OPEN, thereby increasing steam generator level.
  (Unit 1 HI LEVEL TURBINE TRIP at 84%)
  (Unit 2 HI LEVEL TURBINE TRIP at 81.5%)

Comments / Reference: From IPO-001A, Attachment 4	Revision # 21

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-001A
PLANT HEATUP FROM COLD SHUTDOWN TO HOT STANDBY	REVISION NO. 21	PAGE 66 OF 93

ATTACHMENT 4
PAGE 5 OF 7

EQUIPMENT CONDITION CHECKLIST

B. This checklist serves to document that a review of the Control Board status, for equipment NOT normally aligned by SOP or OPT lineups prior to MODE change, has been performed and that the equipment is in an acceptable status for plant heatup. Where equipment is found in other than the normal configuration due to clearances, testing in progress, etc., the "as found" position or condition should be indicated on the data sheet.

Where a normal position is listed as CLOSE or OPEN, this refers to the actual position of the control switch. Normal switch positions listed as AUTO (OPEN) or AUTO (CLOSED) refer to the control switch position and the indicated valve position. A blank in the POSITION column indicates that position is determined by Shift Manager and/or plant conditions. Pot settings are provided to ensure controllers are set per the TDM, the intent of the pot setting position is to check for the proper setting, NOT whether the controller is in AUTO or MANUAL.

BOARD	COMPONENT	NOMENCLATURE	POSITION/ POT SETTING	*AS FOUND	INITIALS
CB-08	43/1-SDA	STM DMP INTLK SELECT	OFF/RESET	_____	_____
CB-08	43/1-SDB	STM DMP INTLK SELECT	OFF/RESET	_____	_____
CB-08	43/1-SD	STM DMP MODE SELECT	STM PRESS	_____	_____
CB-09	1-FS-512C	SG 1 STM FLO CHAN SELECT	FY-512B	_____	_____
CB-09	1-FS-510C	SG 1 FW FLO CHAN SELECT	FY-510B	_____	_____
CB-09	1-LS-519C	SG 1 LVL CHAN SELECT	LQY-551	_____	_____
CB-09	1-FS-522C	SG 2 STM FLO CHAN SELECT	FY-523B	_____	_____
CB-09	1-FS-520C	SG 2 FW FLO CHAN SELECT	FY-520B	_____	_____
CB-09	1-LS-529C	SG 2 LVL CHAN SELECT	LQY-552	_____	_____

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		
K/A #	G 2.2.7	
Importance Rating		3.6

Equipment Control: Knowledge of the process for conducting special or infrequent tests

Proposed Question: SRO 95

Which of the following individuals is expected to be in charge of High Risk, Infrequent Evolution, or Heightened Level of Awareness activities conducted on watch?

A. Plant Manager

B. Director, Nuclear Operations

C. Shift Operations Manager

D. Unit Supervisor

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because consideration is given to assigning this individual with the authority and experience to exercise continuous responsibility for the oversight of a particular test or evolution, however, it is the Unit Supervisor who acts as the SRO in charge.

B. Incorrect. Plausible because the Director, Nuclear Operations normally holds a SRO license, however, they are not in charge of these activities.

C. Incorrect. Plausible because the Shift Operations Manager has management responsibilities at the Station, however, it is the Unit Supervisor who is the SRO in charge.

D. Correct. As prescribed in OWI-107.

Technical Reference(s) OWI-107, Step 6.2.3.F — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DESCRIBE** the responsibilities of and **ASSUME** Control Room command function.

Question Source: Bank # X

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 5

Comments / Reference: From OWI-107, Step 6.2.3.F	Revision # 8

CPNPP OPERATIONS DEPARTMENT WORK INSTRUCTIONS		PROCEDURE NO. OWI-107
OPERATIONS DEPARTMENT TURNOVER AND BRIEFING INSTRUCTIONS	REVISION NO. 8 INFORMATION USE	PAGE 12 OF 15

6.2.3 Pre-job Briefs - High Risk, Heightened Level of Awareness, Infrequent Evolution

A. Classification of activities as High Risk, Heightened Level of Awareness or Infrequent Evolutions should be done by the WC SRO or the SM. During absence of the SM, activity classification may be performed by his/her designee. These activities should normally be preplanned and scheduled as part of the Plan of the Day (POD).

If a High Risk activity is identified due to a potential degradation of nuclear safety, the evolution should be reviewed against the IPTE criteria per STA-122.

B. An activity classified as a High Risk, Heightened Level of Awareness or Infrequent Evolution requires Management Observation per STA-122, Attachment 8.B.

C. The designated performer of a Management Observation for any High Risk, Heightened Level of Awareness or Infrequent Evolution is a manager, or designee for the work group controlling the activity (e.g., Shift Manager or designee for Operations Department led activity, Maintenance Manager or designee for Maintenance Department led activity, observer designated by Plant Manager, Operations, Director, etc.).

D. A formal briefing shall be held for all High Risk, Heightened Level of Awareness and Infrequent Evolutions.

E. The Shift Manager should be present at any High Risk briefing. If the activity involves a maintenance activity, a Maintenance Manager should also be present at any High Risk briefing.

F. The Unit supervisor is expected to be the SRO in charge of High Risk, Heightened Level of Awareness, and Infrequent Evolutions and should give full attention to the activity.

- Activities that can distract the operators or US should be avoided, or an extra SRO should be assigned. The Extra SRO should monitor routine activities for the unit while the Unit Supervisor is involved in the special activity. This responsibility may be reversed at the Shift Manager's discretion.

- If necessary, the SM should determine if the FSS or a Maintenance Supervisor should provide observation of the HRA, HLA or IFE activity in the plant.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		
	K/A #	G 2.2.35	
	Importance Rating		4.5

Equipment Control: Ability to determine Technical Specification Mode of Operation

Proposed Question: SRO 96

Technical Specification Limiting Condition for Operation (LCO) 3.0.4 allows a MODE change under which of the following situations?

A. The MODE change is from MODE 3 to MODE 4.

B. The MODE change is from MODE 5 to MODE 4.

C. The COMPLETION TIME of the REQUIRED ACTION is 30 days.

D. The inoperability will be restored prior to the COMPLETION TIME.

Proposed Answer: A

Explanation:

A. Correct. Per Technical Specification LCO 3.0.4.
B. Incorrect. Plausible because a MODE change is being performed, however, it is not associated with a Plant Shutdown as required in LCO 3.0.4.
C. Incorrect. Plausible if thought that LCO 3.0.4 allows this condition, however, the only allowance in the Applicability is for an unlimited period of time.
D. Incorrect. Plausible because inoperability of a component is addressed in LCO 3.0.4, however, a mode change under this condition must include risk assessment evaluations.

Technical Reference(s) Technical Specification LCO 3.0.4 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:

Question Source:	Bank #	X	
	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	2	

Comments / Reference: From Technical Specification LCO 3.0.4	Amendment # 150

LCO Applicability
3.0

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

LCO 3.0.1 LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2 and LCO 3.0.7.

LCO 3.0.2 Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5 and LCO 3.0.6.

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

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

a. MODE 3 within 7 hours;

b. MODE 4 within 13 hours; and

c. MODE 5 within 37 hours.

Exceptions to this Specification are stated in the individual Specifications.

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

LCO 3.0.3 is only applicable in MODES 1, 2, 3, and 4.

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

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

Comments / Reference: From Technical Specification LCO 3.0.4	Amendment # 150

LCO Applicability
3.0

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

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

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

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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		
K/A #	G 2.3.5	
Importance Rating		2.9

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO 97

Given the following conditions:

- Unit 1 has experienced a manual Reactor Trip and Safety Injection initiation based on lowering Reactor Coolant System (RCS) pressure and lowering Pressurizer level.
- EOP-0.0A, Reactor Trip and Safety Injection, is in progress.
- The following parameters are noted during diagnostics:
  - All three Pressurizer Safety Valves are CLOSED.
  - Both Pressurizer Spray Valves are CLOSED.
  - Both Pressurizer PORVs are CLOSED.
  - RCS subcooling is 15°F and stable.
  - RCS pressure is 1530 psig and stable.
  - All Reactor Coolant Pumps have been STOPPED.
  - All Steam Generator Pressures are 1080 psig and slowly rising.
  - COG-182, Condenser Off-Gas Radiation Monitor, is Normal.
  - MSL-178 through MSL-181, Main Steam Line Radiation Monitors, are GREEN and stable.
  - SGS-164, SG Blowdown Sample Radiation Monitor, is GREEN and stable.
  - All Steam Generator levels are 52% and slowly rising under operator control.
  - Containment Pressure is 0.2 psig and stable.
  - Containment Recirc Sump level is 808' and stable.
  - Containment Radiation on Grid 4 is Normal.
  - The following Area Radiation Monitors on Grid 4 indicate RED and rising.
    - 1-RE-6259A, PENET AREA RM 77S.
    - 1-RE-6259B, PENET AREA RM 77N.
    - X-RE-5570A, S WRGM EFFLUENT.

Which of the following indicates the proper action for optimal recovery?

A. Continue with EOP-0.0A, Reactor Trip or Safety Injection, as no procedure transitions have been identified.

B. Transition to EOS-1.1A, Safety Injection Termination, as ECCS flow is NOT required.

C. Transition to ECA-1.2A, LOCA Outside Containment, based on probable leakage into the Safeguards Building.

D. Transition to EOS-0.0, Rediagnosis, to identify what indication was misinterpreted.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because it could be evaluated that no procedural transitions have yet been identified, however, indications are correct for a transition to ECA-1.2A.

B. Incorrect. Plausible because with none of the major accident (SGTR, Faulted SG or LOCA) transitions identified, it could be interpreted that SI Termination should take place, however, subcooling would not allow SI Termination and a transition to ECA-1.2A is correct.

C. Correct. Interpretation of the readings on Grid 4 Radiation Monitors indicates that a transition to ECA-1.2A is required.

D. Incorrect. Plausible because the major diagnostic steps of EOP-0.0A have been completed, however, transition from EOP-0.0A to EOS-0.0A is not proper when a transition has not been identified.

Technical Reference(s): EOP-0.0A, Step 19 — Attached w/ Revision # See Comments / Reference
ECA-1.2A, Entry Conditions
EOS-0.0A, Entry Conditions
EOS-1.1A, Attachment 1.A

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the diagnostic steps of EOP-0.0, Reactor Trip or Safety Injection.

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

Question History: Last NRC Exam CPNPP 2012

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Comments / Reference: From EOP-0.0A, Step 19	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 13 OF 115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*17	Check SG Levels:	
	a. Narrow range level - GREATER THAN 43%	a. Maintain total AFW flow greater than 460 gpm until narrow range level greater than 43% in at least one SG.
	b. Control AFW flow to maintain narrow range level between 43% and 60%	b. IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.
	c. Any SG level increasing in an uncontrolled manner	c. Go to Step 18.
	d. Go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.	
18	Check Secondary Radiation - NORMAL • Condenser off gas radiation monitor (COG-182, 1RE-2959) • Main steamline radiation (MSL-178 through 181, 1RE-2325 through 2328) • SG blowdown sample radiation monitor (SGS-164, 1RE-4200)	Go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.
19	Check Auxiliary And Safeguards Building Radiation - NORMAL (GRID 4)	Evaluate cause of abnormal conditions. IF the cause is a loss of RCS inventory outside containment, THEN go to ECA-1.2A, LOCA OUTSIDE CONTAINMENT, Step 1.

Comments / Reference: From ECA-1.2A, Entry Conditions	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-1.2A
LOCA OUTSIDE CONTAINMENT	REVISION NO. 8	PAGE 2 OF 6

A. PURPOSE

This procedure provides actions to identify and isolate a LOCA outside containment.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2 and 3. This procedure assumes RHR is not in service. Using this procedure when not in these modes requires a step by step evaluation to determine if the required action is still applicable to current plant conditions.

C. SYMPTOMS OR ENTRY CONDITIONS

This procedure is entered from:

1) EOP-0.0A, REACTOR TRIP OR SAFETY INJECTION, on abnormal radiation in the auxiliary or safeguards building due to a loss of RCS inventory outside containment.

2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, if it is determined that the cause of abnormal radiation is due to a loss of RCS inventory outside containment.

Comments / Reference: From EOS-0.0A, Entry Conditions	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-0.0A
REDIAGNOSIS	REVISION NO. 8	PAGE 2 OF 10

A. PURPOSE

This procedure provides a mechanism to allow the operator to determine or confirm the most appropriate post accident recovery procedure.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2 and 3. This procedure assumes RHR is not in service and SI is operable. Using this procedure when not in these modes requires a step by step evaluation to determine if the required action is still applicable to current plant conditions.

C. SYMPTOMS OR ENTRY CONDITIONS

This procedure is entered based on operator judgement, but shall NOT be entered prior to completing EOP-0.0A.

Comments / Reference: From EOS-1.1A, Attachment 1.A	Revision # 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.1A
SAFETY INJECTION TERMINATION	REVISION NO. 8	PAGE 17 OF 48

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-1.1A, SI TERMINATION

1. SI REINITIATION CRITERIA

Following ECCS termination, (completion of Step 12), manually start ECCS pumps as necessary and go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1, if EITHER condition listed below occurs:

- RCS subcooling - LESS THAN 25°F (55°F FOR ADVERSE CONTAINMENT)
- PRZR level - CANNOT BE MAINTAINED GREATER THAN 13% (34% FOR ADVERSE CONTAINMENT)

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		
	K/A #	G 2.3.15	
	Importance Rating		3.1

Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO 98

Given the following conditions:

- Both Units are in MODE 1.
- Radiation Monitor X-RE-5895A, North Control Room Air intake fails LOW.
- Radiation Monitor X-RE-5895B, North Control Room Air intake is operating normally.
- Radiation Monitor X-RE-5896A, South Control Room Air Intake is operating normally.
- Radiation Monitor X-RE-5896B, South Control Room Air Intake is operating normally.

Which of the following identifies the Technical Specification requirements placed on the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System?

A. Place both Control Room HVAC Trains in the Emergency Recirculation Mode within 30 days.

B. Secure the Control Room Makeup Air Supply Fan from the North Air Intake within 7 days.

C. Restore the affected Control Room Emergency Filtration/Pressurization System Train to OPERABLE status within 7 days.

D. Restore the affected Control Room Air Conditioning System Train to OPERABLE status within 30 days.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because this action would meet the Required Actions, however, the time frame of 30 days would not satisfy Technical Specification 3.3.7.A. This is the Required Action if both Trains of Control Room Emergency Filtration System (CREFS) Actuation instrumentation were INOPERABLE with the incorrect Completion Time.

B. Correct. With one Air Intake Radiation Monitor INOPERABLE, place the associated CREFS Train in the Emergency Recirculation Mode or secure the affected Intake Makeup Air Supply Fan within 7 days per Technical Specification LCO 3.3.7.A.

C. Incorrect. Plausible because it could be thought that the Radiation Monitor failure affected the CREFS per Technical Specification LCO 3.7.10.

D. Incorrect. Plausible because it could be thought that the Radiation Monitor failure affected the Control Room Air Conditioning System per Technical Specification LCO 3.7.11.

Technical Reference(s): Technical Specification LCO 3.3.7.A & B; Technical Specification LCO Table 3.3.7-1; Technical Specification LCO 3.7.10.A; Technical Specification LCO 3.7.11 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **APPLY** the administrative requirements of the Control Room Emergency Filtration System including Technical Specifications, TRM and ODCM.

Question Source:
Bank # X
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 2

Comments / Reference: From Technical Specification LCO 3.3.7.A	Amendment # 156

CREFS Actuation Instrumentation
3.3.7

3.3 INSTRUMENTATION

3.3.7 Control Room Emergency Filtration System (CREFS) Actuation Instrumentation

LCO 3.3.7 The CREFS actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7-1

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel or train inoperable.	A.1 Place the affected CREFS train(s) in emergency recirculation mode.	7 days
	OR	
	A.2 --------------------NOTE-------------------- Applicable only to Functions 3a and 3b. -- Secure the Control Room makeup air supply fan from the affected air intake.	7 days

Comments / Reference: From Technical Specification LCO 3.3.7.B	Amendment # 156

CREFS Actuation Instrumentation
3.3.7

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with two channels or two trains inoperable.	B.1.1 Place one CREFS train in emergency recirculation mode.	Immediately
	AND	
	B.1.2 Enter applicable Conditions and Required Actions for one CREFS train made inoperable by inoperable CREFS actuation instrumentation	Immediately
	OR	
	B.2 -----------------NOTE------------------ Applicable only to Functions 3a and 3b. ---	
	Secure the Control Room makeup air supply fan from the affected air intake.	Immediately

Comments / Reference: From Technical Specification LCO Table 3.3.7-1	Amendment # 156

CREFS Actuation Instrumentation
3.3.7

Table 3.3.7-1 (page 1 of 1)
CREFS Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Manual Initiation	1, 2, 3, 4, 5, and 6, (a)	2 trains	SR 3.3.7.6	NA
2. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 5, and 6, (a)	2 trains	SR 3.3.7.2	NA
3. Control Room Radiation				
a. Control Room Air North Intake	1, 2, 3, 4, 5, and 6, (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	1.4×10^{-4} μCi/ml
b. Control Room Air South Intake	1, 2, 3, 4, 5, and 6, (a)	2	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.7	1.4×10^{-4} μCi/ml
4. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.			

(a) During movement of irradiated fuel assemblies.

Comments / Reference: From Technical Specification LCO 3.7.10.A	Amendment # 156

CREFS
3.7.10

3.7 PLANT SYSTEMS

3.7.10 Control Room Emergency Filtration/Pressurization System (CREFS)

LCO 3.7.10 Two CREFS trains shall be OPERABLE

--NOTE--
The Control Room envelope (CRE) boundary may be opened intermittently under administrative controls.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREFS train inoperable for reasons other than Condition B.	A.1 Restore CREFS train to OPERABLE status.	7 days

Comments / Reference: From Technical Specification LCO 3.7.11.A	Amendment # 156

CRACS
3.7.11

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning System (CRACS)

LCO 3.7.11 Two CRACS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRACS train inoperable.	A.1 Restore CRACS train to OPERABLE status.	30 days

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		
K/A #	G 2.4.29	
Importance Rating		4.4

Emergency Procedures / Plan: Knowledge of the emergency plan

Proposed Question: SRO 99

Given the following conditions:

- The Emergency Response Organization has been activated.
- A Site Area Emergency has been declared and a Site Evacuation is in progress.
- The Emergency Coordinator is in the Emergency Offsite Facility (EOF).

Which of the following actions may be delegated by the Emergency Coordinator?

A. Authorizing re-entry into evacuated areas.

B. Making Protective Action Recommendations to off-site authorities.

C. Restore Comanche Peak to fully operational status.

D. Approving Notification Message Forms prior to sending.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that the Operations Support Center Manager can authorize re-entry as the position controls ERDC Teams.
B. Incorrect. Plausible because PARS are reviewed by Radiation Protection prior to sending, however, this function cannot be delegated.
C. Correct. As listed in EPP-109, Step 4.1.2 and is a responsibility of the Recovery Manager when the Recovery Organization is formed.
D. Incorrect. Plausible because the EOF Communicator sends the messages, however, the Emergency Coordinator must approve Notification Message Forms.

Technical Reference(s) EPP-109, Steps 4.1.1, 4.1.2, & 4.2 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **INTERPRET** and **ENSURE** compliance with plant administrative and operational procedures, guidelines, and policies.

Question Source:	Bank #	X	
	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	5	

Comments / Reference: From EPP-109, Step 4.2	Revision # 14

4.2 Non-delegatable Duties

The Emergency Coordinator shall not delegate decision making authority for:

- recommending use of potassium iodide.
- authorizing re-entry into evacuated onsite areas.
- authorizing personnel exposures in excess of 10CFR, Part 20 limits. [C-06380]
- making protective action recommendations to offsite authorities.
- approving notification messages. [C-05325]

Comments / Reference: From EPP-109, Step 4.1.1	Revision # 14

4.1.1 (Continued)

NOTE: Duties and responsibilities assigned to the Emergency Coordinator are transferred to the Recovery Manager when the Recovery Organization is formed. [C-05765], [C-05761]

- Immediately and unilaterally initiate all provisions of the CPNPP Emergency Plan, and for evaluation, coordination and control of all onsite activities related to the CPNPP emergency response until the event is closed out or the Recovery Organization is formed.

Comments / Reference: From EPP-109, Step 4.1.2	Revision # 14

4.1.2 Recovery Manager

Direct actions of the CPNPP Recovery Organization and to restore CPNPP to fully operational status.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		
	K/A #	G 2.4.40	
	Importance Rating		4.5

Emergency Procedures / Plan: Knowledge of SRO responsibilities in emergency plan implementation

Proposed Question: SRO 100

Given the following condition:

- A security-related imminent threat has occurred at the station.

Which of the following identifies the requirement to notify the Nuclear Regulatory Commission using the Emergency Notification System?

The notification should be made [1] calls are made to local agencies, state agencies and plant personnel NOT to exceed [2] hour(s) from the declaration.

	[1]	[2]
A.	after	1
B.	after	4
C.	before	1
D.	before	4

Proposed Answer: A

Explanation:

A. Correct. As outlined in the EPP-203.

B. Incorrect. Plausible because the NRC is notified after the local and state agencies, however, the NRC must be notified within 1 hour.

C. Incorrect. Plausible because the NRC is to be informed as soon as possible under these conditions but after the local and state agencies. The NRC must be notified within 1 hour.

D. Incorrect. Plausible because the NRC is to be informed as soon as possible under these conditions but after the local and state agencies, the NRC must be notified within 1 hour.

Technical Reference(s) EPP-203, Step 4.1.5 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **IDENTIFY** time requirements for emergency notifications.

Question Source: Bank # X
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 5

Comments / Reference: From EPP-203, Step 4.1.5	Revision # 16

4.1.5 Notify the NRC Operations Center of any emergency at CPNPP by using the Emergency Notification System (ENS).

4.1.5.1 The Center should be contacted **immediately after** the notification to local agencies, state agencies, plant personnel and ERO personnel, **not to exceed one hour from the initial declaration of an emergency**.

4.1.5.2 For a security related imminent threat or attack against the station, the notification should be immediate with a goal to initiate the call within approximately 15 minutes.

4.1.5.3 Once contacted, the NRC duty officer may request continuous communications. If so, a person knowledgeable of CPNPP plant operations shall be required to remain on the ENS line throughout the emergency.

4.1.5.4 During the event, the NRC Operations Center shall be immediately notified of any further degradation in the level of safety, worsening plant conditions, escalation of the emergency classification, or termination of the emergency. [C-05429]

CPNPP NRC 2013 RO Written Exam Reference List

1. NRC Generic Fundamentals Equation Sheet
2. Steam Tables

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$\dot{Q} = \dot{m}c_p\Delta T$

$\dot{Q} = \dot{m}\Delta h$

$\dot{Q} = UA\Delta T$

$\dot{Q} \propto \dot{m}^3_{Nat\ Circ}$

$\Delta T \propto \dot{m}^2_{Nat\ Circ}$

$K_{eff} = 1/(1 - \rho)$

$\rho = (K_{eff} - 1)/K_{eff}$

$SUR = 26.06/\tau$

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\ \rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\bar{\beta}_{eff}}{1 + \lambda_{eff}\ \tau}$$

$\ell^* = 1 \times 10^{-4}$ sec

$\lambda_{eff} = 0.1\ sec^{-1}$ (for small positive ρ)

$DRW \propto \varphi^2_{tip}/\varphi^2_{avg}$

$P = P_o 10^{SUR(t)}$

$P = P_o e^{(t/\tau)}$

$A = A_o e^{-\lambda t}$

$CR_{S/D} = S/(1 - K_{eff})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$1/M = CR_1/CR_X$

$A = \pi r^2$

$F = PA$

$\dot{m} = \rho A\vec{v}$

$\dot{W}_{Pump} = \dot{m}\Delta P\upsilon$

$E = IR$

Thermal Efficiency = Net Work Out/Energy In

$$\frac{g(z_2 - z_1)}{g_c} + \frac{(\vec{v}_2^{\,2} - \vec{v}_1^{\,2})}{2g_c} + \upsilon(P_2 - P_1) + (u_2 - u_1) + (q - w) = 0$$

$g_c = 32.2$ lbm-ft/lbf-sec^2

CONVERSIONS

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1 Mw	= 3.41×10^6 Btu/hr	1 Curie	= 3.7×10^{10} dps
1 hp	= 2.54×10^3 Btu/hr	1 kg	= 2.21 lbm
1 Btu	= 778 ft-lbf	1 gal_{water}	= 8.35 lbm
°C	= (5/9)(°F - 32)	1 ft^3_{water}	= 7.48 gal
°F	= (9/5)(°C) + 32		

CPNPP NRC 2013 SRO Written Exam Reference List

1. NRC Generic Fundamentals Equation Sheet
2. EPP-201, Emergency Action Level Classification Charts
   (available in Exam Room)
3. EPP-201, Emergency Action Level Technical Bases Document
   (available in Exam Room)
4. Steam Tables

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

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