CPNPP 2013 NRC Written Examination
Senior Reactor Operator
Answer Key

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

Examination Outline Cross-reference:

	RO	SRO
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

What are the power supplies for Component Cooling Water (CCW) Pumps 2-01 and 2-02?

	CCWP 2-01	CCWP 2-02
A.	2EA1	2EB1
B.	2EA1	2EB2
C.	2EB1	2EB2
D.	2EA1	2EA2

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if believed that the ‘A’ and ‘B’ in the bus designator are related to the Train they supply power from.
B. Incorrect. Plausible if believed that the ‘A’ and ‘B’ in the bus designator are related to the Train they supply power from and the 1 and 2 are also part of the Train designator.
C. Incorrect. Plausible if believed that the CCWPs are powered form 480V AC.
D. Correct. 2-EA1 and 2EA2 are the correct power supplies.

Technical Reference(s) LO21.SYS.CC1 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 X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 3, 7
55.43 ______

Comments / Reference: From LO21.SYS.CC1.LN, Page 14	Revision 5/1/2011

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.

Examination Outline Cross-reference:

	RO	SRO
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

Given the following conditions:

- 1-PT-0456, PRESSURIZER 1-01 PRESSURE TRANSMITTER 0456 PROT CHAN II has failed.
- Maintenance was dispatched and reported that the transmitter had no power.

What is the power supply for 1-PT-0456, PRESSURIZER 1-01 PRESSURE TRANSMITTER 0456 PROT CHAN II?

A. 1PC1, 118 VAC DISTRIBUTION PANEL 1PC1

B. 1EC1, 118 VAC DISTRIBUTION PANEL 1EC1

C. 1PC2, 118 VAC DISTRIBUTION PANEL 1PC2

D. 1EC2, 118 VAC DISTRIBUTION PANEL 1EC2

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that because the channel is a controlling channel it is powered from channel I, however PT-456 is a channel II instrument.
B. Incorrect. Plausible if thought that because the channel is a controlling channel it is powered from channel I, however PT-456 is a channel II instrument that it is powered from an instrument bus vice a protection bus.
C. Correct. The power supply for PT-456 is 1PC2.
D. Incorrect. Plausible if thought that PT-456 is powered from an instrument bus vice a protection bus.

Technical Reference(s) ABN-603, Attachment 1 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 #		
	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, Attachment 1	Revision 8, PCN 1

CPNPP ABNORMAL CONDITIONS PROCEDURES	UNIT 1 AND 2	PROCEDURE N ABN-603
LOSS OF PROTECTION OR INSTRUMENT BUS	REVISION NO. 8	PAGE 24 OF 3

ATTACHMENT 1
PAGE 1 OF 2

PROTECTION SET CABINETS AND MISCELLANEOUS LOADS

		PROTECTION BUS			
	DESCRIPTION	u PC1	u PC2	u PC3	u PC4
PROTECTION SET	RCL 1 FLOW	FT-414	FT-415	FT-416	-----------
	RCL 2 FLOW	FT-424	FT-425	FT-426	-----------
	RCL 3 FLOW	FT-434	FT-435	FT-436	-----------
	RCL 4 FLOW	FT-444	FT-445	FT-446	-----------
	BAT 1 LEVEL	XLT-102	-----------	-----------	XLT-104
	BAT 2 LEVEL	-----------	-----------	XLT-105	XLT-106
	PRZR LEVEL	LT-459(S)	LT-460(S)	LT-461(S)	-----------
	S/G LEVEL-WIDE RANGE	LT-501	LT-502	LT-503	LT-504
	S/G 1 LEVEL	LT-551(S)	LT-519(S)	LT-518	LT-517
	S/G 2 LEVEL	LT-529(S)	LT-552(S)	LT-528	LT-527
	S/G 3 LEVEL	LT-539(S)	LT-553(S)	LT-538	LT-537
	S/G 4 LEVEL	LT-554(S)	LT-549(S)	LT-548	LT-547
	RWST LEVEL	LT-930	LT-931	LT-932	LT-933
	SOURCE RANGE	N31	N32	-----------	-----------
	INTERMEDIATE RANGE	N35	N36	-----------	-----------
	POWER RANGE	N41	N42	N43	N44
	RCS PRESSURE	PT-405	-----------	-----------	PT-403
	PRZR PRESSURE	PT-455(S)	PT-456(S)	PT-457(S)	PT-458(S)

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	Attached w/ Revision # See Comments / Reference
	ABN-104, Steps 8.3.9, 8.3.10, & 8.3.11	

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. <u>Ensure a Hot Leg vent path, and initiate Hot Leg injection with an SIP.</u>

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

Given the following conditions:

- Unit 1 has experienced a Large Break Loss of Coolant Accident.
- All protection systems responded as expected during the initial phase of the accident.
- Containment pressure rose to 43 psig before being lowered by Containment Spray.
- The crew has placed the unit in hot leg recirculation per EOS-1.4A, Transfer to Hot Leg Recirculation.
- 1-HCV-607, RHR HX 1-02 FLO CTRL VLV has closed and CANNOT be reopened.

Which of the following describes the impact on the Emergency Core Cooling System and design analysis assumptions with regard to the Emergency Core Cooling System?

Overall cooling to the core is…

A. …unchanged because only one train of Emergency Core Cooling is assumed available.

B. …reduced but design analysis assumptions remain valid.

C. …unchanged because only one train of Emergency Core Cooling is used during hot leg recirculation.

D. …reduced and design analysis assumptions are NO longer valid.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that loss of a single train of RHR is assumed in the design analysis and that that therefore does not impact cooling.
B. Correct. Core cooling is reduced but design analysis assumptions remain valid because loss of a single train is covered in the analysis.
C. Incorrect. Plausible if thought that a single train is used for hot leg Recirc due to how the system is configured (only hot legs 2 & 3 are injected into by RHR in hot leg Recirc).
D. Incorrect. Plausible because overall core cooling is reduced but the design assumptions account for the loss of a single train of cooling.

Technical Reference(s) LO21.SYS.SI1 Attached w/ Revision # See Comments / Reference
LO21.SYS.RH1

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 # (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, 7
55.43

Comments / Reference: From LO21.SYS.SI1, PAGES 7 & 8	Revision 5/2/2011
The Emergency Core Cooling System is designed to remain functional after a Safe Shutdown Earthquake. The system is also protected from flooding, pipe whips, jet forces, and missiles. The system is designed to tolerate a single failure without a loss of its core protective functions. This failure is limited to an *active failure* during the short-term (injection) phase following a LOCA. When applied to the Emergency Core Cooling System, an *active failure* is defined as the failure of a powered component, such as a pump, power actuated valve, a component of the electrical supply system or instrumentation and control equipment, to act to accomplish its design functions. The worst-case single active failure would be the loss of one entire safeguards electrical train. The single failure is also limited to an *active or passive failure* during the long-term (recirculation) cooling phase. A *passive failure* is defined as the structural failure of a static component. When applied to a fluid system, this means a break in the pressure boundary resulting in abnormal leakage not exceeding 50 gpm for 30 minutes. Such leaks are consistent with limited cracks in pipes, sprung flanges, valve packing leaks or pump seal failures.	

Comments / Reference: From LO21.SYS.RH1, Page 25	Revision 10/20/2011

The rate of heat removal is controlled manually by the operator by adjusting u-HC-606 (Train A) or u-HC-607 (Train B) to position the Flow Control Valves. The RHR Heat Exchanger Bypass Valves are operated in automatic control. As the operator manually positions the RHR Heat Exchanger Flow Control Valve to control Reactor Coolant temperature or cooldown rate, the RHR Heat Exchanger Bypass Valve repositions to maintain constant flow through the system

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 EOS-1.2A, Post LOCA Cooldown and Depressurization.
- All Reactor Coolant Pumps have been stopped.
- Residual Heat Removal Pumps are in STANDBY.
- Safety Injection Pumps are in STANDBY.
- Normal charging flow has been established.
- Containment pressure is 6 psig and stable.
- The following indications are observed;
    - Reactor Coolant System Hot Leg temperatures are all 460°F and rising.
    - Reactor Coolant System pressure is 835 psig and lowering.
    - Subcooling is 65°F and becoming less subcooled.
    - Pressurizer level is 38% and slowly lowering.

Which of the following actions should be taken in response to the given conditions per EOS-1.2A, Post LOCA Cooldown and Depressurization?

A. Immediately manually start and align Emergency Core Cooling System pumps.

B. Immediately re-actuate Safety Injection to align Emergency Core Cooling System.

C. When subcooling or pressurizer level meets foldout page criteria then manually start and align Emergency Core Cooling System pumps.

D. When subcooling or pressurizer level meets foldout page criteria then re-actuate Safety Injection to align Emergency Core Cooling System.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because manually starting and aligning ECCS pumps is required per EOS-1.2 foldout page, however, subcooling is NOT less than 55°F or PZRZ level is NOT less than 34% for adverse containment.

B. Incorrect. Plausible because it may be thought that re-actuating SI is the proper action because EOS-1.2A, foldout page title for step is SI REINITIATION CRITERIA, however, subcooling is NOT less than 55°F or PZRZ level is NOT less than 34% for adverse containment.

C. Correct. Manually starting and aligning ECCS pumps is correct action when subcooling is less than 55°F or PZRZ level is less than 34%.

D. Incorrect. Plausible because it may be thought that re-actuating SI is the proper action because EOS-1.2A, foldout page title for step is SI REINITIATION CRITERIA when subcooling is less than 55°F or PZRZ level is less than 34%.

Technical Reference(s) EOS-1.2A, foldout page and bases 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 EOS-1.2A Foldout Page	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.2A
POST LOCA COOLDOWN AND DEPRESSURIZATION	REVISION NO. 8	PAGE 26 OF 68

ATTACHMENT 1.A
PAGE 1 OF 1

FOLDOUT FOR EOS-1.2A, POST LOCA COOLDOWN AND DEPRESSURIZATION

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. 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-2.0A, FAULTED STEAM GENERATOR ISOLATION, Step 1.

3. EOP-3.0A TRANSITION CRITERIA

   Manually start ECCS pumps as necessary and go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1, if any SG level increases in an uncontrolled manner or any SG has abnormal radiation.

4. COLD LEG RECIRCULATION SWITCHOVER CRITERION

   IF RWST level decreases to less than LO-LO LEVEL, THEN go to EOS-1.3A, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

5. AFW SUPPLY SWITCHOVER CRITERION

   IF CST level decreases to less than 10%, THEN switch to alternate AFW water supply per ABN-305, AUXILIARY FEEDWATER SYSTEM MALFUNCTION.

Comments / Reference: From EOS-1.2A, Attachment 1.A bases	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.2A
POST LOCA COOLDOWN AND DEPRESSURIZATION	REVISION NO. 8	PAGE 63 OF 68

ATTACHMENT 5
PAGE 26 OF 31

BASES

ATTACHMENT 1.A

SI REINITIATION CRITERIA - If RCS subcooling is lost or if pressurizer level cannot be maintained with charging flow, the operator is instructed to start ECCS pumps as necessary. No preference is given about which pumps to start to establish ECCS flow. The operator has the option to select either the SI pumps or the CCPs. If the CCPs are selected, the suction and discharge path of the pump must be realigned to the Safety Injection mode, and the normal charging and suction path must be isolated. Failure to maintain the limits for the reinitiation criteria indicate that control of the plant is not being maintained as expected in this procedure and additional ECCS flow is necessary.

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

Given the following:

- A 3" Small Break Loss of Coolant Accident has occurred on Unit 1.

Which of the following describes the expected response of the Safety Injection Pump discharge flow as Reactor Coolant System pressure lowers from 1400 psig to 1200 psig?

Each Safety Injection Pump is initially discharging at...

A. …≈100 gpm and stable.

B. …0 gpm and stable.

C. …≈100 gpm and steadily rising to ≈250 gpm.

D. …0 gpm and steadily rising to ≈150 gpm.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that the Safety Injection Pump curves are vertical between 1400 and 1200 psig RCS pressure, however the flow is rising at a fairly substantial rate along this portion of the pump curve as the curve is more horizontal than vertical.

B. Incorrect. Plausible if thought that the Safety Injection Pump begin injecting into the RCS below 1200 psig, however, based on shutoff head and line losses the pumps begin injecting into the RCS at a pressure of approximately 1450 psig.

C. Correct. At 1400 psig RCS pressure (which equates to approximately 3500 ft of head, when RWST suction pressure and injection line losses are considered), each Safety Injection Pump would be injecting approximately 100 gpm with 45 gpm of recirculation flow. At 1200 psig RCS pressure (with similar assumptions as the 1400 psig condition) the pump would be injecting approximately 250 gpm based on the CPNPP Safety Injection Pumps.

D. Incorrect. Plausible if thought that the Safety Injection Pump begin injecting slightly below 1400 psig RCS pressure and flow would then increase at approximately the same rate as the actual pump head curve, however, based on shutoff head and line losses the pumps begin injecting into the RCS at a pressure of approximately 1450 psig.

Technical Reference(s) OPT-513A-1 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 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.

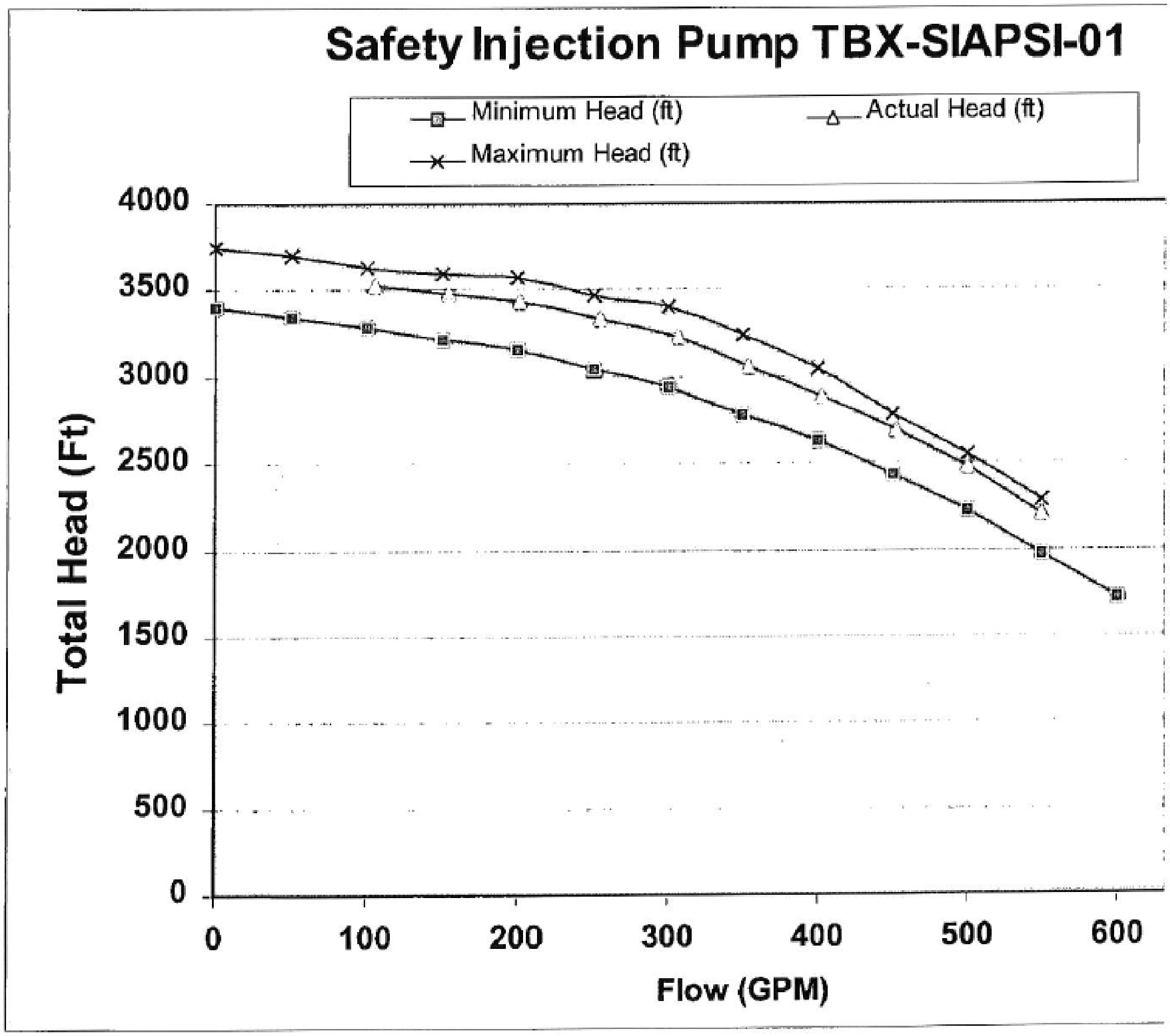
Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

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 OPT-513A-1	4/17/2001 Data



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 228°F.

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

A. ≈ 2 psig

B. ≈ 5 psig

C. ≈ 10 psig

D. ≈ 15 psig

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if Mollier Diagram is improperly read.
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 228°F intersects the Saturation Curve corresponds to a Containment pressure of approximately 20 psia, which is equal to 5 psig.
C. Incorrect. Plausible if Mollier Diagram is improperly read.
D. Incorrect. Plausible if Mollier Diagram is improperly read.

Technical Reference(s) Mollier Diagram — 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.

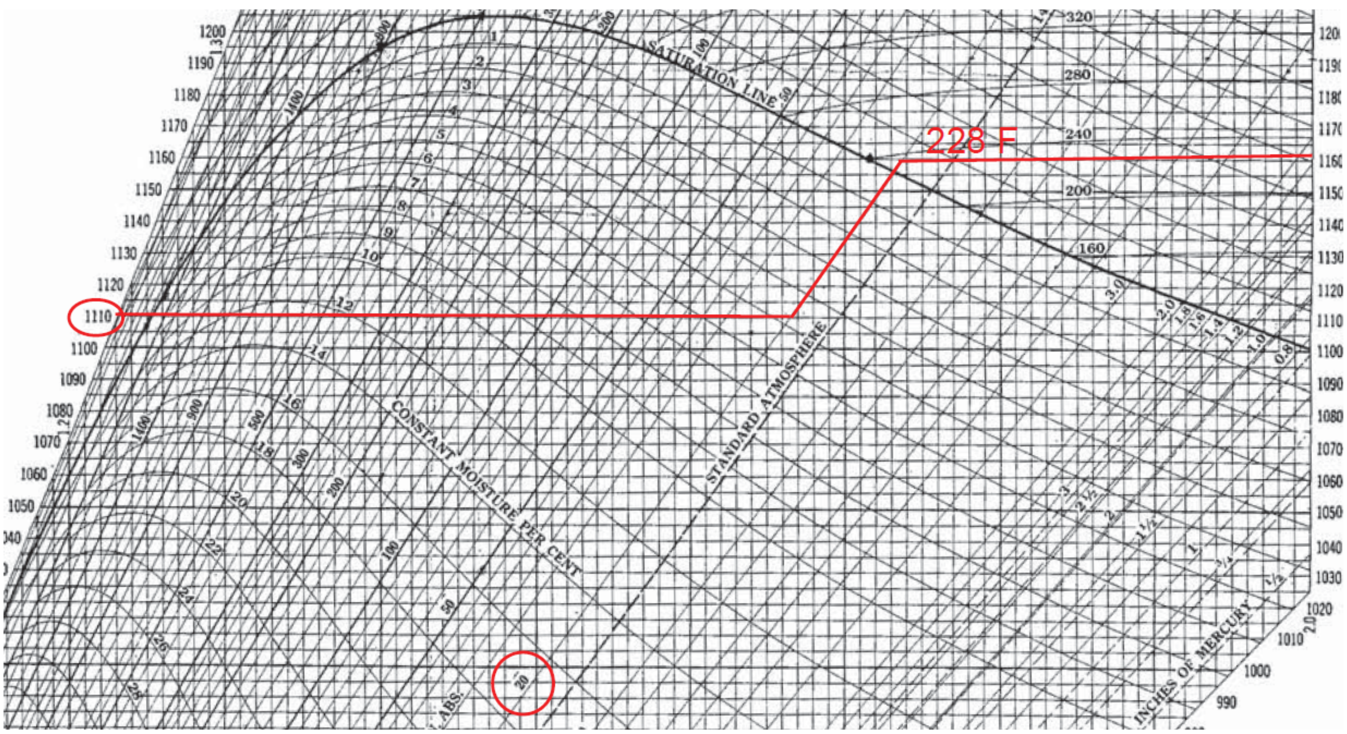
Question Source: Bank # ______
Modified Bank # X (Note changes or attach parent)
New ______

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 7, 14
55.43 ______

Comments / Reference: From Mollier Diagram	Revision N/A



Original Bank Question
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.

Examination Outline Cross-reference:

	RO	SRO
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.

B. Train A CCW Pump trips; Train B CCW Pump remains in STANDBY.

C. Train A Safeguards Loop Isolation Valves close; Train B CCW Pump AUTO starts.

D. Train A Safeguards Loop Isolation Valves close; Train B CCW Pump remains in STANDBY.

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 Attached w/ Revision # See Comments / Reference
ABN-502, Steps 2.1, 2.2, & 3.2

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	

Pressurizer Pressure Control System: 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 at approximately 2185 psig.

D. Unit remains at power, with pressure being controlled at approximately 2335 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	

Reactor Protection System: 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 DC bus 2ED1?

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: With the Reactor Trip breakers being part of the Reactor Protection System the following plausibility statements are provided;

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 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

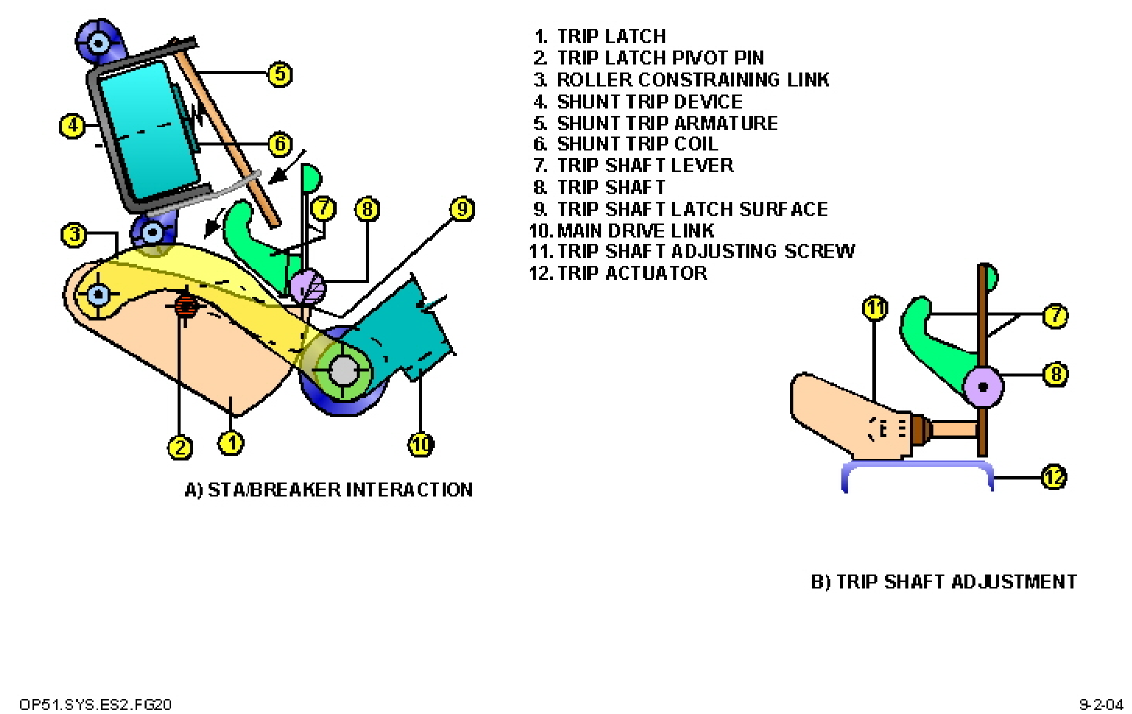
THE SHUNT TRIP COIL

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

What pumps make up a complete Emergency Core Cooling System safety train in MODE 4?

A. A Centrifugal Charging Pump, a Safety Injection Pump and a Residual Heat Removal Pump.

B. A Safety Injection Pump and a Residual Heat Removal Pump.

C. A Safety Injection Pump and a Centrifugal Charging Pump.

D. A Centrifugal Charging Pump and a Residual Heat Removal Pump.

Proposed Answer: D

Explanation: In order to define a safety train the Technical Specification bases are used as the source of the definition. That definition changes based on MODE.

A. Incorrect. Plausible because a Centrifugal Charging Pump, a Safety Injection Pump and a Residual Heat Removal Pump are required in MODE 1, 2 & 3 per T/S LCO 3.5.2.
B. Incorrect. Plausible if thought that a Safety Injection Pump and a Residual Heat Removal Pump meet the requirement in MODE 4 per T/S LCO 3.5.3.
C. Incorrect. Plausible if thought that a Safety Injection Pump and a Centrifugal Charging Pump meet the requirement in MODE 4 per T/S LCO 3.5.3.
D. Correct. A Centrifugal Charging Pump and a Residual Heat Removal Pump are required in MODE 4 per T/S LCO 3.5.3.

Technical Reference(s): Technical Specification 3.5.2; Technical Specification 3.5.3 — 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 Solid State Protection System.

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

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, 7, 8
55.43 ________

Comments / Reference: From Technical Specification B3.5.2	Revision 67

BASES (continued)

LCO

In MODES 1, 2, and 3, two independent (and redundant) ECCS trains are required to ensure that sufficient ECCS flow is available, assuming a single failure affecting either train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

In MODES 1, 2, and 3, an ECCS train consists of a centrifugal charging subsystem, an SI subsystem, and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST upon an SI signal and initiating semi-automatic switchover of suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the four cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to supply its flow to the RCS hot and cold legs.

The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains.

Comments / Reference: From Technical Specification B3.5.3	Revision 67

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS - Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS - Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: centrifugal charging (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of 10CFR50.36(c)(2)(ii).

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

In MODE 4, an ECCS train consists of a centrifugal charging subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

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.

What MIMIMUM containment pressure must be maintained during the containment vent and what is the basis for this value?

A. -0.3 psig to ensure containment will NOT exceed design negative differential pressure following inadvertent Containment Spray system actuation.

B. 0.0 psig to ensure containment will NOT exceed design negative differential pressure following inadvertent Containment Spray system actuation.

C. -0.3 psig to ensure instrumentation impacted by containment pressure will operate as designed during normal, abnormal and emergency conditions.

D. 0.0 psig to ensure instrumentation impacted by containment pressure will operate as designed during normal, abnormal and emergency conditions.

Proposed Answer: A

Explanation:

A. Correct. Per Technical Specification LCO 3.6.4 bases.
B. Incorrect. Plausible because the reason is correct but the minimum pressure is not correct.
C. Incorrect. Plausible because the pressure is correct but the reason is not correct.
D. Incorrect. Plausible if thought the pressure and the reason are correct.

Technical Reference(s): Technical Specification LCO 3.6.4; Technical Specification LCO 3.6.4 bases — 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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	8 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Comments / Reference: From Technical Specification LCO 3.6.4 bases	Revision 67

BASES

APPLICABLE SAFETY ANALYSES (continued)

backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 2).

Containment pressure satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure following the inadvertent actuation of the Containment Spray System. An instrument uncertainty of ± 0.2 psi is conservatively included in the pressure limits (-0.3 to +1.3 psig) to allow the use of installed instrumentation for pressure measurements.

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 1-HS-5405A, CNTMT FN CLR FN 1, two minutes following a Safety Injection?

	GREEN	AMBER	RED
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 40 seconds after 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, however control power for the circuit is 125 VDC.

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. Place Containment Spray Pump 1-02 in PULLOUT and check for flow and pressure improvement.

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. Step 12 of EOS-1.3A does have the operator makeup to the RWST which could lead an individual to select distractor B.
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 the Loop 1 T_{AVE} Channel failure?

A. Steam Dumps will OPEN and reduce T_{AVE} to LESS than No-Load T_{AVE}.

B. Steam Dumps will OPEN and reduce T_{AVE} to No-Load T_{AVE}.

C. Steam Dump System will automatically shift from the Plant Trip Controller to the Steam Pressure Mode and reduce steam pressure to LESS than 1092 psig.

D. Steam Dump System will automatically shift from the Plant Trip Controller to the Steam Pressure Mode and reduce steam pressure to 1092 psig.

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.

B. Incorrect. Plausible because the Steam Dump Valves will normally control temperature at no load temperature without the channel failure.

C. Incorrect. Plausible because it could be thought that the system would swap to Pressure Control because this is the mode normally used when below no load temperature, however, the system does not automatically shift to Pressure Control mode.

D. Incorrect. Plausible because it could be thought that the system would swap to Pressure Control because this is the mode normally used when below no load temperature, however, the system does not automatically shift to Pressure Control mode and temperature will stabilize below no load temperature.

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 # ____
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 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 <u>OR</u> TRENDING TO 557°F	<u>IF</u> temperature less than 557°F and decreasing, <u>THEN</u> perform the following: a. Stop dumping steam.

Original Bank Question

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 557°F. 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 557°F. 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 557°F.

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

Examination Outline Cross-reference:

	RO	SRO
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 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

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 rising.
- 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 1-01 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 4 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 4	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 6 OF 103

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION: If any ruptured SG is faulted, feed flow to that SG should remain isolated during subsequent recovery actions unless needed for RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 4	Check Ruptured SG(s) Level:	
	a. Narrow range level - GREATER THAN 43% (50% FOR ADVERSE CONTAINMENT)	a. Maintain AFW flow to ruptured SG until level greater than 43% (50% FOR ADVERSE CONTAINMENT). Continue with Step 5. OBSERVE CAUTION PRIOR TO STEP 5. **WHEN** ruptured SG level greater than 43% (50% FOR ADVERSE CONTAINMENT), **THEN** stop AFW flow to ruptured SG(s).
	b. Stop AFW flow to ruptured SG(s).	

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 indications on Unit 1 DC Safeguards Switch Panels (Buses) and Batteries:

- DC Switch Panel 1ED1 is at 135 volts with battery BT1ED1 at 2 amp CHARGE.
- DC Switch Panel 1ED2 is at 136 volts with battery BT1ED2 at 2 amp CHARGE.
- DC Switch Panel 1ED3 is at 135 volts with battery BT1ED3 at 1 amp CHARGE.
- DC Switch Panel 1ED4 is at 140 volts with battery BT1ED4 at 10 amp CHARGE.

Which of the following lists the status of the Unit 1 DC Safeguards batteries?

	BT1ED1	BT1ED2	BT1ED3	BT1ED4
A.	FLOAT	EQUALIZE	FLOAT	EQUALIZE
B.	EQUALIZE	FLOAT	EQUALIZE	FLOAT
C.	EQUALIZE	EQUALIZE	EQUALIZE	FLOAT
D.	FLOAT	FLOAT	FLOAT	EQUALIZE

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the higher voltages on 1ED2 and 1ED4 indicate the battery is in EQUALIZE.
B. Incorrect. Plausible because it could be thought that the higher voltages on 1ED2 and 1ED4 indicate the battery is in FLOAT.
C. Incorrect. Plausible because of a misconception with regard to indications of FLOAT and EQUALIZE being reversed.
D. Correct. Voltage of between 138 VDC and 140 VDC is expected during an EQUALIZE charge on a battery with increased charging current.

Technical Reference(s) SOP-605A 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 # ____
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, 8
55.43 ____

Comments / Reference: From SOP-605A	Revision 11

CPSES SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1 and COMMON	PROCEDURE NO. SOP-605A
125 VDC SWITCHGEAR AND DISTRIBUTION SYSTEMS, BATTERIES AND BATTERY CHARGERS	REVISION NO. 11	PAGE 35 OF 74

NOTE: If the equalize timer is set for all zeros and the EQUALIZE button is depressed, the charger will remain in equalize until the FLOAT button is depressed.

5.2 C. Perform the following for the selected Battery Charger:

☐ 1) Ensure the timer mode is set to D.

☐ 2) Set Timer to desired time [normally set to zero (0)].

3) Depress EQUALIZE pushbutton and verify the following:

- ☐ EQUALIZE amber light is lit.
- ☐ Voltage is ≥ 138 VDC but ≤ 140 VDC on the selected battery charger.

☐ 4) WHEN directed by Electrical Maintenance, depress the FLOAT pushbutton AND verify the following:

- ☐ FLOAT green light is lit.
- ☐ Voltage indicates FLOAT voltage (128 – 135 VDC) on the selected battery charger.

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 / ABN-602, Steps 2.3.4 & 2.3.5 — 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 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 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, 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:

- Turbine Building Sumps are being released per STA-603-16, Secondary Waste Release Data Sheet.
- 1-RE-5100 (TBD172) TURBINE BUILDING DRAINS RADIATION MONITOR has experienced a loss of power.

Which of the following actions will occur?

A. 1-RV-5100A, TURB BLDG SMP 1-02 DISCH DRN HDR TO LVW/EVAP POND ISOL VLV will CLOSE and 1-RV-5100B, TURB BLDG SMP 1-02 DISCH HDR TO WWHT ISOL VLV will CLOSE.

B. 1-RV-5100A, TURB BLDG SMP 1-02 DISCH DRN HDR TO LVW/EVAP POND ISOL VLV will OPEN and 1-RV-5100B, TURB BLDG SMP 1-02 DISCH HDR TO WWHT ISOL VLV will CLOSE.

C. 1-RV-5100A, TURB BLDG SMP 1-02 DISCH DRN HDR TO LVW/EVAP POND ISOL VLV will CLOSE and 1-RV-5100B, TURB BLDG SMP 1-02 DISCH HDR TO WWHT ISOL VLV will OPEN.

D. 1-RV-5100A, TURB BLDG SMP 1-02 DISCH DRN HDR TO LVW/EVAP POND ISOL VLV will OPEN and 1-RV-5100B, TURB BLDG SMP 1-02 DISCH HDR TO WWHT ISOL VLV will OPEN.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that all discharge will isolate on a loss of power to the radiation monitor, however the loss of power to the monitor has the same effect as a high radiation signal which would cause 1-RV-5100A to close isolating the Low Volume Waste flowpath and 1-RV-5100B to open aligning the Co-Current Waste flowpath.

B. Incorrect. Plausible if thought normal release flowpath is to Co-Current Waste, however 1-RV-5100A will close isolating the Low Volume Waste flowpath and 1-RV-5100B will open aligning the Co-Current Waste flowpath.

C. Correct. The normal release path is to Low Volume Waste, so1-RV-5100A will close isolating the Low Volume Waste flowpath and 1-RV-5100B will open aligning the Co-Current Waste flowpath.

D. Incorrect. Plausible if thought that 1-RV-5100A and 1-RV-5100B open in response to the loss of power to the radiation monitor which controls valve positions.

Technical Reference(s) ALM-3200 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 ALM-3200	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
Turbine Building Drains	u-RE-5100	Closes the discharge to Low Volume Waste (u-RV-5100A) and opens discharge to Co-Current Waste	E1-0055 Sh 61/62 E2-0055 Sh 61/62
Containment Air Gaseous and Particulate	u-RE-5503 u-RE-5502	Causes Containment Ventilation Isolation on High Radiation	E1-0046 Sh 62/64 E2-0046 Sh 62/64
Control Room Air Supply (Gas)	X-RE-5895A/B X-RE-5896A/B	Initiates Control Room Emergency Recirculation on High Radiation	E1-0046 Sh 62/63 E1-0035 Sh 76/77
Secondary Sample	u-RE-4200	Isolates Steam Generator Blowdown and Sampling System on High Radiation	E1-0040 Sh 97 E2-0040 Sh 97
Common discharge AB, DG Sumps and CCW Drain Tanks	X-RE-5251A	Diverts to Cocurrent Waste System Wastewater Holdup Tanks on High Radiation or any OPERATE FAILURE	E1-0065 Sh 58

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 in MODE 3.
- The Main Steam Isolation Valves are CLOSED.
- Reactor Coolant System temperature is being maintained by steaming through the Atmospheric Relief Valves.
- Steam Generator Blowdown is being maintained at maximum for Chemistry control.
- A Safety Injection actuates on Low Pressurizer Pressure.

When performing EOP-0.0A, Reactor Trip or Safety Injection, which of the following Process Radiation Monitor Trends would be used in determining that a Steam Generator Tube Rupture was the initiating event for the Safety Injection?

A. Condenser Off Gas

B. Steam Generator Blowdown Sample

C. Main Steamline Radiation

D. Main Steamline N-16 Radiation

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the Condenser Off Gas Monitor is commonly used as indication of a Steam Generator Tube Rupture, however with the MSIVs closed the radioactive steam is bypassing the condenser and the trend would not aid is identifying a SGTR.

B. Correct. In accordance with EOP-0.0A the Steam Generator Blowdown Sample Radiation Monitor would indicate an increasing trend until either reaching the point that the monitor isolates Steam Generator Blowdown or the Safety Injection isolated the Steam Generator Blowdown.

C. Incorrect. Plausible because the Main Steamline Radiation Monitors are commonly used as indication of a Steam Generator Tube Rupture, however with the MSIVs closed the radioactive steam is bypassing the Main Steamline Monitors and the trend would not aid is identifying a SGTR.

D. Incorrect. Plausible because the Main Steamline N-16 monitors are the most sensitive during full power operation, however in MODE 3, no N-16 is being produced. Thus the Main Steamline N-16 Monitors would not aid in identifying a SGTR.

Technical Reference(s) EOP-0.0A, Step 13 Attached w/ Revision # See Comments / Reference
LO21.SYS.MR1
LO21.SYS.SB1

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 # ______
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, 11
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 115

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 LO21.SYS.MR1 p. 32	Revision 6-9-2011

Main Steam Line Radiation Monitors

To detect SG primary-to-secondary tube leakage, each Main Steam line is equipped with a two types of adjacent-to-line (ATL) radiation monitors. Using different detector types and discrimination settings, the monitors are capable of detecting a range of very small to very large leakage amounts. The first monitor type meets the accuracy and range requirements of Regulatory Guide 1.97 and detects gross gamma radiation emitted during radioactive decay of noble fission product gases, particularly Xe-133. The second monitor type detects N-16 gamma radiation produced during reactor operation.

Located between the ARV and the first SG safety valve of each steam line, Main Steam Line Monitors (u-RE-2325 → u-RE-2328) are Geiger Mueller type detectors. Because direct measurement of the steam beta radiation activity concentration is not possible, Main Steam Line Monitors measure the equivalent gamma radiation dose rates adjacent to the steam line. Gamma radiation is given off during beta minus decay of noble gases such as Xe-133. To prevent area (background) radiation from affecting the detector response and indication, the detector is mounted inside a 3 1/4" lead shield with a thin steel window facing the steam pipe. The major limitation of these monitors is they are not sensitive to small leak rate changes and are therefore limited to post-accident assessment of significant releases. They are only capable of detecting primary-to-secondary leakage beginning at around 3600 gallons per day (2.5 gpm). Two Main Steam Line Monitors interface with a single remote RM-80 microprocessor.

Located just upstream of the MSIV for each steam line, the Steam Generator Leak Rate Monitors (u-RE-2325A → u-RE-2328A) is an insulated, lightly shielded gamma scintillator type detector, sensitive to N-16 at 6.13 MeV. By utilizing the response of the N-16 gamma, this monitor provides early detection of very small amounts of primary-to-secondary leakage in the gallons-per-day (gpd) range. If primary-to-secondary leakage or a Steam Generator Tube Rupture (SGTR) were to occur while the reactor is at power, this monitor provides a rapid and clear indication of a primary-to-secondary leak. Once the reactor trips, production of N-16 essentially ceases, so readouts typically return to their normal levels. This radiation monitor is capable of detecting SG tube leakage ranging from about 1 gallon per day (.0007 gpm) up to 150 gallons per day. Two Steam Generator Leak Rate Monitors interface with a single remote RM-80 microprocessor.

Comments / Reference: From LO21.SYS.SB1 p. 27	Revision 5-7-2011

EFFECTS ON SG BLOWDOWN

The effects of a Steam Generator Tube Leak/Rupture on the SG Blowdown System are significant. If activity levels reach the threshold limit for radiation monitor u-RE-4200, then flow through the system is stopped by the SG Blowdown Isolation Valves closing. This prevents the SG Blowdown System from spreading the radioactive elements to other secondary plant components. Another effect is exhaustion of the resin in the system's demineralizers. The SG Blowdown System's demineralizers utilize resin specifically designed to remove ionic impurities found in the secondary plant water. Introduction of reactor coolant into the blowdown stream provides the demineralizers access to additional ionic impurities and boron. These additional ionic impurities and boric acid cause rapid depletion of the resin. Once the resin is exhausted, radioactivity levels increase in the blowdown stream. When radioactivity levels reach 1.0×10^{-5} µci/cm^3, an annunciator on the Main Control Board warns the operator of a trouble alarm on the SG Blowdown Control Panel. Prior to receiving the annunciator alarm, the PC-11 will sound an audible alarm and provide information of the radiation monitor in "ALERT" status. Shortly after these alarms, flow through the SG Blowdown System isolates when the radiation monitor reaches its "ALARM" setpoint.

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.
- Component Cooling Water Heat Exchanger 1-01 outlet temperature is 95°F and stable.

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. Safety Injection Pump 1-01
6. Safety Chiller 1-05

A. 1, 5, 6

B. 2, 3, 4

C. 2, 4, 6

D. 1, 2, 5

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because CCP 1-01 and SIP 1-01 must be secured and Safety Chiller 1-05 is 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 CCW HX outlet temperature exceeds 122°F.
C. Incorrect. Plausible because EDG 1-01, CTPs 1-01 & 1-03 and Safety Chiller 1-05 is secured if CCWP 1-01 trips.
D. Correct. CCP 1-01, EDG 1-01 and SIP 1-01 must be secured.

Technical Reference(s): ABN-501, Steps 2.3.1 & 2.3.5; ABN-502, Step 2.3.6 — Attached w/ Revision # See 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 # ____
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
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 4 OF 50

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 Operator Actions

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 AND start required equipment as necessary. IF CCW flow is NOT available to an operating RHR HX, THEN perform the following: a. Verify safeguard loop isolation valves open (u-HS-4512, 4513, 4514, and 4515) b. Ensure the idle CCW Pump Recirc Valve closed. • u-HS-4536, CCWP 1 RECIRC VLV • u-HS-4537, CCWP 2 RECIRC VLV c. Align RHR HX return valves as necessary to supply the operating heat exchanger. • u-HS-4572, RHR HX 1 CCW RET VLV • 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:

	RO	SRO
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 complete 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.	Fire Protection Water	Unit 2 Station Service Water
B.	Unit 2 Station Service Water	Fire Protection Water
C.	Fire Protection Water	Demineralized 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 and fire protection water is the 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 ABN-501, Attachment 1	Attached w/ Revision # See 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
STATION SERVICE WATER SYSTEM MALFUNCTION	REVISION NO. 9	PAGE 28 OF 50

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

What is the power supply for Instrument Air Compressor 1-01?

A. 1EB3-1

B. 1EB4-1

C. 1EB3

D. 1EB4

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this is the control power supply to 1-01 Instrument Air Dryer, however, the power supply to Instrument Air Compressor 1-01 is 1EB3.
B. Incorrect. Plausible because this is the control power supply to 1-02 Instrument Air Dryer, however, the power supply to Instrument Air Compressor 1-01 is 1EB3.
C. Correct. This is the power supply to Instrument Air Compressor 1-01.
D. Incorrect. Plausible because this is the power supply to Instrument Air Compressor 1-02.

Technical Reference(s) LO21.SYS.IA1 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 X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5, 7
55.43

Comments / Reference: From LO21.SYS.IA1.LN, Page 23	Revision # 05/07/11

ELECTRICAL

The following table provides the source of power for the major components of the Instrument Air system:

Component	*Component Power Supply*	*Control Power*
1-01 Instrument Air Compressor	1EB3/11D	1EB3/11D
1-02 Instrument Air Compressor	1EB4/11D	1EB4/11D
X-01 Instrument Air Compressor	uB4/11C	uB4/11C
X-02 Instrument Air Compressor	XB1/5C	XB1/5C
2-01 Instrument Air Compressor	2EB3/10C	2EB3/10C
2-02 Instrument Air Compressor	2EB4/10D	2EB4/10D
1-01 Instrument Air Dryer	---	1EB3-1/2BR
1-02 Instrument Air Dryer	---	1EB4-1/12FL
X-01 Instrument Air Dryer	---	XB2-1/3M
X-02 Instrument Air Dryer	---	XB1-6/4RB
2-01 Instrument Air Dryer	---	2EB3-1/2BR
2-02 Instrument Air Dryer	---	2EB4-2/9FL

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 in accordance with 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 – STBY
    - 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 then rose to 111 psig and has been stable for 22 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.	not running	running unloaded	not running
D.	not running	running loaded	not running

Proposed Answer: A

Explanation:

A. Correct. Instrument air header pressure dropping to 83 psig causes IAC 2-02 to load at 105 psig, IAC 2-01 to start and load at 100 psig and IAC X-02 to start and load at 95 psig. All IACs must reach 115 psig before they unload or stop.

B. Incorrect. Plausible if thought that 110 psig is the unloading pressure and that automatic shutdown does not occur until 20 minutes of running unloaded (based on Unit difference).

C. Incorrect. Plausible if thought that 110 psig is the unloading pressure and the LEAD compressor (2-02) would remain running unloaded.

D. Incorrect. Plausible if thought that 110 psig and 20 minutes would stop IACs 2-01 and X-02 with IAC 2-02 running loaded as the LEAD compressor.

Technical Reference(s) LO21.SYS.IA1 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

INSTRUMENT AIR

- **1-01/1- 02 Run Program**
  - Lead - 105 psig (Load)/115 psig (Unload)
  - Backup - 100 psig (Load)/115 psig (Unload)
    - Load Time Delay approx. 25 seconds
    - Run Unloaded 20 minutes (U1)
- **2-01/2-02, X-01/X-02 Air Compressors Setpoints (Elektronikon Control)**
  - Pressure Band 1 (lead)
    - 105 psig (Load)/115 psig (Unload)
  - Pressure Band 2 (backup)
    - 2-01/2-02
      - 100 psig (Load)/115 psig (Unload)
    - X-01/X-02
      - 95 psig (Load)/115 psig (Unload)
    - Unloaded shutdown selectable 0 – 72 times/24 hours

•

Examination Outline Cross-reference:

	RO	SRO
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 1-8880, SI/PORV ACCUM N2 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-FCV-0510, SG 1-01 FW FLO CTRL 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 is a Feedwater Isolation Valve under its own Tech Spec, 3.7.3, which is tied to Tech Spec 3.6.3 in the NOTE “above the line”.

Technical Reference(s): Technical Specification LCO 3.6.3; OPT-408A, Attachment 10.1.1 — 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

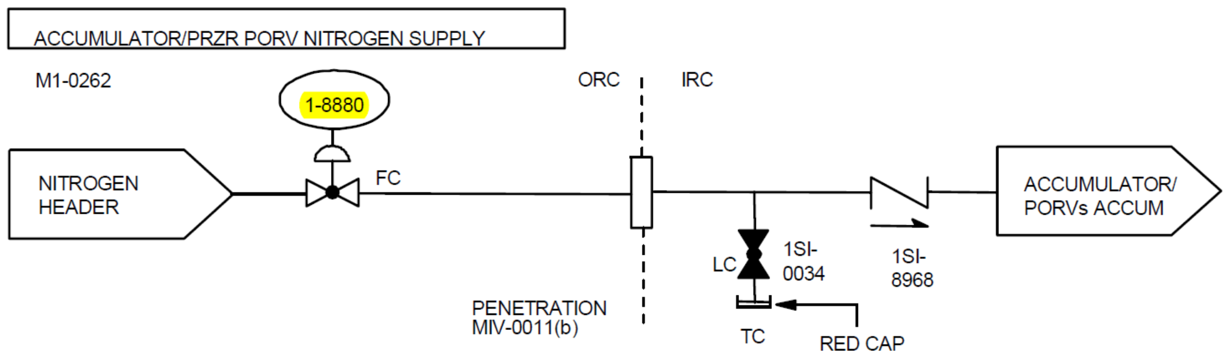
3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

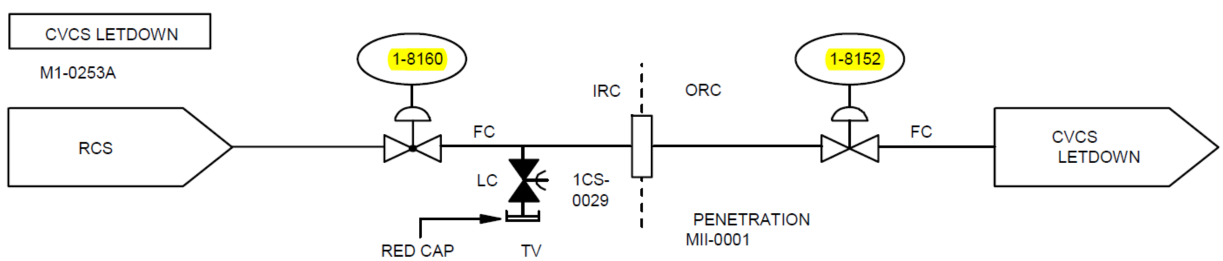
---NOTE---
Not applicable to Main Steam Safety Valves (MSSVs), Main Steam Isolation Valves (MSIVs), Feedwater Isolation Valves (FIVs) and Associated Bypass Valves, and Steam Generator Atmospheric Relief Valves (ARVs).

APPLICABILITY: MODES 1, 2, 3, and 4

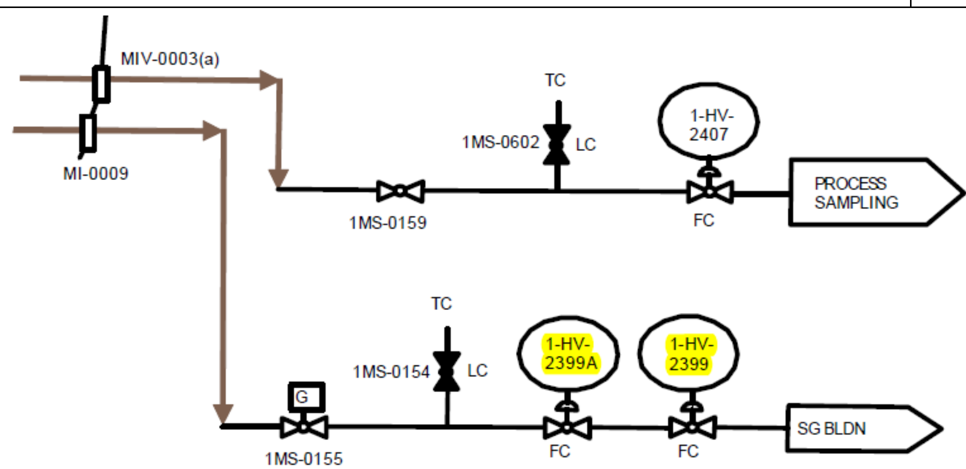
Comments / Reference: From OPT-408A, Attachment 10.1.1, page 11	Revision 11



Comments / Reference: From OPT-408A, Attachment 10.1.1, page 6	Revision 11



Comments / Reference: From OPT-408A, Attachment 10.1.1, page 33	Revision 11



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 open.
  - CRDM GENERATOR LINE VOLTS are 261 volts and stable.

Which of the following describes the effect on the Control Rod Drive System?

The Reactor is [1] due to the loss of 480 Volt Bus [2].

	[1]	[2]
A.	tripped	1B3
B.	not tripped	1B3
C.	tripped	1B4
D.	not tripped	1B4

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if believed that the reactor trips with the loss of one CRDS MG and that 480 V Bus 1B3 powers CRDS MG 1-02.
B. Incorrect. Plausible because the Reactor is not tripped and if believed that that 480 V Bus 1B3 powers CRDS MG 1-02.
C. Incorrect. Plausible because 480 V Bus 1B4 powers CRDS MG 1-02 but the Reactor is not tripped..
D. Correct. Annunciator 1-ALB-10B, Window 4.12 monitors the status of AC power to the CRDM Generators. When Bus 1B4 is tripped, the supply breaker to CRDM Generator 1-02 is also tripped which results in a motor breaker OPEN indication as listed in the Stem. 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; 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:		RO	SRO
	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 predicts the impact on Pressurizer Level Control?

Pressurizer Level will...

A. ...remain stable as Reactor Makeup initiates automatic makeup at 46% and results in VCT level cycling between 46% and 56%; CCP 1-01 maintains Net Positive Suction Head.

B. ...NOT remain stable as Reactor Makeup initiates automatic makeup at 46% but VCT level will continue to lower; CCP 1-01 will lose Net Positive Suction Head as suction will NOT automatically shift to the Refueling Water Storage Tank.

C. ...remain stable as Reactor Makeup does NOT initiate automatic makeup; CCP 1-01 maintains Net Positive Suction Head as suction automatically shifts to the Refueling Water Storage Tank.

D. ...NOT remain stable as Reactor Makeup does NOT initiate automatic makeup; CCP 1-01 will lose Net Positive Suction Head as 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 1-LT-112 level channel lowered to 46% and the cycling setpoint is correct, however, 1-LT-112 will continue to indicate 100% VCT level. Thus automatic makeup will not occur and maintain VCT level in the band.

B. Incorrect. Plausible because automatic makeup would be initiated when 1-LT-112 level channel lowered to 46%, however, charging flow is exceeding automatic makeup capability and therefore VCT level could be thought to continue trending down even with automatic makeup, however, with RCP Seal Return flow included VCT level would not continue to drop if automatic makeup to the VCT was occurring..

C. Incorrect. Plausible because automatic makeup will not be initiated, however, the Centrifugal Charging Pump suction remains aligned to the VCT as both 1-LT-112 and 1-LT-185 must indicate 2% for suction to swap to the RWST and 1-LT-112 is failed to 100%.

D. Correct. As outlined in ALM-0061A, VCT level will not actuate at 46% as 1-LT-112 is failed to 100%. Additionally, CCP suction will not transfer to the RWST as both 1-LT-112 and 1-LT-185 must indicate 2% for suction to swap to the RWST and 1-LT-112 is failed to 100%. CCP 1-01 will lose suction and Pressurizer Level will lower dramatically.

Technical Reference(s) ALM-0061A, 1-ALB-6A, Window 2.5 & 4.5 / LO21.SYS.CS1 — 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 # ______
- 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 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.

Comments / Reference: From LO21.SYS.CS1 LN p. 24, 25	Revision 4/28/11

Table 5 summarizes the functions initiated at various level setpoints on u-LT-0112 and u-LT-0185.

Table 1: VCT Level Transmitter Automatic Functions

LVL (%)	u-LT-0112	u-LT-0185	BOTH u-LT-0112 AND u-LT-0185
98	---	LCV to HUT high-high level alarm	---
70	high level alarm	high level alarm	---
62	modulate LCV	---	---
56	stop auto MU	---	---
46	start auto MU	---	---
16	low level alarm and allow chg realign	low level alarm and allow chg realign	---
2	low-low level alarm	low-low level alarm	---
2	---	---	align charging to RWST

At 62% indicated level on u-LI-0112 (or at a value set on the controller), VCT Level Controller u-LK-0112C (a M/A station on CB-06) will send a signal to an I/P converter, which pneumatically modulates u-LCV-0112A to direct letdown to the recycle holdup tank. The valve will continue to position further to the recycle holdup tank if level increases above setpoint. At 98% level from the other level transmitter, u-LT-0185, a solenoid operated valve directs full air pressure to the valve positioner to fully position u-LCV-0112A to the recycle holdup tank.

The 62% setpoint for initiating diversion of letdown to the recycle holdup tank is based on maintaining a gas volume in the volume control tank sufficient to absorb the liquid which results from a 4EF error in Tave during a 0 to 100% power swing and which is not absorbed by the pressurizer.

Volume Control Tank Level Transmitter u-LT-0112A actuates level bistables which initiate automatic makeup to the volume control tank at 46% level and stop automatic makeup at 56% level.

At 2% level (low-low level) on **both** u-LT-0112 and u-LT-0185, the charging pump suction piping will automatically realign to the refueling water storage tanks. The low-low level realignment opens the parallel charging pump suction valves from the RWST (u-LCV-0112D and u-LCV-0112E) and closes the series charging pump suction valves from the VCT (u-LCV-0112B and u-LCV-0112C). The charging pump suction may be manually realigned after the volume control tank level returns to ≥ 16% on either channel.

Examination Outline Cross-reference:

	RO	SRO
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:

- Unit 1 is at 100% power.
- The 120 VAC Distribution System is in the Normal alignment.
- A failure of 120/208 VAC DISTRIBUTION PANEL 1C1 (CP1-ECDPNC-02) occurs.

What is the effect on the Digital Rod Position Indication (DRPI) system and the procedural action necessary?

A. DRPI is in Half-Accuracy. Energize Train C 120 VAC Distribution Panel 1C14 from CP1-ECDPNC-03, 120/208 VAC DISTRIBUTION PANEL 1C4.

B. Loss of DRPI. Energize Train C 120 VAC Distribution Panel 1C14 from CP1-ECDPNC-03, 120/208 VAC DISTRIBUTION PANEL 1C4.

C. DRPI is in Half-Accuracy. Energize Train C 120 VAC Distribution Panel 1C14 from CPX-EPDPNB-03 120/208 VAC MISCELLANEOUS POWER PANEL XC4-4.

D. Loss of DRPI. Energize Train C 120 VAC Distribution Panel 1C14 from CPX-EPDPNB-03 120/208 VAC MISCELLANEOUS POWER PANEL XC4-4.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if believed that Data A was powered from 1C1 and Data B was powered from 1C14 during normal alignment, however all of DRPI is powered from 1C14 which is normally powered from 1C1.
B. Correct. As DRPI is powered from 1C14, which is normally aligned from 1C1, a loss of 1C1 would result in a loss of DRPI. In accordance with ABN-712, the power supply should be shifted to 1C4.
C. Incorrect. Plausible if believed that Data A was powered from 1C1 and Data B was powered from 1C14 during normal alignment, however all of DRPI is powered from 1C14 which is normally powered from 1C1. Further XC4-4 could be thought to be the alternate power supply to 1C14.
D. Incorrect. Plausible as this is the correct response of DRPI, however, this is the incorrect alternate power supply and XC4-4 could be thought to be the alternate power supply to 1C14.

Technical Reference(s) ABN-712, Sections 4.3 / SOP-608A, Section 5.1.3 & 5.1.4 — 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 Rod Control Indication and Rod Insertion Limit (RIL) Monitor Systems.

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

Question History: Last NRC Exam ______

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

10 CFR Part 55 Content: 55.41 6, 10
55.43 ______

Comments / Reference: From SOP-608A 5.1.4	Revision # 12-1

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-608A
120, 208, 208/120 AND 120/240 VAC DISTRIBUTION SYSTEM	REVISION NO. 12	PAGE 19 OF 77
	CONTINUOUS USE	

5.1.4 Placing Unit 1 Train C 120 VAC Distribution Panel 1C14 in service.

The following describes the steps necessary for placing the 120 VAC Distribution Panel 1C14 in service.

☐ A. ENSURE the prerequisites of Section 2.1 are met.

B. ENSURE one or both 208/120 VAC supply distribution panels are energized to feed 1C14.

☐ • CP1-ECDPNC-02, 120/208 VAC DISTRIBUTION PANEL 1C1

☐ • CP1-ECDPNC-03, 120/208 VAC DISTRIBUTION PANEL 1C4

☐ C. ENSURE 1C14 has the distribution breakers placed in the OFF position.

D. At Distribution Panel 1C14, PLACE one of the supply breakers to the ON position. The other breaker shall be in the OFF position. (Mechanically Interlocked)

☐ • 1C14/00/BKR-1, 1C1 TO 120 VAC DISTRIBUTION PANEL 1C14 PREFERRED FEEDER BREAKER

☐ • 1C14/00/BKR-2, 1C4 TO 120 VAC DISTRIBUTION PANEL 1C14 ALTERNATE FEEDER BREAKER

E. PLACE one OR both of the AC feeder breaker(s) for 1C14 to the ON position:

☐ • 1C1/7/BKR, 120 VAC DISTRIBUTION PANEL 1C14 PREFERRED FEEDER BREAKER

☐ • 1C4/12/BKR, 120 VAC DISTRIBUTION PANEL 1C14 ALTERNATE FEEDER BREAKER

Comments / Reference: From ABN-712, Section 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.

Comments / Reference: From SOP-608A 5.1.3	Revision # 12-1

CPNPP SYSTEM OPERATING PROCEDURE MANUAL	UNIT 1	PROCEDURE NO. SOP-608A
120, 208, 208/120 AND 120/240 VAC DISTRIBUTION SYSTEM	REVISION NO. 12 CONTINUOUS USE	PAGE 15 OF 77

5.1.3 Placing the Unit 1 and Common Train C 120, 208/120 and 208 VAC Distribution Panels (1C1, 1C4, 1C2-4, 1C5-1, 1C6-1, XC1-2, XC2-2, XC4-4, XC10-1-1, XC10-1-2, and XC10-1-3) in service.

The following describes the steps necessary for placing the 120, 208/120 and 208 VAC Distribution Panel(s) 1C1, 1C4, 1C5-1, 1C6-1, XC1-2, XC2-2, XC4-4, XC10-1-1, XC10-1-2 and XC10-1-3 in service.

A. SELECT the distribution panel(s) to be placed in service.

☐ • 120/208 VAC DISTRIBUTION PANEL 1C1 (CP1-ECDPNC-02)

☐ • 120/208 VAC DISTRIBUTION PANEL 1C4 (CP1-ECDPNC-03)

☐ • 120/208 VAC CHEM LAB POWER PANEL 1C2-4 (CP1-EPDPNC-10)

☐ • 208 VAC DISTRIBUTION PANEL 1C5-1 (CP1-ECDPNC-15)

☐ • 208 VAC DISTRIBUTION PANEL 1C6-1 (CP1-ECDPNC-14)

☐ • 120 VAC DISTRIBUTION PANEL XC1-2 (CPX-ECDPNC-07)

☐ • 120/208 VAC REGULATED DISTRIBUTION PANEL XC2-2 (CPX-EPDPNC-02)

☐ • 120/208 VAC MISCELLANEOUS POWER PANEL XC4-4 (CPX-EPDPNB-03)

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 conditions on Unit 1:

- A Small Break Loss of Coolant Accident has occurred.
- Reactor Coolant Pumps have been tripped.
- Reactor Vessel Level indicating System (RVLIS) currently indicates 11 and 22 inches above core plate light LIT. All other RVLIS lights DARK.
- The Emergency Core Cooling System is injecting
- Reactor Coolant System Wide Range Pressure is 1385 psig.
- All Wide Range Hot Leg Temperatures indicate 587ºF.
- All Core Exit Thermocouples indicate between 570ºF and 575ºF.

Which of the following correctly states the fluid properties of the water cooling the top of the core?

The top of the core is being cooled by…

A. …saturated liquid.

B. …superheated steam.

C. …saturated steam.

D. …subcooled liquid.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if calculation is performed using hot leg temperature of 587°F and believed that Quality is 0.
B. Incorrect. Plausible if thought that the saturated conditions in the hot leg were indicative of superheated conditions at the hotter core outlet.
C. Incorrect. Plausible if thought that the saturated conditions in the hot legs were indicative of saturated conditions at the core outlet, however, as the fluid level is below the hot legs only steam is being carried over to the hot leg RTDs.
D. Correct. The Core Exit Thermocouples indicate 12 to 17°F subcooled.

Technical Reference(s) LO21.SYS.RC3 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 # ______
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, 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.

Comments / Reference: From Steam Tables

		Specific Volume		Internal Energy			Enthalpy			Entropy		
Press. Lbf./Sq.In. p	Temp. Fahr. t	Sat. Liquid v_f	Sat. Vapor v_g	Sat. Liquid u_f	Evap. u_{fg}	Sat. Vapor u_g	Sat. Liquid h_f	Evap. h_{fg}	Sat. Vapor h_g	Sat. Liquid s_f	Evap. s_{fg}	Sat. Vapor s_g
800	518.36	.02087	.5691	506.6	608.4	1115.0	509.7	689.6	1199.3	.7110	.7050	1.4160
820	521.21	.02094	.5542	510.0	604.6	1114.6	513.2	685.5	1198.7	.7145	.6988	1.4133
840	524.01	.02101	.5400	513.3	600.8	1114.1	516.6	681.5	1198.0	.7179	.6927	1.4106
860	526.76	.02109	.5264	516.6	597.0	1113.6	519.9	677.5	1197.4	.7212	.6867	1.4080
880	529.46	.02116	.5134	519.8	593.3	1113.1	523.3	673.5	1196.7	.7245	.6808	1.4053
900	532.12	.02123	.5009	523.0	589.6	1112.6	526.6	669.5	1196.0	.7277	.6750	1.4027
920	534.73	.02130	.4890	526.2	585.9	1112.1	529.8	665.6	1195.4	.7309	.6693	1.4002
940	537.29	.02137	.4776	529.3	582.3	1111.6	533.0	661.6	1194.6	.7341	.6636	1.3977
960	539.82	.02145	.4666	532.4	578.7	1111.0	536.2	657.7	1193.9	.7372	.6580	1.3952
980	542.30	.02152	.4561	535.4	575.1	1110.4	539.3	653.9	1193.2	.7402	.6525	1.3927
1000	544.75	.02159	.4459	538.4	571.5	1109.9	542.4	650.0	1192.4	.7432	.6471	1.3903
1050	550.71	.02177	.4222	545.7	562.6	1108.4	550.0	640.4	1190.4	.7505	.6338	1.3844
1100	556.45	.02195	.4005	552.9	553.9	1106.8	557.4	631.0	1188.3	.7576	.6209	1.3786
1150	562.00	.02214	.3806	559.9	545.3	1105.2	564.6	621.6	1186.2	.7645	.6084	1.3729
1200	567.37	.02232	.3623	566.7	536.8	1103.5	571.7	612.3	1183.9	.7712	.5961	1.3673
1250	572.56	.02250	.3454	573.4	528.3	1101.7	578.6	603.0	1181.6	.7778	.5841	1.3619
1300	577.60	.02269	.3297	579.9	519.9	1099.8	585.4	593.8	1179.2	.7841	.5724	1.3565
1350	582.50	.02288	.3152	586.3	511.6	1097.9	592.1	584.6	1176.7	.7903	.5609	1.3513
1400	587.25	.02307	.3016	592.7	503.3	1096.0	598.6	575.5	1174.1	.7964	.5497	1.3461
1450	591.88	.02326	.2888	598.9	495.1	1093.9	605.1	566.3	1171.4	.8024	.5385	1.3409

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 480 VAC Motor Control Centers supply power to the Containment Pre-Access Filtration System Fans?

	Containment Pre-Access Filtration Fan 1-01	Containment Pre-Access Filtration Fan 1-02
A.	1B1-2	1B2-1
B.	XB1-2	XB2-2
C.	XEB2-1	XEB2-2
D.	1EB1-2	1EB2-2

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if thought that fans are powered from a unit non-safeguards power supply because they are normally operated prior to a containment entry to reduce radiation exposure.
B. Incorrect. Plausible if thought that fans are powered from a common non-safeguards power supply because they are normally operated prior to a containment entry to reduce radiation exposure.
C. Incorrect. Plausible if thought that the fans are powered from a common safeguards MCC because they could be used for post accident iodine removal.
D. Correct. Train A fan is powered from MCC 1EB1-2 and Train B from MCC 1EB2-2.

Technical Reference(s) EOP-0.0A, Attachment 8 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 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)
☐ **	• 1EB1 TO 480 VAC MCC XEB1-3 FEEDER BREAKER (MCC located in AUX Bldg 810 U2 Side N. End East Hall)	MCCXEB1-3 (1EB1/5B/BKR)
☐	• 1EB1 TO 480 VAC MCC 1EB1-3 FEEDER BREAKER (MCC located in SFGDs Bldg 790 South hallway)	MCC1EB1-3 (1EB1/5D/BKR)
☐	• 1EB1 TO 480 VAC MCC 1EB1-2 FEEDER BREAKER (MCC located in SFGDs Bldg 810 Train A Swgr)	MCC1EB1-2 (1EB1/5C/BKR)

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

1.4 LOAD SHEDDING COMPLETE 1EB2 & 1EB4 (SFGDs 852 Train B Swgr)

1EB2

☐	• PRZR BACKUP HTR GROUP B	1/1-PCPR2 (CB-05)
☐	• CNTMT FN CLR FN 2	1-HS-5409A (CB-03)
☐	• 1EB2 TO 480 VAC MCC 1EB2-3 FEEDER BREAKER (MCC located in AUX Bldg 832 N. End East Hall	MCC1EB2-3 (1EB2/5D/BKR)
☐	• 1EB2 TO 480 VAC MCC 1EB2-2 FEEDER BREAKER (MCC located in SFGD Bldg 852 Train B Swgr Rm)	MCC1EB2-2 (1EB2/5C/BKR)

Examination Outline Cross-reference:

	RO	SRO
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, when the Equipment Hatch is off and the Personnel Airlock is open then 1-HS-5572 is closed to ensure that air flow is from the Safeguards Building into the Containment. This is contrary to the normal pressure within the Safeguards Building during MODES 1 through 4 when it is maintained at a slight negative pressure to ensure contaminants can be monitored via the Primary Plant Ventilation System.

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; SOP-816, Section 4.1 — 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 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.

Comments / Reference: From SOP-816, Section 4.1	Revision # 13

CPNPP SYSTEM OPERATING PROCEDURES MANUAL	UNIT COMMON	PROCEDURE NO. SOP-816
PRIMARY PLANT VENTILATION SYSTEM	REVISION NO. 13 CONTINUOUS USE	PAGE 4 OF 33

3.0 PRECAUTIONS

- If avoidable, ESF Filter Units should not be run during normal operation.

[C] • The supply fans are electrically interlocked with two exhaust fans, with the exception of supply fan 23 and 24 which are not interlocked with any exhaust fans. Two exhaust fans must be running to run a supply fan. IF exhaust fans are lost or tripped, THEN the corresponding supply fans should also be stopped.

- Re-alignment of the Primary Plant Ventilation System may cause changes in differential pressure between the Containment and the Fuel Building. When the transfer tube is open, differential pressure will cause level changes between the Spent Fuel Pools and the Refueling Cavity.

4.0 LIMITATIONS AND NOTES

4.1 Limitations

- Two independent ESF Filtration Trains shall be operable in Modes 1, 2, 3, and 4 (TS 3.7.12). If one of the two ESF filtration units in a train can maintain the required negative pressure alone, it would satisfy the operability requirement.

[C] • The Auxiliary, Safeguards and Fuel Buildings shall be maintained at a slight negative pressure in MODES 1, 2, 3 AND 4 (TS 3.7.12) (TE-91-2558).

- Refer to VFTP, following painting, fires or chemical release in any ventilation zone communicating with the ESF Trains of the Primary Plant Ventilation System (TE-SG-90-689).

- When the containment is being purged per SOP-801A or SOP-801B, two additional exhaust units and one additional supply unit may need to be started.

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.

What is the Steam Generator level setpoint for Auxiliary Feedwater Pump start and why is Auxiliary Feedwater automatically initiated?

A. ● 35.4% Narrow Range Level
 ● Provide a secondary heat sink.

B. ● 35.4% Narrow Range Level
 ● Prevent Steam Generator dryout.

C. ● 38% Narrow Range Level
 ● Provide a secondary heat sink.

D. ● 38% Narrow Range Level
 ● Prevent Steam Generator dryout.

Proposed Answer: A

Explanation:

A. Correct. Steam Generator level below 35.4% (U-2) initiates Auxiliary Feedwater to provide secondary heat removal.

B. Incorrect. Plausible because the Steam Generator level setpoint is correct but keeping the SG components “wet” is why a minimum of 100 gpm to each SG is required in ECA-2.1B.

C. Incorrect. Plausible because 38% is the Unit 1 setpoint which initiates Auxiliary Feedwater to provide secondary heat removal.

D. Incorrect. Plausible because 38% is the Unit 1 setpoint is correct but keeping the SG components “wet” is why a minimum of 100 gpm to each SG is required in ECA-2.1B.

Technical Reference(s) LO21.SYS.AF1 — Attached w/ Revision # See Comments / Reference
ECA-2.1B
Technical Specification 3.3.2 Bases

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 ____

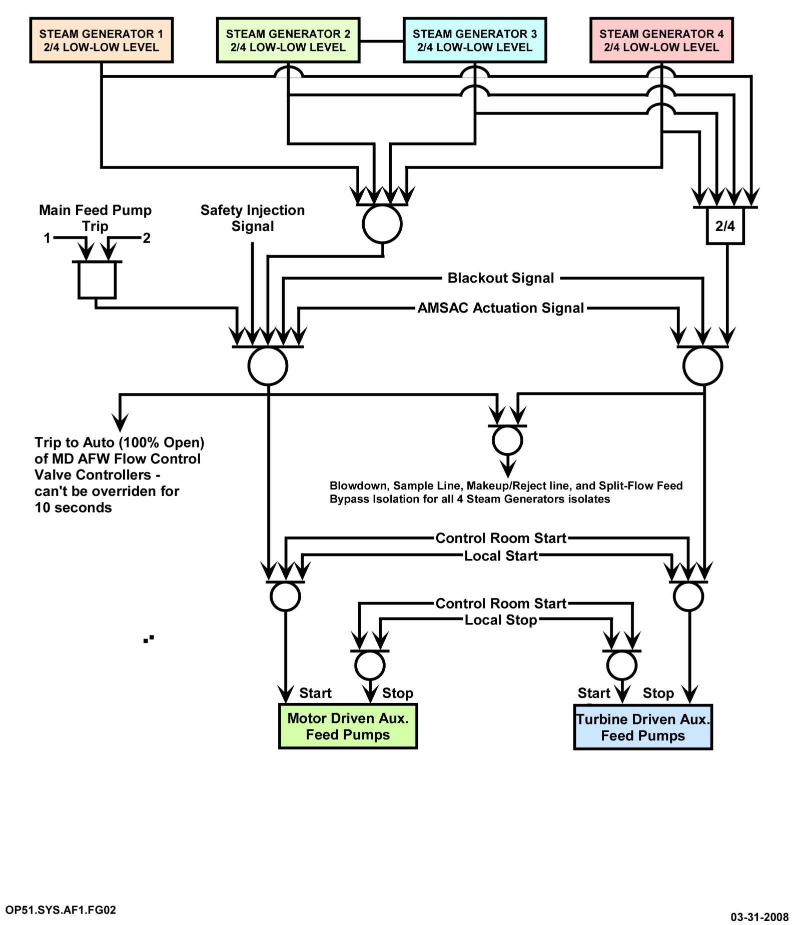
Comments / Reference: From LO21.SYS.AF1.LN, Page 11	Revision 05/11/11

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



Comments / Reference: From ECA-2.1B, Attachment 4	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. ECA-2.1B
UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS	REVISION NO. 8	PAGE 42 OF 71

ATTACHMENT 4
PAGE 2 OF 31

BASES

CAUTION: If AFW flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable AFW flow to the SG allows the components to remain in a "wet" condition, thereby minimizing any thermal shock effects if feed flow is increased.

Comments / Reference: From Tech Spec 3.3.2 Bases	Revision 67

ESFAS Instrumentation
B 3.3.2

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

a. Auxiliary Feedwater - Automatic Actuation Logic and Actuation Relays (Solid State Protection System)

Automatic actuation logic and actuation relays consist of the similar features and operate in the similar manner as described for ESFAS Function 1.b.

b. Not used.

c. Auxiliary Feedwater - Steam Generator Water Level-Low Low

SG Water Level-Low Low provides protection against a loss of heat sink. A feed line break, inside or outside of containment, or a loss of MFW, would result in a loss of SG water level. SG Water Level-Low Low provides input to the SG Level Control System. Therefore, the actuation logic must be able to withstand both an input failure to the control system which may then require a protection function actuation and a single failure in the other channels providing the protection function actuation. Thus, four OPERABLE channels are required to satisfy the requirements with two-out-of-four logic. Two-out-of-four low-low level signals in any SG starts the motor-driven AFW pumps; in two or more SGs starts the turbine-driven AFW pump.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045 A3.05	
	Importance Rating	2.6	

Main Turbine Generator System: 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.

What is the status of the LP Turbine Control Valves and the HP Turbine Control Valves at this point in Main Turbine startup?

	LP Turbine Control Valves	HP Turbine Control Valves
A.	FULLY OPEN	FULLY OPEN
B.	NOT FULLY OPEN	FULLY OPEN
C.	FULLY OPEN	NOT FULLY OPEN
D.	NOT FULLY OPEN	NOT FULLY OPEN

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that latching the turbine and raising speed to 1800 rpm (full speed) would open all control valves.

B. Incorrect. Plausible if thought that the HP vice the LP control valves would be fully open on turbine startup and the LP control valves throttle to control speed and the throttle to control load once synchronized to the Grid.

C. Correct. Given the conditions listed and with no operator action once the Main Turbine reaches 1800 RPM the LP control valves will be fully open and HP control valves throttle to control speed and will throttle to control load once synchronized to the Grid.

D. Incorrect. Plausible if thought that both the LP and HP control valves throttle to control load.

Technical Reference(s) LO21.SYS.MT1 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 LO21.SYS.MT1	Revision 5/4/11

The Turbine-Generator load is controlled by stop and control valves. The stop valves admit steam to the turbine as operating conditions require, and provide extremely fast closure to isolate the turbine from the steam supply system in the event of a protective device actuation. This rapid closure of the stop valves is referred to as a turbine trip. The control valves regulate the amount of steam admitted to the Main Turbine to control the output load of the Main Generator or the speed of the turbine during startup.

Steam flow to the LP Turbines is regulated by the EHC system which uses hydraulic fluid pressure to position control valves in response to various inputs much like the HP Stop and Control valves. The LP Stop and Control valves are normally fully open with Turbine-Generator load being controlled by the HP Turbine.

During normal operation the LP Control Valves are fully open and will only throttle down upon a large loss of electrical load to help prevent overspeeding of the Main Turbine.

Examination Outline Cross-reference:	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 Steam Generator tube leak is in progress.
- 1-RE-2959 (COG-182), CONDENSER OFF GAS Radiation Monitor is in service.
- Two Condenser Exhausting Vacuum Pumps are in service.
- A rapid plant shutdown is in progress.

Which of the following describes the flow path of fission products from the condenser?

A. A portion of the fission products will be monitored by 1-RE-2959 before being transmitted to the Primary Plant Ventilation System.

B. All of the fission products will be monitored by 1-RE-2959 before being transmitted to the Primary Plant Ventilation System.

C. A portion of the fission products will be monitored by 1-RE-2959 before being transmitted to the Waste Gas System.

D. All of the fission products will be monitored by 1-RE-2959 before being transmitted to the Waste Gas System.

Proposed Answer: A

Explanation:

A. Correct. A bypass line from the main 10” discharge header goes to the Process Monitor. Once sampled at the monitor the bypass line rejoins the discharge header and is routed to the Primary Plant Ventilation System for a filtered and monitored discharge to the environment.

B. Incorrect. Plausible if believed that the entire discharge from the CEVs was monitored by 1-RE-2959, however only a portion of the discharge is monitored.

C. Incorrect. Plausible as only a portion of the fission products will be monitored, however, the CEV discharge is routed to the Primary Plant Ventilation System and cannot be routed to the Waste Gas System for storage and later discharge.

D. Incorrect. Plausible if believed that the entire discharge from the CEVs was monitored by 1-RE-2959, and if believed that the CEV discharge can be routed to the Waste Gas System.

Technical Reference(s) LO21.SYS.CV1 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 # ______
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, 7, 14
55.43 ______

Comments / Reference: From LO21.SYS.CV1.LN, Page 8	Revision # 05/25/11

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:

	RO	SRO
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: For this question we assumed the Reactor Trip status panel is equivalent to our First Out Panel.

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

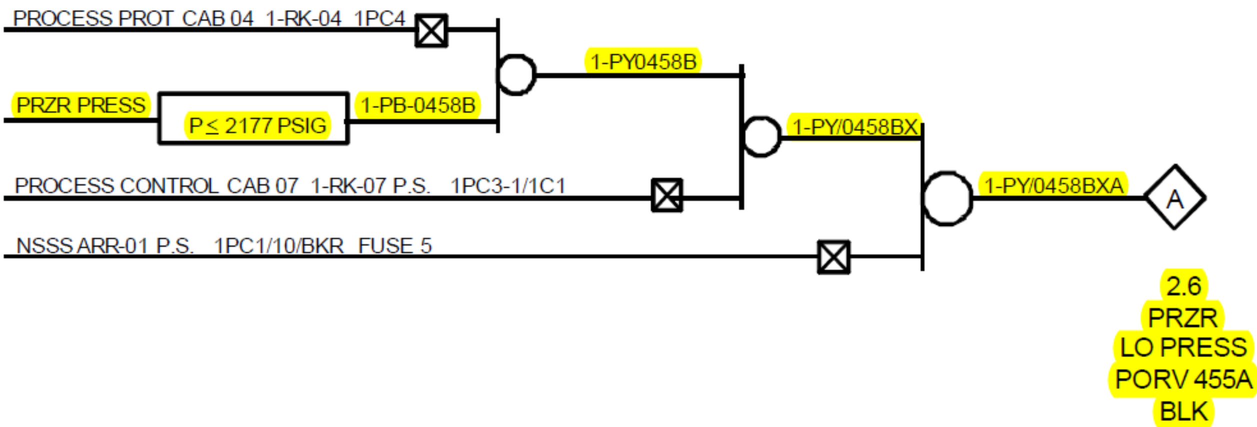
10 CFR Part 55 Content:
55.41 7
55.43

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:



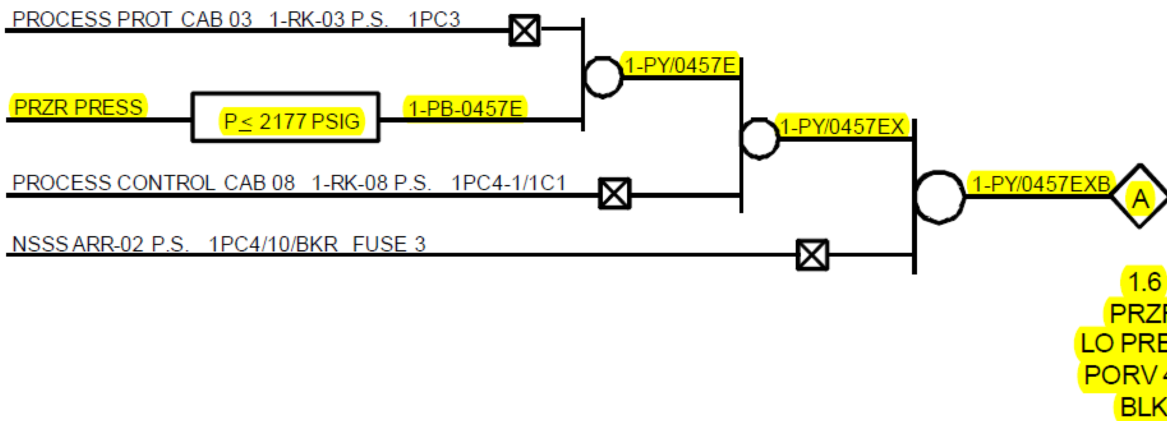
NOTE: The PORV permissive is enabled >2185 psig (increasing), but the alarm comes on <2177 psig (decreasing), this allows for a 1% deadband for the bistable.

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 Steam Dump Valves should be closed farther.

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 result 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 due to loss of heat input to SG 2-02; 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, SG shrink due to loss of heat input to SG 2-02.
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 initially shrink due to loss of heat input to SG 2-02.
D. Incorrect. Plausible because the Reactor will not trip, however, SG level will shrink due to loss of heat input to SG 2-02.

Technical Reference(s) ABN-101, Section 2.2 Attached w/ Revision # See Comments / Reference
ABN-101, Step 2.3.1 NOTE

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	

Loss of RHR System: 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.
- A transition has been made to EOS-1.3A, Transfer to Cold Leg Recirculation.
- Emergency Core Cooling System (ECCS) has been transferred to Cold Leg Recirculation and the RWST level is 8% and lowering.
- The following ECCS pumps are running:
    - Centrifugal Charging Pumps (CCP) 1-01 and 1-02.
    - Safety Injection Pumps (SIP) 1-01 and 1-02.
    - Residual Heat Removal Pumps (RHRP) 1-01 and 1-02.
- RHRP 1-01 is cavitating.

Which of the following lists the required action and reason for stopping the ECCS pumps in accordance with EOS-1.3A, Transfer to Cold Leg Recirculation?

Stop…

A. …SIP 1-01 to improve NPSH for RHRP 1-01.

B. …CCP 1-01 to improve NPSH for RHRP 1-01.

C. …RHRP 1-01 to prevent pump damage.

D. …CCP 1-01 to prevent pump damage.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible as securing SIP 1-01 would cause the greatest improvement to RHRP 1-01 NPSH, however RHRP 1-01 should be secured due to cavitation.
B. Incorrect. Plausible as securing CCP 1-01 would cause some improvement to RHRP 1-01 NPSH, however RHRP 1-01 should be secured due to cavitation.
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 to prevent pump damage and the suction supply to the SIPs and CCPs is cross-connected so that either RHRP will provide adequate suction. The 8804 valves in the RHR system and the 8807 valves in the SI system provide this dual suction source alignment.
D. Incorrect. Plausible since CCP 1-01 is most susceptible to damage due to cavitation and if RHRP 1-01 were the only suction source to CCP 1-01 this action would be appropriate. However, there is no indication of CCP 1-01 cavitation.

Technical Reference(s) EOS-1.3A, Step 3 CAUTION Bases; LO21.SYS.RH1; LO21.SYS.SI1 — 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 Bases	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOS-1.3A
TRANSFER TO COLD LEG RECIRCULATION	REVISION NO. 8	PAGE 38 OF 53

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BASES

Upon receipt of an RWST EMPTY alarm, the immediate concern is adequate suction to the ECCS pumps. The suction line from the RWST to the CCPs is higher than the SI sparger and presents the most limiting NPSH and vortexing concern.

Stopping the CCPs is the highest priority and should be performed as soon as possible, after receiving the RWST empty alarm.

The intent of the level at which the containment spray pumps are stopped is to maximize use of RWST water, which can be indicated by 0% indication or evidence of pump cavitation. At 0% indication any spray pump taking suction from the RWST is stopped until the suction from the tank can be isolated.

Operator is expected to restart secured ECCS & Containment Spray pumps when adequate suction source has been reestablished.

CAUTION: When any pump loses its suction source, operators should stop the pump to prevent potential pump damage. Following transfer to recirculation the potential exists for loss of pump suction due to recirculation sump screen blockage, and the operators must be aware of the indications for cavitation. Symptoms of pump cavitation provide an indirect indication of sump blockage. Indications are available for monitoring cavitation following establishment of recirculation flow:

EXAMPLE: ECCS: RHR pump current,RHR pump discharge and suction pressure, RHR injection flow, SI pump discharge flow, SI pump discharge pressure,CCP SI flow. Containment Spray System: Containment Spray pump suction and discharge pressure,Containment Spray pump flow.

Comments / Reference: From LO21.SYS.RH1 Lesson Notes	Revision 10/21/11
RHR PUMP TO CENTRIFUGAL CHARGING PUMP / SAFETY INJECTION PUMP SUCTION ISOLATION VALVES (u-8804A&B) The RHR Pump to Centrifugal Charging Pump / Safety Injection Pump Suction Isolation Valves function to align one of the discharge paths of the RHR Pumps to the suction piping of the Centrifugal Charging Pumps and Safety Injection Pumps. u-8804A provides flow to the suction of the Centrifugal Charging Pump and u-8804B provides flow to the suction of the Safety Injection Pumps. This path is used when emergency operating procedures require the Emergency Core Cooling System to be placed in cold leg or hot leg recirculation mode. u-8804A&B are motor-operated valves controlled from CB-04.	

Comments / Reference: From LO21.SYS.SI1 Lesson Notes	Revision 5/2/11
Safety Injection Pump/Centrifugal Charging Pump Suction Header Cross-Tie Isolation Valves (u-8807A & B) The Safety Injection Pump/Centrifugal Charging Pump Suction Header Cross-Tie Valves allow either of the Residual Heat Removal Pumps to deliver flow to the suction of the Centrifugal Charging and Safety Injection Pumps during the long-term recirculation mode. The valves are arranged in parallel to assure that at least one valve can be opened to provide a flowpath between the pump suction headers. u-8807A & B are normally left in the closed position, and are opened by manual operation from the Main Control Board when transferring the Emergency Core Cooling System to Cold Leg Recirculation. The valves are located in the Train A ECCS Valve Room on the 790' level of the SFGD □s Bldg.	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027 AK2.03	
	Importance Rating	2.6	

Pressurizer Pressure Control Malfunction: Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners.

Proposed Question: Common 44

Given the following conditions:

- Unit 2 is in MODE 1
- All Pressurizer Backup Heaters have failed ON and cannot be secured.

What is the plant response?

A. Both Pressurizer spray valves open to restore pressure between 2220 psig and 2250 psig.

B. Only one Pressurizer spray valve opens to restore pressure between 2220 psig and 2250 psig.

C. Both Pressurizer Power Operated Relief Valves open to maintain pressure below 2335 psig.

D. Only one Pressurizer Power Operated Relief Valve opens to maintain pressure below 2335 psig.

Proposed Answer: A

Explanation:

A. Correct. With all backup heaters energized, both pressurizer spray valves will open. Pressurizer pressure will initially peak at approximately 2270 psig and then lower to the normal control band.

B. Incorrect. Plausible if it is believed that a single pressurizer spray valve is sufficient to overcome the energy added by all backup heaters.

C. Incorrect. Plausible if it believed that both pressurizer spray valves are not sufficient to mitigate the energy added by all backup heaters.

D. Incorrect. Plausible if it believed that both pressurizer spray valves are not sufficient to mitigate the energy added by all backup heaters and both PORVs open at the same setpoint.

Technical Reference(s)		Attached w/ Revision # See Comments / Reference
	LO21.SYS.PP1	
	DBD-ME-250 Reactor Coolant System	
	ABN-705	

Proposed references to be provided during examination: None

Learning Objective: Analyze the response to a pressurizer pressure 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	7	
	55.43		

Comments / Reference: From LO21.SYS.PP1	Revision 5/5/11
During normal plant operations, the pressurizer is filled with boiling water and steam. The temperature of the boiling (or saturated) water determines the pressure inside the entire Reactor Coolant System. Pressurizer temperature is controlled to regulate RCS pressure by energizing electric heaters in the bottom of the pressurizer to raise pressure, and by spraying the steam space (or steam bubble) in the top of the pressurizer with cooler water to reduce pressure. Under normal operating conditions, the Pressurizer Pressure Control System will automatically maintain the plant at 2235 psig. Heaters maintain a saturated condition in the pressurizer and spray valves throttle open to hold pressure at the 2235 psig setpoint. Backup banks of heaters energize on decreasing RCS pressure. On increasing pressure, spray valves open automatically to cause partial steam bubble condensation.	

Comments / Reference: From DBD-ME-250 Reactor Coolant System	Revision 45
The pressurizer spray valves are required to pass the maximum cold leg spray flow sufficient to maintain the system pressure below the power operated relief valve setpoint for a 12 percent load rejection at 100 percent power. The pressurizer spray line is required to deliver 900 gpm to the pressurizer. RCS pressure is controlled by the use of the pressurizer where water and steam are maintained in equilibrium by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to reduce pressure variations due to contraction and expansion of the reactor coolant.	

Comments / Reference: From ABN-705, Section 3.0	Revision 12

This reference is included to indicate that spray flow will overcome heater input if a spray valve were to fail open. So that indicates that if heaters were to all come on that spray flow would overcome heater input and restore pressure to the control band.

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PRESSURIZER PRESSURE MALFUNCTION	REVISION NO. 12	PAGE 8 OF 26

3.0 Pressurizer Spray Valve Failure

3.1 Symptoms

a. Annunciator Alarms

- PRZR 1 OF 4 PRESS LO (5B-3.4)
- PRZR 1 OF 4 SI PRESS LO (5B-4.4)
- PRZR LO PRESS PORV 456 BLK (5B-1.6)
- PRZR LO PRESS PORV 455A BLK (5B-2.6)

b. Plant Indication

- Spray valve indicated open when not called for by master controller.

3.2 Automatic Actions

a. Control response for failed open spray valve(s)

1) Control and backup heaters come on.

- 1/u-PCPR, PRZR CTRL HTR GROUP C
- 1/u-PCPR1, PRZR BACKUP HTR GROUP A
- 1/u-PCPR2, PRZR BACKUP HTR GROUP B
- 1/u-PCPR3, PRZR BACKUP HTR GROUP D

NOTE: A reactor trip at high power and low pressure may result in an SI due to AFW flow.

2) Reactor trip at 1880 psig.

3) Safety Injection at 1820 psig.

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:

- EOP-3.0A, Steam Generator Tube Rupture (SGTR), is in progress.
- A Loss of Offsite Power (LOOP) occurs.

Assuming a full crew compliment is available which of the following is the anticipated response strategy?

A. Continue in EOP-3.0A, while the LOOP is addressed via the Abnormal Conditions Procedures (ABNs).

B. Suspend actions of EOP-3.0A and immediately enter ECA-0.0A, Loss of All AC Power.

C. Continue in EOP-3.0A and immediately enter ECA-0.0A, Loss of All AC Power.

D. Suspend actions of EOP-3.0A, while the LOOP is addressed via the Abnormal Conditions Procedures (ABNs).

Proposed Answer: A

Explanation:

A. Correct. Crew should continue to address the tube rupture in EOP-3.0A. The crew should address the LOOP using Abnormal Conditions Procedures (ABNs) in accordance with step 33 of EOP-3.0A.
B. Incorrect. Plausible if it is believed that a LOOP is a Loss of All AC Power. However, the Emergency Diesel Generators will power the safeguards buses and the LOOP will be addressed by ABN-601 in accordance with step 33 of EOP-3.0A.
C. Incorrect. Plausible if it is believed that a LOOP is a Loss of All AC Power and that the actions of EOPs are performed concurrently with ECA-0.0A. However, the Emergency Diesel Generators will power the safeguards buses and the LOOP will be addressed by ABN-601 in accordance with step 33 of EOP-3.0A and on an actual loss of all AC, actions in accordance with EOP-3.0A would be suspended while the crew performed ECA-0.0A
D. Incorrect. Plausible if it is believed that actions to restore offsite power must be performed prior to completing actions of EOP-3.0A. However, the LOOP will be addressed by ABN-601 in accordance with step 33 of EOP-3.0A.

Technical Reference(s)	EOP-3.0A, Step 33	Attached w/ Revision # See Comments / Reference
	ODA-407, Attachment 8.A	

Proposed references to be provided during examination: None

Learning Objective: **ANALYZE** the diagnostic steps of 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	10	
	55.43		

Comments / Reference: From EOP-3.0A, Step 33	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-3.0A
STEAM GENERATOR TUBE RUPTURE	REVISION NO. 8	PAGE 28 OF 103

STEP — ACTION/EXPECTED RESPONSE — RESPONSE NOT OBTAINED

<u>NOTE</u>: The ruptured loop Cold Leg Wide Range Temperature indication may now be considered in the INTEGRITY STATUS TREE (FRP) priority.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*33	Check If Diesel Generators Should Be Stopped:	
	a. Verify AC safeguard busses - ENERGIZED BY OFFSITE POWER	a. Restore offsite power to AC busses 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.

Comments / Reference: From ODA-407, Attachment 8.A, page 3	Revision 15

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OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 15 INFORMATION USE	PAGE 21 OF 56

ATTACHMENT 8.A
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ERG RULES OF USAGE

5. C. Some steps may be best facilitated for performance by use of an attachment vice the normal verbal performance of the ERG step(s). In these cases, a unique symbol for each attachment is provided next to the ERG step(s) number to identify the existence of the attachment to the SRO directing emergency response activities. The symbol [1C] next to the step number for the action to establish letdown identifies that Attachment 1.C exists and the attachment may be delegated to the RO for performance. The step instruction presented on these attachments are redundant since the ERG procedure steps are maintained and available for monitoring by the SRO directing emergency response activities (e.g., should questions arise). When these attachments are developed, the unique attachment number (e.g., [1.D], [1.E], [1.F] should be used throughout the ERG network in order to minimize confusion between attachment numbers and activities being performed. When attachments for the same activity contain the same steps, an operator aid may be developed for attachment performance per Step I.13 of this procedure.

   The attachments with ERG actions that may be delegated (Attachment 1 series actions) are maintained in the procedure being implemented and in separate location for the RO and BOP positions. The SRO directing emergency response activities identifies when the ERG steps may be performed using the ERG attachment when direction is given for step performance. The operator given direction to perform the ERG step(s) may use the attachment provided in this location, or on an operator aid if one exists.

6. ERG entry is as follows:

   - IF at anytime a reactor trip or safety injection occurs or is required when the plant is initially in Mode 1, 2, or 3 (RCS >1,000 psi), THEN the operator enters EOP-0.0A/B, or

   - IF at anytime a complete loss of power to both AC safeguards busses takes place, THEN the operator enters ECA-0.0A/B. This includes anytime during performance of any other ERG.

Comments / Reference: From ODA-407, Attachment 8.A, page 4	Revision 15

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ERG RULES OF USAGE

7. EOS-0.0A/B, "Rediagnosis" provides reassurance to the operator that the correct ORG is being performed, or provides the necessary transition direction to get to the correct ORG for the existing symptoms.

   - EOS-0.0A/B is entered based on operator judgement, but shall not be entered prior to completing EOP-0.0A/B. The intent is that all automatic actuation verification steps be completed.

   - EOS-0.0A/B should only be used if SI is actuated or is required.

8. Some ABNs are required to be implemented during ERG performance. In some cases, the ABN will direct actions based on preventing equipment damage (e.g., ABN-101, "Reactor Coolant Pump Trip/Malfunction" requires trip of RCP(s) due to exceeding parameter limits). An evaluation by the SRO may be required prior to full compliance with the ABN to ensure the intent of the ERG is maintained.

   EXAMPLE - FRC-0.1A/B requires the start of RCPs to maintain core exit temperatures less than 1200°F. The direction for RCP start does not consider RCP support conditions or parameter limitations. If the RCPs are the only source of core cooling, the ABN RCP trip criteria would not apply.

   During ERG performance, actions in accordance with ABNs may be required to be performed to restore appropriate system or equipment function. ABN entry may be directed by an ERG step OR should be performed without ERG direction when the symptoms are satisfied to enter the ABN (e.g., The ERGs direct the operator to enter ABN-601, "Response To A 138/345 KV System Malfunction" when safeguards power is degraded. ABN-501, "Station Service Water System Malfunction" should be entered for response to a loss of all Station Service Water.) The actions to restore the system or equipment should be continued during emergency response activities to ensure the applicable function is available for ERG performance.

Examination Outline Cross-reference:

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.
- 1RE-5503, CNTMT AIR GAS (CAG-197) is GREEN 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, stable containment air gaseous activity below the alarm setpoint is not consistent with a primary coolant leak in containment.
D. Incorrect. Plausible because Containment Sump level and pressure are both rising and Steam Generator pressures are stable, however, stable containment air gaseous activity below the alarm setpoint is not consistent with a primary coolant leak in containment.

Technical Reference(s) WOG Background Document for EOP-2.0 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 ______

Comments / Reference: From WOG Background Document for EOP-2.0	Revision 2
For an intermediate feedline break in which the control systems are incapable of compensating for the loss of flow, the secondary side would experience a slowly decreasing steam generator water level in at least one steam generator. Depending upon the height of the low or low-low level trip setpoint in the steam generator and size of the break, a slowly increasing primary average temperature prior to reactor trip may occur due to the loss of main feedwater and degraded steam generator heat transfer. The transient is eventually terminated by manual reactor trip or when the low or low-low level trip setpoint is reached in any one steam generator. This results in a reactor trip and auxiliary feedwater initiation. A subsequent turbine trip occurs due to reactor trip. If the break occurs downstream of the main feedline non-return valves, all steam generators continue to experience a reverse blow down through the steam generator associated with the faulted loop until a low steamline pressure setpoint is attained resulting in a safety injection initiation and steamline and feedline isolation. The faulted steam generator will then blow down until atmospheric pressure is reached. If the break occurs upstream of the feedline non-return valves, the feedwater spillage is terminated and the auxiliary feedwater system is sufficient to mitigate the consequences of the resultant loss of normal feedwater transient. The system parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include decreasing water level in at least one steam generator and slowly rising primary system average temperature prior to reactor trip.	

Examination Outline Cross-reference:

	RO	SRO
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 actions is the preferred method to restore offsite power?

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. Aligning the 138 KV DeCordova line is the preferred Black Start corridor to restore power to Comanche Peak.

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 is 5% and slowly lowering.

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 per EOS-0.1A, Reactor Trip Response.

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.

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. EOS-0.1A Foldout Page requires manual initiation of SI when PRZR level cannot be maintained greater than 6% and a transition back to EOP-0.0A.
D. Incorrect. Plausible because Natural Circulation would be verified if SI was not required per the Foldout Page. Additionally, letdown is isolated if Pressurizer Level is less than 17%.

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 AA2.15	
Importance Rating	3.8	

Loss of Vital AC Instrument Bus: Ability to determine and interpret the following as they apply to a Loss of Vital AC Instrument Bus: That a loss of ac has occurred.

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 ____
55.43 5

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.
- Large non-essential loads have been shed.

When is additional load shedding required to be performed and what is the reason?

When Plant Staff has determined additional load shedding is required and DC Bus Voltage is…

A. …less than 110 VDC. This will ensure that all control room indications remain available.

B. …less than 110 VDC. This will maintain the ability to flash the Emergency Diesel Generator field.

C. …less than 115 VDC. This will ensure that all control room indications remain available.

D. …less than 115 VDC. This will maintain the ability to flash the Emergency Diesel Generator field.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because additional loads are shed at 110 VDC, however, all indication is NOT maintained (channels 3 and 4 are load shed).
B. Correct. Additional loads are shed at 110 VDC to maintain the ability to field flash the EDG.
C. Incorrect. Plausible if believed that additional load shedding is completed to maintain all channels of control room indication at 115 VDC. However, additional load shedding is conducted at 110 VDC to maintain the ability to field flash the EDG.
D. Incorrect. Plausible because additional load shedding is conducted to maintain the ability to field flash the EDG. However, additional load shedding is conducted at 110 VDC.

Technical Reference(s) ECA-0.0A, Attachment 7, Step 16 Bases
ECA-0.0A, Step 16
ECA-0.0A, Attachment 2 Bases

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. IF narrow range level in any SG continues to increase in an uncontrolled manner, THEN isolate ruptured SG: • Manually isolate AFW flow AND dispatch operator to locally isolate AFW flow. • IF SG 1 or 4 ruptured, THEN 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. WHEN SG pressure less than 1160 psig, THEN verify SG atmospheric closed. IF NOT, THEN manually close. IF SG atmospheric can NOT be closed, THEN 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.

Comments / Reference: From ECA-0.0A, Attachment 2, Bases	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. ECA-0.0A
LOSS OF ALL AC POWER	REVISION NO. 8	PAGE 86 OF 88

ATTACHMENT 7
PAGE 28 OF 30

BASES

ATTACHMENT 2

This attachment provides instructions for shedding of DC safeguards bus loads in order to conserve capacity to assist in future actions to restore AC power, while maintaining that minimum instrumentation necessary to monitor plant conditions. The intent of DC load shedding is to remove large loads not necessary for the Loss of All AC Power event as soon as practical. However, the specific battery sizing calculation for CPSES Station Blackout shows that, even without load shedding, the heaviest loaded battery with an assumed electrolyte temperature of 65EF, has sufficient capacity to not only carry its loads for a four (4) hour period, but also provide sufficient DC power for Diesel Generator field flashing.

The DC loads shed for this evolution have been reviewed, and the necessary instrumentation to provide monitoring during a Loss of All AC Power event are maintained. During the shedding activity, Protection Set Channel III and IV instruments are de-energized to conserve DC loads. The DC load shedding will cause a loss of some indication, and some signal actuation (e.g., Safety Injection actuation) by virtue removing power from the channels associated with protection set III and IV.

Examination Outline Cross-reference:

	RO	SRO
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 body to bonnet leak has developed on the Station Service Water (SSW) system Emergency Diesel Generator 1-02 Jacket Water Cooler inlet isolation valve.
- The crew is determining how to isolate the leak that will impact the FEWEST components.

Which of the following describes what should be isolated and which of the following components will be affected by isolating the Station Service Water system leak?

1. 1-02 Centrifugal Charging Pump lube oil cooler
2. X-02 Station Service Water Screen Wash Pump
3. 1-02 Component Cooling Water Heat Exchanger
4. 1-02 Containment Spray Pump bearing cooler

A. Train B Station Service Water should be isolated
   1 and 4

B. Train B Station Service Water 10” header should be isolated
   2 and 3

C. Train B Station Service Water should be isolated
   2 and 3

D. Train B Station Service Water 10” header should be isolated
   1 and 4

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because it could be thought that the entire train must be isolate, however only the 10” header should be isolated and the CCP and CTP are the correct components if the 10” header were isolated.
B. Incorrect. Plausible because the 10” header should be isolated however the CCW heat exchanger and screen wash pump are not supplied from the 10” header.
C. Incorrect. Plausible because it could be thought that the entire train must be isolate, however only the 10” header should be isolated and isolating the entire train would affect the CCW heat exchanger and screen was pump.
D. Correct. The CCP and CTP are loads on the 10 inch SSW header that would be affected by isolating the 10” header.

Technical Reference(s): SOP-501A, Step 5.5.2 CAUTION; ABN-501, Attachment 3; LO21.SYS.SW1 — 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

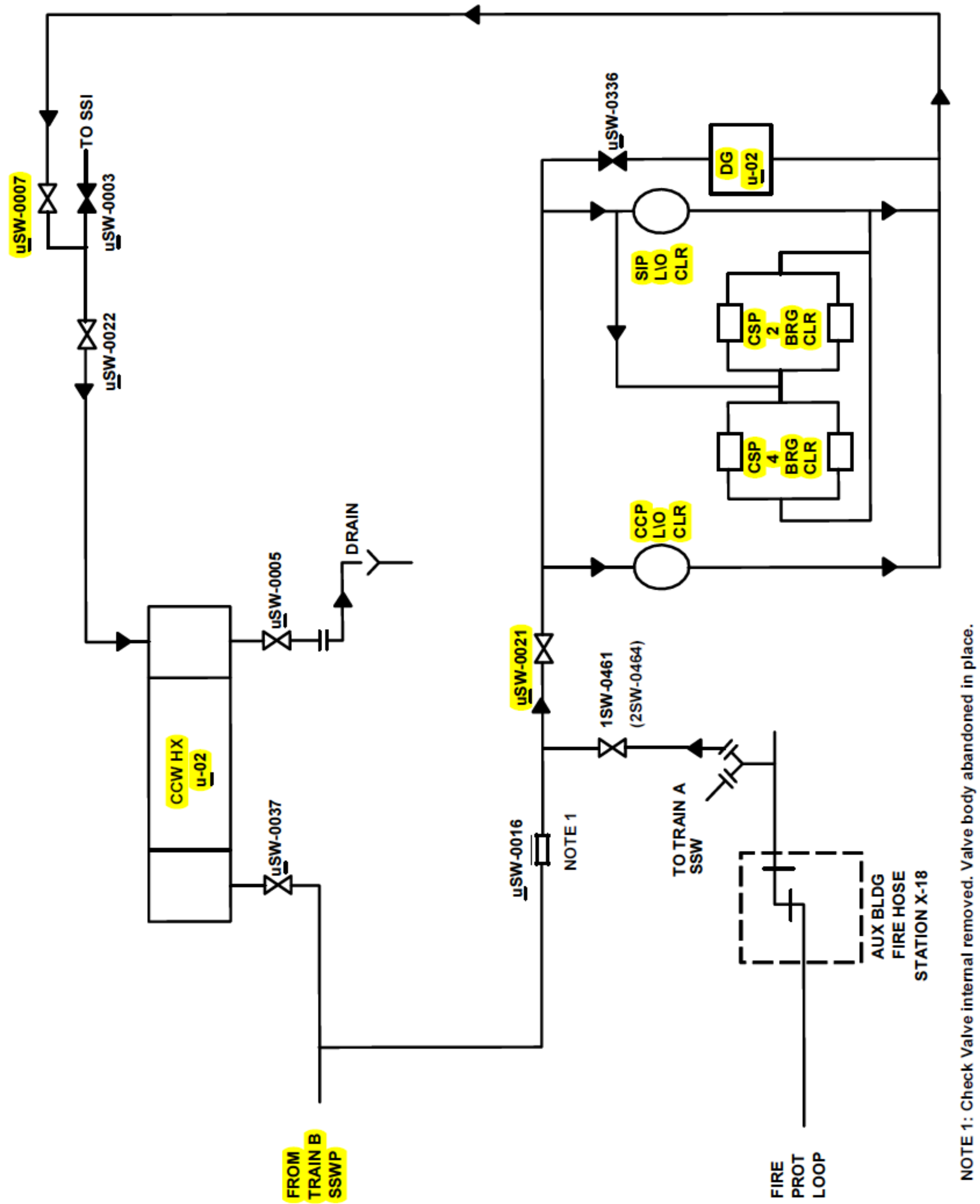
10 CFR Part 55 Content:
55.41 4, 10
55.43 ____

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.

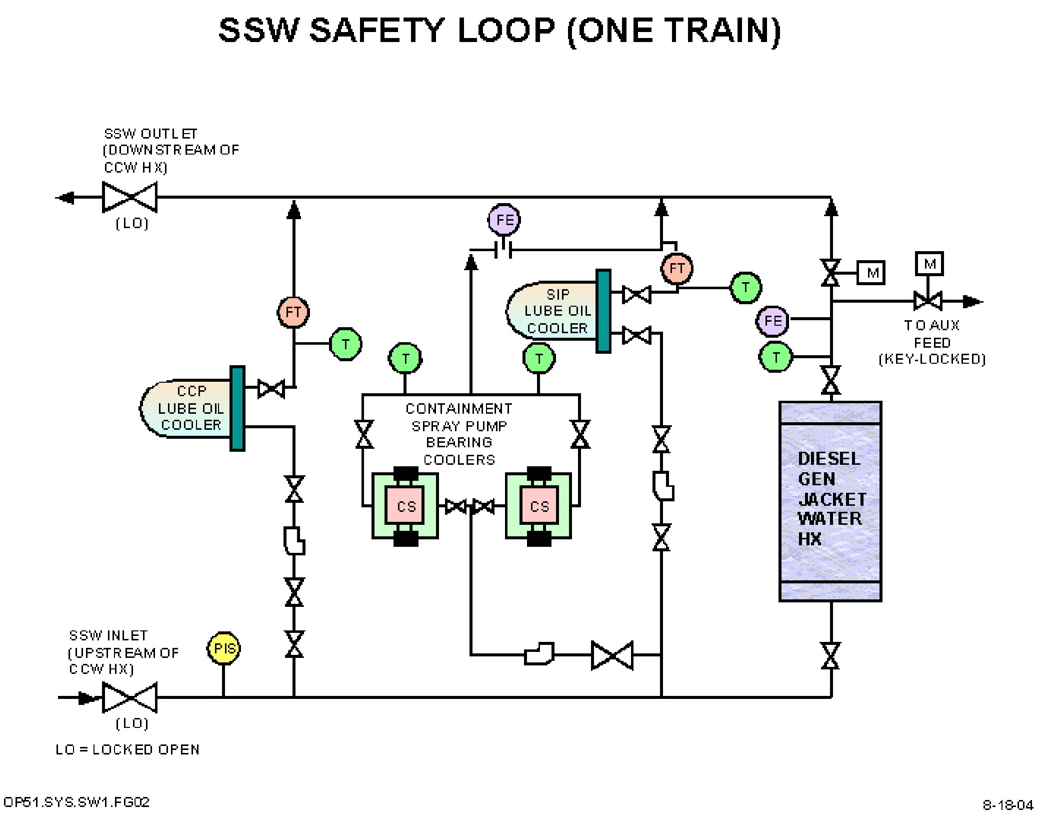


Figure 2 - SSW Safety Loop (One Train)

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**).

OP51.SYS.SW1.FIG1

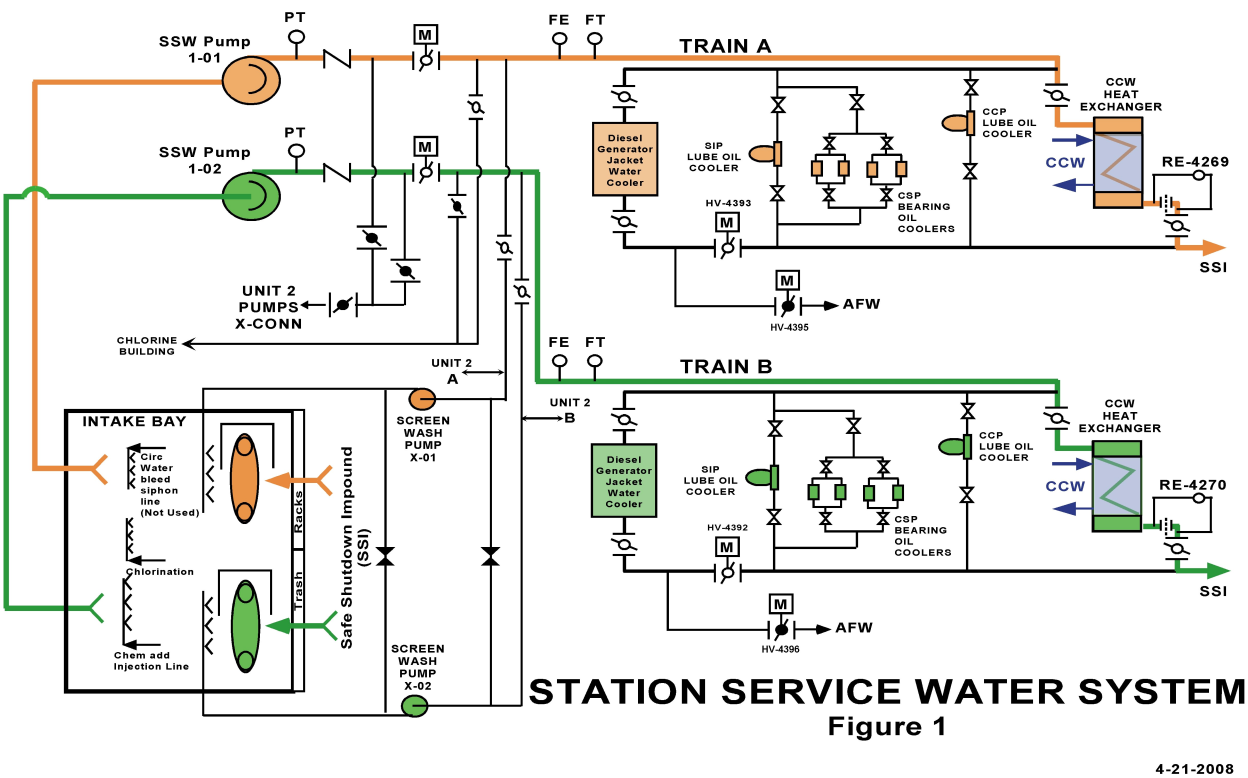


Figure 1 - Station Service Water System

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 is the reason the Turbine Driven Auxiliary Feedwater Pump steam supply valves are equipped with air accumulators?

Following a Loss of Instrument Air the air accumulators provide air to close and maintain closed its steam supply valve for...

A. ...30 minutes to isolate a faulted Steam Generator that has a steam or feed line break.

B. ...30 minutes following a Steam Generator Tube Rupture to mitigate radiological release.

C. ...7.5 hours to isolate a faulted Steam Generator that has a steam or feed line break.

D. ...7.5 hours following a Steam Generator Tube Rupture to mitigate radiological release.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Turbine Driven Auxiliary Feedwater Pump is isolated in EOP-2.0 response, however the 30 minutes accumulator use time is for the AFW flow control valves during a faulted SG isolation.
B. Incorrect. Plausible because the accumulators are provided for SGTR isolation, however they are sized for 7.5 hours.
C. Incorrect. Plausible because the accumulators are sized for 7.5 hours, however they are provided for SGTR isolation.
D. Correct. The accumulators are sized for 7.5 hours to isolate a SGTR to minimize radiological release.

Technical Reference(s) LO21.SYS.IA1 Attached w/ Revision # See Comments / Reference
LO21.SYS.MR1

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 #		
	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, 7	
	55.43		

Comments / Reference: From LO21.SYS.IA1.LN, Page 19	Revision 05/07/11

INDIVIDUAL VALVE ACCUMULATORS

Auxiliary Feedwater Control Valves & Recirculation Valves

Accumulators are provided for the Auxiliary Feedwater control valves in the individual feed lines, and for the motor driven auxiliary feedwater pumps recirculation valves. These valves share accumulators. These accumulators are sized to allow the operator remote, manual control to regulate flow or isolate a faulted steam generator for a period of 30 minutes after loss of air. The basis of the recirculation valves is to cycle one valve, then maintain the valve in the closed position for the remainder of the 30 minute period.

Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Turbine Main Steam Isolation Valves

Accumulators are provided for the turbine driven auxiliary feedwater pump (TDAFWP) turbine main steam isolation valves. These accumulators are sized to have sufficient air capacity to drive the valve closed and maintain it closed for 7 hours plus an additional one-half hour for the TDAFWP steam supply lines to then be locally-manually isolated by operator action, including any air leakage criteria.

Comments / Reference: From LO21.SYS.MR1.LN, Page 8	Revision 06/09/11

Turbine Driven Auxiliary Feedwater Pump Steam Supply

Redundant valves and piping provide a secure steam supply to the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) over a range of steam pressures from the lowest set main steam safety valve plus 3% accumulation (1185 psig +3% = 1221 psig or 1236 psia) to the lowest SG pressure (100 psia on the secondary side) associated with RHR cut-in conditions at 350˚F. The design ensures the TDAFWP can be operated even if one steam supply valve fails to open upon demand.

Normally closed isolation valves to the TDAFWP have the capability to open upon loss of all AC power. "Fail open" air operated valves are used for this application therefore, air accumulators tanks are provided to permit valve closure following the loss of air.

These valves are required to be closed following a SGTR to mitigate radiological consequences. Thus the air accumulators must contain sufficient air capacity to drive each valve closed and maintain it closed post accident until an operator can be dispatched to manually isolate the associated steam line.

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. Transition to ABN-601, Response to a 138/345 KV System Malfunction.

D. Transition to 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 ABN-601 is the RNO for EOP-0.0A step 3.b.

D. Incorrect. Plausible because ABN-602 is the RNO for EOP-0.0A step 3.b.

Technical Reference(s) EOP-0.0A, Step 3 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.

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
Comprehension or Analysis X

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: 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	Manually close valve(s). IF valve(s) can NOT be manually closed, THEN locally close valve(s).
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 — Attached w/ Revision # See Comments / Reference
FRH-0.1A, Steps 3, 20, & 21

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). IF no low pressure water source can be aligned, THEN 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 to the Refueling Water Storage Tank to extend ECCS pump availability.

B. ...decrease total injection flow to minimize Refueling Water Storage Tank depletion.

C. ...increase the injection flow rate to maintain Reactor Coolant System heat removal.

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. This action is based on ECA-1.1, major action category B, increase/conserve RWST level.
B. Correct. Reducing injection flow will minimize RWST depletion.
C. Incorrect. Plausible if thought that injection flowrate is increased in accordance with ECA-1.1 major action category G, maintain RCS heat removal.
D. Incorrect. Plausible based on misconception that it is desirable to prevent SI Accumulators from discharging in accordance with ECA-1.1 major action category D, depress RCS to minimize RCS break flow.

Technical Reference(s) ECA-1.1A, Steps 17 & 22 Attached w/ Revision # See Comments / Reference
ECA-1.1A, flowchart

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-

Comments / Reference: From ECA-1.1A, Flowchart	Revision 8

ECA-1.1A REV. 8 **LOSS OF EMERGENCY COOLANT RECIRCULATION**

A.	1. CHECK IF EMERGENCY COOLANT RECIRCULATION EQUIPMENT – AVAILABLE PER ATTACHMENT 2
B.	2. IF THE DIESELS ARE RUNNING, THEN PLACE BOTH DG EMER STOP/START HANDSWITCHES IN START
	3. RESET SI IF NECESSARY
	4. RESET SI SEQUENCERS IF NECESSARY
	5. RESET CONTAINMENT ISOLATION PHASE A AND PHASE B
	6. RESET CONTAINMENT SPRAY SIGNAL
	7. RESET RHR AUTO SWITCHOVER
	8. CHECK IF CONTAINMENT FAN COOLERS SHOULD BE STARTED

MAJOR ACTION CATEGORIES

A. CONTINUE ATTEMPTS TO RESTORE RECIRCULATION
B. INCREASE/CONSERVE RWST LEVEL
C. INITIATE COOLDOWN TO COLD SHUTDOWN
D. DEPRESS RCS TO MINIMIZE RCS BREAK FLOW
E. TRY TO ADD MAKEUP TO RCS FROM ALTERNATE SOURCE
F. DEPRESSURIZE SGs TO COOLDOWN & DEPRESSURIZE RCS
G. MAINTAIN RCS HEAT REMOVAL

Examination Outline Cross-reference:

	RO	SRO
Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	003 AK1.03	
Importance Rating	3.5	

Dropped Control Rod: Knowledge of the operational implications of the following concepts as they apply Dropped Control Rod: Relationship of reactivity and reactor power to rod movement.

Proposed Question: Common 57

Given the following conditions:

- Unit 1 is operating at Middle-of-Life 100% power.
- 1/1-RBSS CONTROL ROD BANK SELECT is in MANUAL.
- T_{AVE}-T_{REF} Deviation is 0ºF and stable.
- Control Bank D Control Rod H-8 drops to the bottom of the core.

After the Control Rod drops the following conditions exist:

- T_{AVE}-T_{REF} Deviation is -7.5ºF and stable.
- Reactor Power is 99% and stable.

What is the reactivity worth of Control Rod H-8?

A. 153 pcm

B. 138 pcm

C. 133 pcm

D. 123 pcm

Proposed Answer: A

Explanation:

A. Correct. In accordance with the provided MOL Reactivity Briefing Sheet a 7.5ºF lower temperature would be equivalent to -7.5 x -18.4 = 138 pcm/ºF plus a 1% power change which would be equivalent to 1 x 15.3 = 15.3 pcm/% RTP; therefore the total reactivity worth of the Control Rod would be 138 + 15.3 = 153 pcm.

B. Incorrect. Plausible as 138 pcm would be equivalent to the reactivity change from the temperature change only.

C. Incorrect. Plausible as 133 pcm would be calculated if the power and temperature coefficients were reversed in the calculation, -7.5 x 15.3 = 115 plus 1 x 18.4 = 18; 115 + 18 = 133 pcm.

D. Incorrect. Plausible as 123 pcm would be calculated if the reactivity change from power was subtracted from the reactivity change due to temperature; 138 – 15.3 = 123 pcm.

Technical Reference(s) Reactivity Briefing Sheet for MOL Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: Reactivity Briefing Sheet for MOL

Learning Objective: ANALYZE the response to a Dropped or Misaligned rod in MODE 1 or 2 in accordance with ABN-712, Rod 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 1
55.43 ___

Comments / Reference: From Reactivity Briefing Sheet for MOL	Revision 2/20/12

Reactivity Briefing Sheet for Stable Operation

MOL PROJECTIONS - SIMULATOR USE ONLY

Calculations based on core design values, and assume:

Burnup =	**12261.2**	MWD/MTU
	276.9	EFPD
Power =	**100**	RTP
Boron =	**907.97**	ppm
B10 Conc =	**.1834**	w/o
Control Bank D =	**215.0**	steps

Burnup in the MOL range

NOTE: Re-create the Briefing Sheet if current values significantly differ from assumed inputs.

Reactivity affects of Control Bank D

HFP Diff Worth @ 215.0 steps = **-1.4** pcm / step

HFP Integral Rod Worth for CBD Step Positions:

Steps	pcm	Steps	pcm	Steps	pcm	Steps	pcm
225	0.0	218	-4.7	211	-14.8	200	-41.4
224	0.0	217	-5.8	210	-16.8	195	-56.3
223	-1.2	216	-7.0	209	-18.8	190	-72.3
222	-1.6	215	-8.3	208	-21.0	185	-88.7
221	-2.2	214	-9.8	207	-23.3	180	-105.3
220	-2.9	213	-11.3	206	-25.7	175	-121.9
219	-3.7	212	-13.0	205	-28.1	170	-138.3

Reactivity affects of Boron

HFP Diff Boron Worth @ 908 ppm = **-7.5** pcm / ppm

1-FK-110 Pot Setting for Blended Flow @ 908 ppm = **3.87**

(Assuming BAT concentration of 7447.0 ppm)

Reactivity affects of Power

Power Coefficient of Reactivity = **-15.3** pcm / % RTP

Dilution to equal 1% Power Increase = **154.1** gallons RMUW

Boration to equal 1% Power Decrease = **20.6** gallons boric acid

Reactivity affects of RCS Temperature

Temperature Coefficient of Reactivity (ITC) = **-18.4** pcm / °F

Boration to equal 1°F Temperature Decrease = **24.7** gallons boric acid

Dilution to equal 1°F Temperature Increase = **185.4** gallons RMUW

Load Reduction equal to 1°F T_{ave} Increase = **14.0** MWe

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; 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.
- The Digital Radiation Monitoring System (PC-11) receives a high radiation alarm (RED) 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 complement 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	Granbury Fire Department
D.	Fire Brigade Leader	Granbury 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 Granbury 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 X
Comprehension or Analysis

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
using temperature correction of indicated level
using temperature correction of indicated level

B. plotting indicated temperature and pressure
using temperature correction of indicated level
using temperature correction of indicated level

C. reading RCS Saturation meter
reading Pressurizer level meter
reading Steam Generator level meter

D. plotting 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 X
Comprehension or Analysis ____

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

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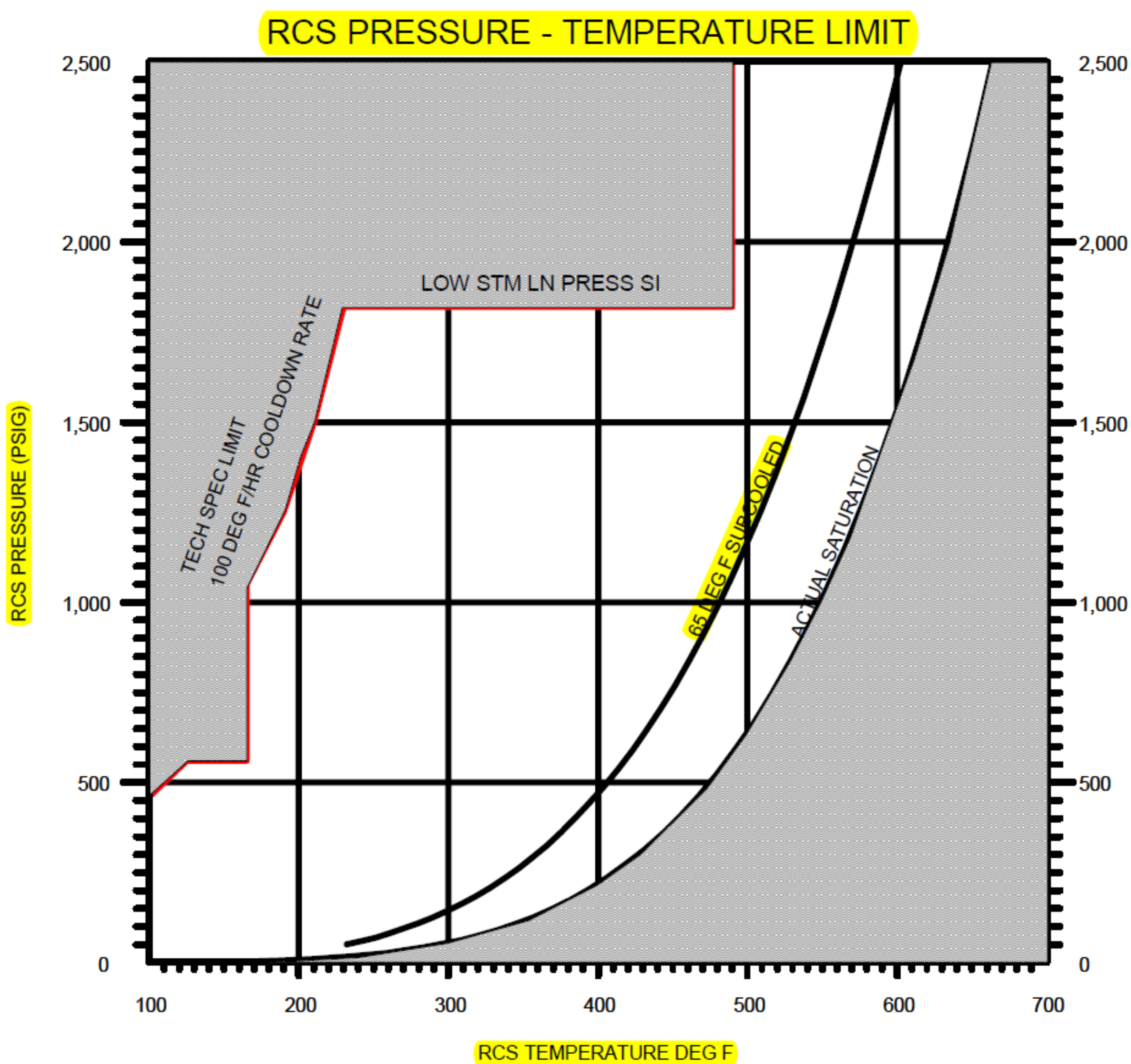
ATTACHMENT 13
PAGE 1 OF 1

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

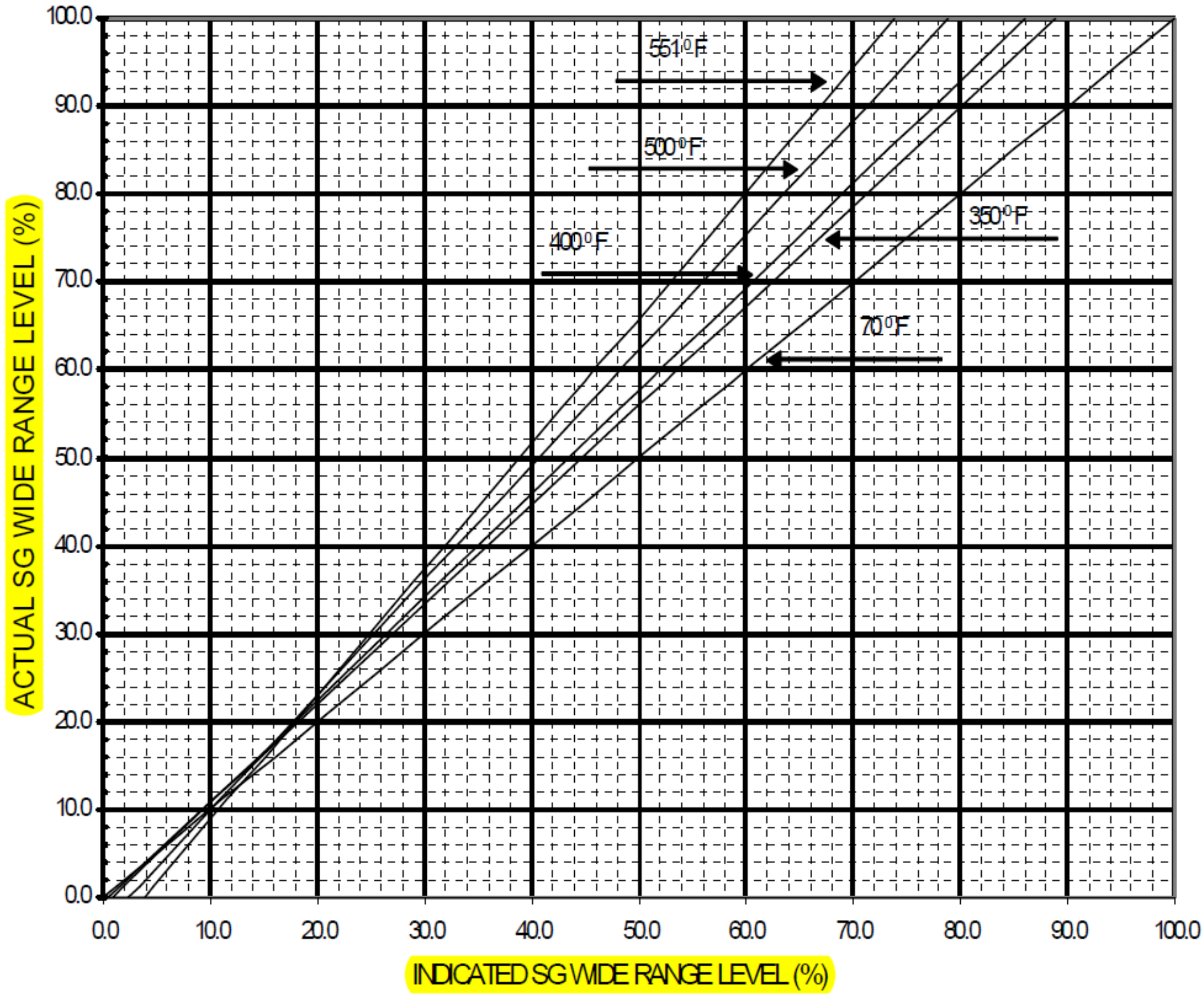
ATTACHMENT 16
PAGE 1 OF 1

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)



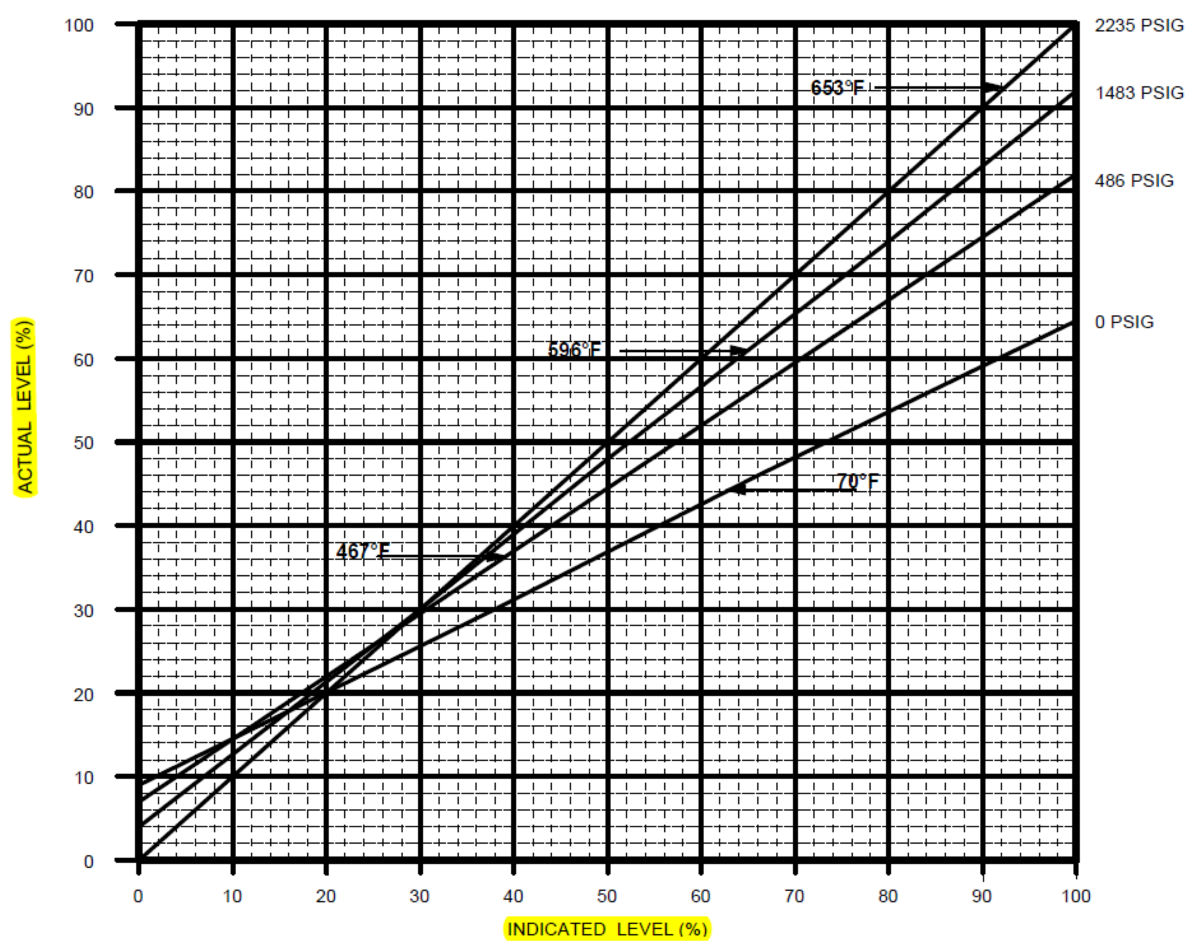
Comments / Reference: From ABN-905A, Attachment 17	Revision # 9

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:		RO	SRO
	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.
- Chemistry has sampled the Reactor Coolant System and determined that Co-58 and Co-60 levels are increasing.

Which of the following lists the expected indication and the most probable cause for the indication?

1-RE-406 (FFL160) indication…

A. …rising at a steady rate due to an oxygen induced CRUD burst.

B. …rising at a steady rate due to oxygen induced cladding creep.

C. …spiking and returning to normal due to an oxygen induced CRUD burst.

D. …spiking and returning to normal due to oxygen induced cladding creep.

Proposed Answer: A

Explanation:

A. Correct. An increase in Co-58 and Co-60 are the result of oxygen induced CRUD burst from shifting to the Refueling Water Storage Tank.
B. Incorrect. Plausible because a FFL160 indication would be rising but cobalt is not an indication of cladding damage.
C. Incorrect. Plausible because FFL160 indication would rise not spike and return to normal and cobalt comes from a CRUD burst not clad failure.
D. Incorrect. Plausible because FFL160 indication would rise not spike and return to normal and cobalt comes from a CRUD burst not clad failure.

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 reason for terminating Safety Injection flow during the performance of FRP-0.1A, Response to Imminent Pressurized Thermal Shock Conditions?

A. Prevent entry into a condition to the right of Limit A on the Integrity Status Tree.

B. Correct cause of event to allow exit from FRP-0.1A prior to completion.

C. Maintain RWST inventory in the event of flaw propagation.

D. Stop RCS cooldown and minimize RCS pressure increase.

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 correcting cause would be desired but once entered FRP-0.1A must be completed.
C. Incorrect. Plausible because terminating SI would conserve inventory but it is not the reason stated in the bases for FRP-0.1A, Step 7.
D. Correct. Following SI actuation, RCS conditions may be restored to within acceptable limits for SI termination to be allowed. 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 # ____ Modified Bank # X (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.

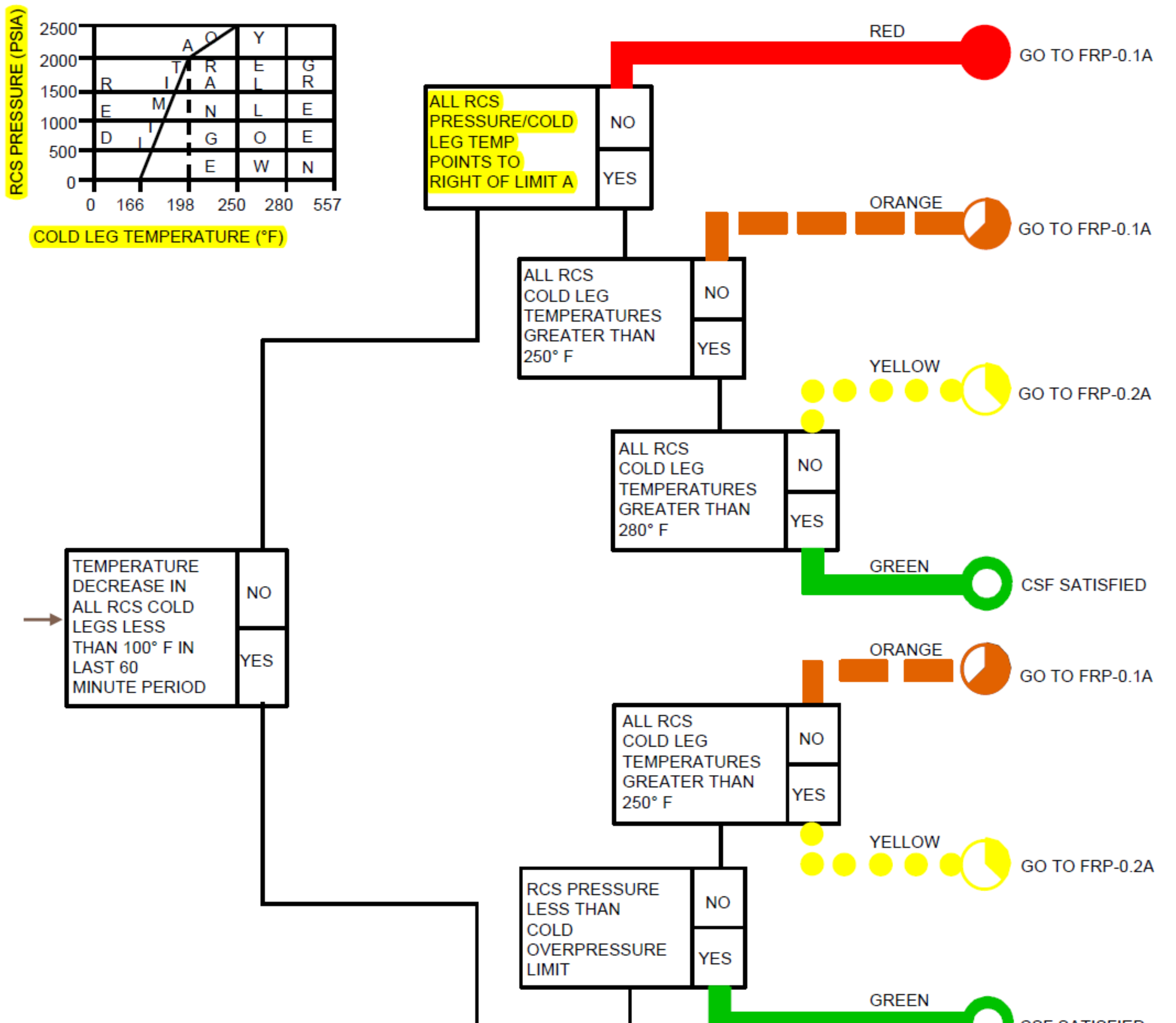
Based on the preceding caution, this action is applicable during recovery actions; therefore, the step has been designated as a continuous action step.

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

INTEGRITY



COLD OVER PRESSURE LIMIT

RCS TEMPERATURE	RCS PRESSURE
70	375
150	375
200	447
220	447
250	573
350	573

Original Bank Question

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

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 EOS-0.4B, Natural Circulation with Steam Void in Vessel (without RVLIS), when starting the first Reactor Coolant Pump, ensure pressurizer level is…

A. ...between 30% and 40% to compensate for the level increase due to the void formation.

B. ...between 30% and 40% to compensate for the level increase due to pump start.

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 eliminating a non-condensable void.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because IF a Reactor Coolant Pump CANNOT be started, EOS-0.4B contains guidance to maintain Pressurizer level between 30% and 40%. However, level will decrease due to void collapse when a Reactor Coolant Pump is started, and must be established at greater than 90%.
B. Incorrect. Plausible because IF a Reactor Coolant Pump CANNOT be started, EOS-0.4B contains guidance to maintain Pressurizer level between 30% and 40%. However, level will decrease due to void collapse when a Reactor Coolant Pump is started, and must be established at greater than 90%.
C. Correct. Pressurizer level is maintained above 90% to accommodate void collapse when starting the first Reactor Coolant Pump.
D. Incorrect. Plausible because the Pressurizer level value is correct, however, if non-condensable gases were indicated, they would be addressed via FRI-0.3B, Response to Voids in Reactor Vessel.

Technical Reference(s): EOS-0.4B, Step 1 & 2; FRI-0.3B, Attachment 7, 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-0.4B, Step 1 & 2	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. EOS-0.4B
NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITHOUT RVLIS)	REVISION NO. 8	PAGE 3 OF 42

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1	Restart An RCP:	
	a. Establish conditions for starting an RCP per Attachment 3.	a. Go to Step 2. OBSERVE NOTE PRIOR TO STEP 2.
	b. Check PRZR Level - GREATER THAN 90%	b. Control charging and letdown as necessary to increase level to greater than 90%.
2	Establish PRZR Level To Accommodate Void Growth:	
	a. Check PRZR level - BETWEEN 30% AND 40%	a. Control charging and letdown as necessary.
	b. Place PRZR level controls in manual.	

Comments / Reference: From FRI-0.3B, Attachment 7, Bases	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 2	PROCEDURE NO. FRI-0.3B
RESPONSE TO VOIDS IN REACTOR VESSEL	REVISION NO. 8	PAGE 31 OF 44

ATTACHMENT 7
PAGE 5 OF 18

BASES

NOTE: The approach in Step 9 assumes that the voids consist almost entirely of steam with very little non-condensible constituents. If, for any reason (e.g., core uncovery, accumulator injection), the operator suspects a large fraction of the voids to be non-condensibles, Step 9 may be bypassed in favor of direct vessel venting. This course of action would retain the non-condensibles in the upper head and venting is the most effective means of eliminating them from the RCS. If, however, the voids are suspected to be primarily steam, starting an RCP provides the most effective actions to deal with the void.

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 Loss of Coolant Accident.
- Containment pressure is 8 psig and rising.
- While responding in EOP-1.0A, Loss of Reactor or Secondary Coolant, the Control Room observed Containment Sump level greater than 816' and entered FRZ-0.2A, Response to Containment Flooding.
- 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.

Which of the following systems may still be contributing to the Containment Flooding?

A. Main Feedwater

B. Fire Protection Water

C. Ventilation Chilled Water

D. Component Cooling Water

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because Main Feedwater could be a contributor if not isolated on reactor trip with low Tave.
B. Incorrect. Plausible because Fire Protection Water could be a contributor if not isolated by Phase A containment isolation.
C. Incorrect. Plausible because Ventilation Chilled Water could be a contributor if not isolated by Phase A containment isolation.
D. Correct. Containment pressure has not reached 18 psig yet so Component Cooling water is not completely isolated to containment and could be the source.

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 ____
Comprehension or Analysis X

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 #	1	
	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 is enrolled in the Initial Licensed Operator Training Program.
- 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.

Who can approve the trainee performing the dilution and what requirement must be met by the trainee?

A. Shift Manager
The trainee has completed Initial Non-Licensed Operator training classroom instruction for BTRS.

B. Shift Operations Manager
The trainee has completed Initial Licensed Operator training classroom instruction for BTRS.

C. Shift Operations Manager
The trainee has completed Initial Non-Licensed Operator training classroom instruction for BTRS.

D. Shift Manager
The trainee has completed Initial Licensed Operator training classroom instruction for BTRS.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible because the Shift Manager must approve the use of a trainee for a reactivity manipulation, however, a trainee must have completed the classroom phase of license training covering the evolution in question prior to manipulating controls.
B. Incorrect. Plausible because one requirement for a trainee to manipulate controls is to be enrolled in a replacement licensed operator training program. However, the Shift Operations Manager is required to approve the use of a trainee for a reactor startup only and a trainee must have completed the classroom phase of license training covering the evolution in question prior to manipulating controls.
C. Incorrect. Plausible because one requirement for a trainee to manipulate controls is to be enrolled in a replacement licensed operator training program. However, the Shift Operations Manager is required to approve the use of a trainee for a reactor startup only and a trainee must have completed the classroom phase of license training covering the evolution in question prior to manipulating controls.
D. Correct. The Shift Manager must approve the use of a trainee for a reactivity manipulation. Additionally, 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 #	1	
	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 ≥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 #	2	
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 SHALL NOT be opened until Reactor Vessel level is raised above 80".

B. The penetration can be opened for maintenance with NO administrative controls.

C. The penetration can be opened for maintenance if tracked per Technical Specifications.

D. The penetration SHALL 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 #	2	
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. 139ºF

B. 199ºF

C. 319ºF

D. 349ºF

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 350ºF 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; Technical Specification Definitions 1.1

Attached w/ Revision # See Comments / Reference

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	$\geq$ 0.99	> 5	NA
2	Startup	$\geq$ 0.99	$\leq$ 5	NA
3	Hot Standby	< 0.99	NA	$\geq$ 350
4	Hot Shutdown(b)	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown(b)	< 0.99	NA	$\leq$ 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 #	3	
	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 amount of additional 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 amount of additional 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; 5000 mrem

D. 3000 mrem; 5000 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, 5000 mrem is the legal limit. Therefore, the person may only receive 4000 mrem additional.
D. Incorrect. Plausible because the Admin limit previously was 4000 mrem and a Planned Special Exposure is also 4000 mrem, however, 1000 mrem is the admin limit. Additionally, 5000 mrem is the legal limit and the person may only receive 2000 mrem additional.

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 # 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 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)	4000 mrem
	NOT TO EXCEED:	
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

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	3	
K/A #	G 2.3.12	
Importance Rating	3.2	

Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

Proposed Question: Common 71

Given the following:

- Radiography is in progress in the Unit 2 Safeguards Building.
- When the source is exposed the dose rates in the area are rising to 1440 mrem/hr.
- The radiography is being performed in an open area of the Safeguards Building with NO enclosure.

Which of the following describes the type of radiological area and type of radiation monitoring required for entry?

A. Very High Radiation Area (VHRA)
Electronic Dosimeter and TLD

B. Locked High Radiation Area (LHRA)
Electronic Dosimeter and TLD

C. Very High Radiation Area (VHRA)
TLD only

D. Locked High Radiation Area (LHRA)
TLD only

Proposed Answer: B

Explanation:

A. Incorrect. Plausible if thought that greater than 1000mr/hr is VHRA.
B. Correct. Area meets requirements for LHRA and Electronic dosimeter may be used for entry dose monitoring along with the TLD.
C. Incorrect. Plausible if thought that greater than 1000mr/hr is VHRA but area is LHRA due to dose rate and TLD only is not sufficient for monitoring.
D. Incorrect. Plausible LHRA is correct but TLD only is not sufficient for monitoring.

Technical Reference(s) STA-660 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 # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

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

10 CFR Part 55 Content: 55.41 12
55.43 ____

Comments / Reference: From STA-660	Revision 15 PCN 1

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15 INFORMATION USE	Page 3 of 11

4.2 Dose Margin - The remaining allowable total effective dose equivalent an individual may receive during a specified monitoring period.

4.3 High Radiation Area (HRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 100 mrem in 1 hour at 30 centimeters (approximately 1 foot) from the radiation source or from any surface that the radiation penetrates.

4.4 Locked High Radiation Area (LHRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 1000 milli-rem in 1 hour at 30 centimeters (approximately 1 foot) from the radiation source or from any surface that the radiation penetrates.

4.5 RAD Key – A mechanical key that provides access to Locked High Radiation Areas by the ability to open an associated RAD Lock.

4.6 RAD Lock – A lock used exclusively for controlling access to Locked High Radiation Areas.

4.7 Radiologically Significant ALARA Briefing – A documented briefing between Radiation Protection and participants prior to the commencement of work activities where radiological conditions are subject to frequent or rapid change. This briefing shall be performed prior to entry into a posted LHRA or VHRA.
[TS 5.7]

4.8 Electronic Dosimeter – A radiation monitoring device which continuously integrates the radiation dose rate and alarms when a preset integrated dose or dose rate is received.

4.9 Expected Dose – The dose that is expected for the duration of an entry into an area. The expected dose may be a dose calculated during job planning for all persons entering the area (e.g., Steam Generator channel head entries may have a dose setting of 750 mrem for the planned activities).

4.10 Stay Time – The length of time an individual may be allowed into an area based on radiation levels and remaining dose margin or expected dose, whichever is the most limiting.

4.11 Very High Radiation Area (VHRA) – An area, accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in 1 hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

Comments / Reference: From STA-660	Revision 15 PCN 1

CPNPP STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-660
CONTROL OF HIGH RADIATION AREAS	REVISION NO. 15	Page 5 of 11
	INFORMATION USE	

6.1.2.1 For injury responses outside of a declared emergency, this requirement is met as long as Radiation Protection provides an escort into the affected area.
[CR-2009-001502]

6.1.3 Prior to entering a High Radiation Area, personnel shall receive an Area ALARA or Radiologically Significant ALARA briefing.
[TS 5.7]

6.1.3.1 For injury responses outside of a declared emergency, this requirement is met as long as Radiation Protection provides an escort into the affected area and emergency response personnel are advised of the dose rates in the area prior to entry.
[CR-2009-001502]

6.1.4 All entries shall be provided with or accompanied by one or more of the following:

- A radiation monitoring device which continuously indicates the radiation dose rate in the area, OR
- Electronic Dosimeter – Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them, OR
- An individual qualified in radiation protection procedures equipped with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and should perform periodic radiation surveillance at the frequency specified by the RWP.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	3	
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 conditions:

- 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 be fully withdrawn from the core.

B. …shall be fully inserted into the core.

C. …shall be stored and tagged out-of-service.

D. …Drive System must be disconnected.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible if thought that withdrawing incore detectors from the core places detectors in storage. However, when incore detectors are fully withdrawn, they are located in the incore instrumentation room. Additionally, the system must be tagged out to prevent inadvertent movement.

B. Incorrect. Plausible because the incore detectors may be fully inserted in the core, however, the incore detectors may also be inserted in their storage locations and the system is required to be 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 would prevent 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	
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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 #	4	
K/A #	G 2.4.49	
Importance Rating	4.6	

Emergency Procedures / Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question: Common 73

Which of the following applies to the performance of Abnormal Conditions Procedure (ABN) Initial Operator Actions?

ABN Initial Operator Actions SHALL be performed…

A. …without verbalization and without a brief pause for SRO intervention.

B. …with verbalization and without a brief pause for SRO intervention.

C. …with verbalization and with a brief pause for SRO intervention.

D. …without verbalization and with a brief pause for SRO intervention.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because this is the expectation for ERG Immediate Action performance.
B. Incorrect. Plausible because actions shall be verbalized, however, the brief pause is required.
C. Correct. This is the expectation for ABN Initial Operator Action performance per Operations Guideline 3.
D. Incorrect. Plausible because verbalization is NOT required for ERG Immediate Action performance; however, the brief pause is required for ABN Initial Operator Action performance.

Technical Reference(s): Operations Guideline 3, Attachment 6; A ODA-102, section 6.9 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:

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 2

Comments / Reference: From Operations Guideline 3, Attachment 6	Revision 110

Operations Guideline 3
Attachment 6

- Reactor Operator communication during initial and immediate actions:
  - ABN Initial operator actions and actions taken in response to instrument failures per ODA-102 should be verbalized by the RO prior to the action being taken, allowing a brief pause for the SRO to intervene. Two or three way communication is neither expected nor desired.
  - ERG Immediate actions should be performed without communication. The actions will be communicated as directed by the SRO when the procedure step is read.

Page **9** of **16**

Comments / Reference: From ODA-102, section 6.9	Revision 26

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-102
CONDUCT OF OPERATIONS	REVISION NO. 26 INFORMATION USE	PAGE 18 OF 37

6.9 Duties/Responsibilities of the Unit Reactor Operator (RO)

- Operate equipment in accordance with approved procedures, Technical Specifications, facility operating license and other regulatory requirements.
- Assist the SM or US in the review and implementation of clearances and in returning equipment to service when authorized.
- Assist in the training of NEOs and license candidates.

[C] • Normally operate the controls on Control Boards CB-4 through CB-8 with overlapping responsibility for CB-9 with the BOP Operator. [06225]

[C] • Perform all immediate/initial actions specified in Emergency Response Guideline (ERG) and Abnormal Operating (ABN) procedures without prior US permission. Verbalize all initial actions of the ABNs to keep the unit personnel informed. [08801, 09447]

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	4	
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 responding to a Steam Generator Tube Rupture and is currently in EOP-0.0A, Reactor Trip or Safety Injection.

Which of the following is correct concerning Transient Annunciator Response?

A. The master silence button may be used repeatedly at the discretion of the Reactor Operator after verbalizing Silencing.

B. The Reactor Operator shall inform the Unit Supervisor of ALL Orange or Yellow annunciators in alarm.

C. The Reactor Operator shall inform the Unit Supervisor of any alarms that indicate failure of an ESF component.

D. Annunciators which cannot be expeditiously addressed should continue to Flash until they can be addressed.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible as the master silence is to be used to allow clear communications, however, repeated use of the master silence requires Unit Supervisor approval.
B. Incorrect. Plausible as all Orange and Yellow annunciators show a level of importance beyond the standard white annunciator, however, the operating guidance is that only those annunciators which indicate that another ERG or ABN should be entered or that indicate ESF component failures or ESF actuations that may have occurred or need to be manually performed should be verbalized reported to the Unit Supervisor.
C. Correct. The failure of an ESF component should be reported to the Unit Supervisor within normal communication standards.
D. Incorrect. Plausible as not all annunciators can be expeditiously addressed, however, the guidance is that the annunciators should be acknowledged and cleared when conditions stabilize in order to identify those that may indicate a subsequent fault or transient.

Technical Reference(s) A Operations Guideline 3, Attachment 6 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 # ___
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 Operations Guideline 3, Attachment 6	Revision 110

Operations Guideline 3
Attachment 6

5.0 <u>Transient Annunciator Response</u>

- This guidance applies after entry into an ABN or ERG entry, or as directed by the Unit Supervisor.

- During transient conditions, it is desirable to silence the alarms as soon as practical to allow clear communications. It is not required to verbalize **"Silencing."** The master silence button may be used when entering EOP 0.0; other uses require the Unit Supervisor permission. When the transient conditions have stabilized the audible alarms should be reinstated and alarms acknowledged. Unit Supervisor permission is required to use the master silence button again if the master silence alarm times out (four minutes).

- Following completion of ABN Initial Operator Actions /ERG Immediate Actions, annunciators should be acknowledged and cleared when conditions stabilize. This will allow the crew to identify those that may be the result of a subsequent fault or transient.

  - During ERG implementation or transients where multiple annunciators are sporadically received, the RO/BOP shall assess all annunciators and inform the Unit Supervisor of any annunciators that:
    - Requires entry into additional ERGs
    - Requires entry into additional ABNs
    - Indicate failure of an ESF component
    - Indicate ESF actuations have occurred or may be needed.

    The Unit Supervisor should repeat back the annunciator.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	3	
Category #	4	
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 alarms is NOT CONSISTENT with given plant conditions?

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 if believed that the automatic runback coupled with feedwater control valve response will maintain steam generator water level within 5% of program. However, the large feedwater flow transient caused by the trip of a Main Feedwater Pump will cause steam generator level deviations of sufficient magnitude to cause level deviation alarms.

B. Incorrect. Plausible if believed that automatic rod control will maintain T_{AVE} within 2.5°F of T_{REF}. However, the rate of turbine power decrease due to the automatic runback (35%/minute) will cause T_{AVE}-T_{REF} deviation to go above the alarm setpoint.

C. Incorrect. Plausible if believed that the automatic runback following a Main Feed Pump trip occurs at the “normal” runback rate of 10 MW/minute. However, the automatic runback occurs at 35%/minute, which is greater than the 10% power per 120 second arming setpoint for C-7.

D. Correct. When a Main Feedwater Pump trips, a turbine runback is automatically initiated. This will cause control rods to step in due to a mismatch in turbine and reactor power. Control Bank D will not move out to 223 steps and 1-ALB-6D, Window 4.14 should not alarm.

Technical Reference(s) ABN-302, Sections 2.1 and 2.2
ALM-0064A, 1-ALB-6D, Window 4.14
ALM-0064A, 1-ALB-06D, Window 1.10
ALM-0065A, PCIP, Window 3.4

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 ABN-302, Sections 2.1 and 2.2	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 3 OF 78

2.0 FEEDWATER PUMP TRIP

2.1 Symptoms

a. Annunciator Alarms

- FWPT A TRIP (7B-1.12)
- FWPT B TRIP (8A-1.3)
- 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)
- ANY TURB RUNBACK EFFECTIVE (6D-1.9)
- Various Digital Alarms (ASD)

b. Plant Indications

4) Steam dump valve actuation in response to loss of load (C-7) signal and Tave-Tref mismatch (greater than 5°F).

5) Control rods stepping in (if in AUTO) in response to Tave-Tref mismatch

- CONTROL ROD MOTION 1/u-RIL

2.2 Automatic Actions

- Turbine runback at 35% per minute to 60% power.

Section 2.0

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:

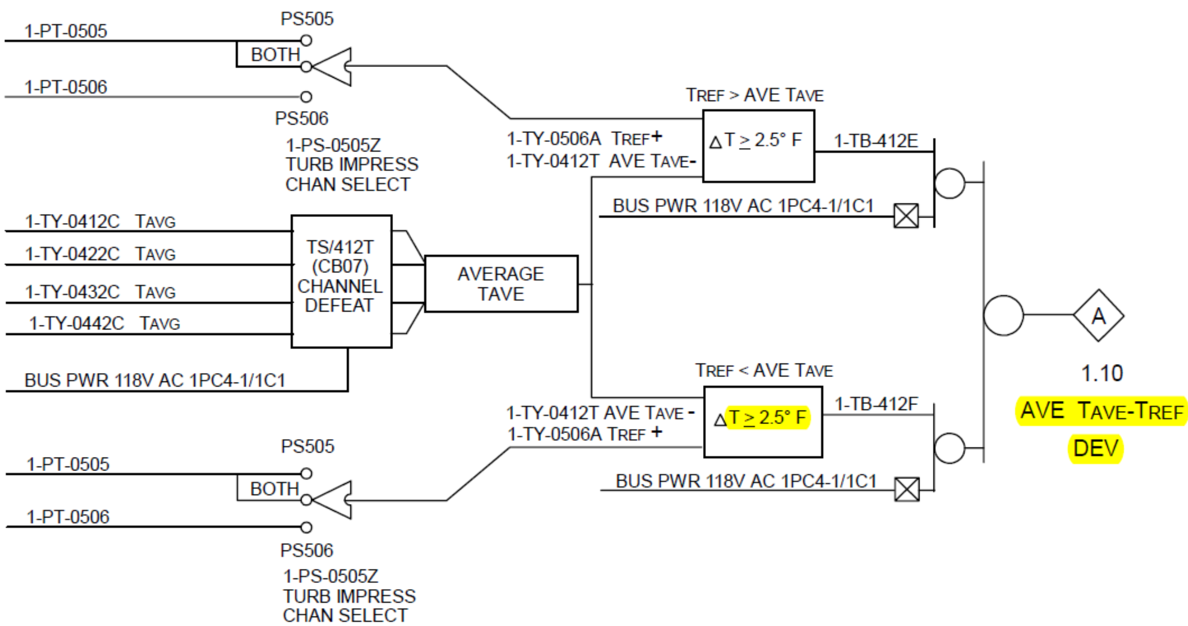
1. Stop all secondary power changes.
2. Verify 1-SC-CBD1, CTRL BANK D GROUP 1 and 1-SC-CBD2, CTRL BANK D GROUP 2 step counters indicates $\leq$231 steps.
   A. If bank D rods are $>$231 steps, refer to ABN-712.

Comments / Reference: From ALM-0064A, 1-ALB-06D, Window 1.10	Revision 6

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0064A
ALARM PROCEDURE 1-ALB-6D	REVISION NO. 6	PAGE 34 OF 147

ANNUNCIATOR NO.: 1.10

LOGIC:

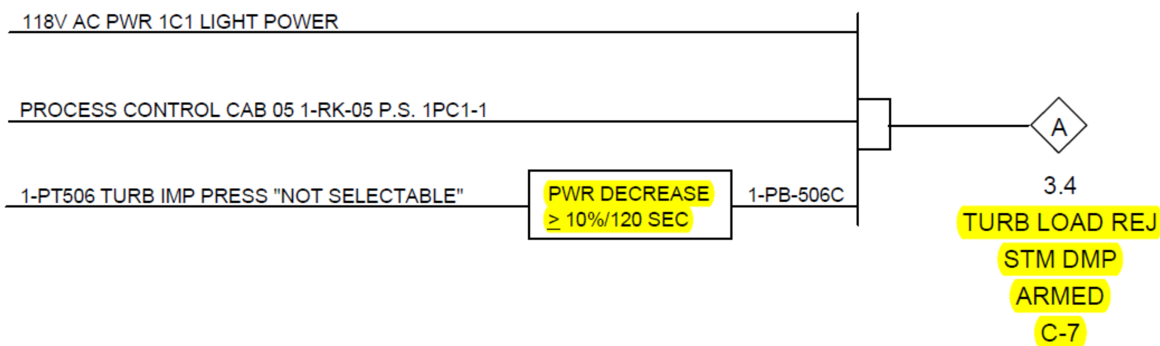


Comments / Reference: From ALM-0065A, PCIP, Window 3.4	Revision 4

CPSES ALARM PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. ALM-0065A
ALARM PROCEDURE 1-PCIP	REVISION NO. 4	PAGE 46 OF 73

ANNUNCIATOR NO.: **3.4**

LOGIC:



Examination Outline Cross-reference:

	RO	SRO
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?

Establish a level band in...

A. ...Steam Generator 2-02 of 10% to 50%.

B. ...Steam Generators 2-02 & 2-03 of 18% to 50%.

C. ...Steam Generators 2-02, 2-03 & 2-04 of 43% to 60%.

D. ...Steam Generators 2-01, 2-02, 2-03 & 2-04 of 50% to 60%.

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. <u>IF</u> narrow range level in any SG continues to increase in an uncontrolled manner, <u>THEN</u> 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. <u>IF</u> narrow range level in any SG continues to increase in an uncontrolled manner, <u>THEN</u> 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.
- The following alarms are received;
    - 1-ALB-6A, Window 1.4, REGEN HX LTDN OUT TEMP HI
    - 1-ALB-6A, Window 3.4, CHRG FLO HI/LO

What action must be taken and the procedure used?

A. Stop Centrifugal Charging Pump 1-01 and then isolate letdown per SOP-103A, Chemical and Volume Control System.

B. Stop Centrifugal Charging Pump 1-01 and then isolate letdown per ABN-105, Chemical and Volume Control System Malfunction.

C. Isolate letdown and then stop Centrifugal Charging Pump 1-01 per SOP-103A, Chemical and Volume Control System.

D. Isolate letdown and then stop Centrifugal Charging Pump 1-01 per ABN-105, Chemical and Volume Control System Malfunctions.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible because the actions stated are correct, however, ABN-105 is the correct procedure to use.
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 actions stated are correct, however, they must be performed in the opposite order in accordance with ABN-105.
D. Incorrect. Plausible because actions stated are correct, however, they must be performed in the opposite order.

Technical Reference(s) ABN-105, Step 7.3.1 RNO — 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.	
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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	029 G 2.4.6	
	Importance Rating		4.7

ATWS: Knowledge of EOP mitigation strategies.

Proposed Question: SRO 78

Given the following conditions:

- An ATWT has occurred on Unit 1.
- FRS-0.1A, Response to Nuclear Power Generation/ATWT is in progress.
- Emergency Boration is in progress.
- Safety Injection has actuated.
- All Steam Generator pressures are ≈ 800 psig and lowering.
- All Reactor Coolant System Cold Leg temperatures are ≈ 521°F and lowering.
- Reactor Power is ≈ 9% and lowering.

Which of the following mitigation strategies is appropriate for the event?

A. Remain in FRS-0.1A, Response to Nuclear Power Generation/ATWT and isolate faulted steam generators. Transition to EOP-0.0A, Reactor Trip or Safety Injection when FRS-0.1A, Response to Nuclear Power Generation/ATWT is complete.

B. Remain in FRS-0.1A, Response to Nuclear Power Generation/ATWT and isolate faulted steam generators. Transition to EOP-0.0A, Reactor Trip or Safety Injection when faulted SG isolation is complete.

C. Transition to EOP-2.0A, Faulted Steam Generator Isolation while performing FRS-0.1A, Response to Nuclear Power Generation/ATWT in parallel. Transition to EOP-0.0A, Reactor Trip or Safety Injection when faulted SG isolation is complete.

D. Transition to EOP-2.0A, Faulted Steam Generator Isolation while performing FRS-0.1A, Response to Nuclear Power Generation/ATWT in parallel. Transition to EOP-0.0A, Reactor Trip or Safety Injection when FRS-0.1A, Response to Nuclear Power Generation/ATWT is complete.

Proposed Answer: A

Explanation:

A. Correct. The faulted SG will be isolated in FRS-0.1A. Transition to EOP-0.0A will be made when FRS-0.1A is complete.
B. Incorrect. Plausible because remaining in FRS-0.1A is correct but transition to EOP-0.0A after fault isolation is not correct. FRS-0.1A must be completed to transition to EOP-0.0A.
C. Incorrect. Plausible because EOP-2.0A can be performed in parallel with other ORGs, however FRS-0.1A must remain as the procedure in affect. FRS-0.1A must be completed to transition to EOP-0.0A.
D. Incorrect. Plausible because EOP-2.0A can be performed in parallel with other ORGs, however FRS-0.1A must remain as the procedure in affect. Transition to EOP-0.0A will be made when FRS-0.1A is complete.

Technical Reference(s): ODA-407, Attachment 8.A; FRS-0.1A, step 15; FRS-0.1A, Step 6; EOP-2.0A, Symptoms and Entry Conditions — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Given a procedure step, note, or caution discuss the reason for the step, note, or caution in FRS-0.1

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
55.43 5

Comments / Reference: From ODA-407, Attachment 8.A	Revision 8

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-407
OPERATIONS DEPARTMENT PROCEDURE USE AND ADHERENCE	REVISION NO. 15 INFORMATION USE	PAGE 31 OF 56

ATTACHMENT 8.A
PAGE 13 OF 22

ERG RULES OF USAGE

15. • FRZ-0.1A/B or EOP-2.0A/B may be performed in parallel with other ORGs provided the "procedure in effect" is maintained as highest priority and is not sacrificed to perform the "parallel procedure". The "procedure in effect" is defined as that procedure in which the ERG network directs present operator actions.

Comments / Reference: From FRS-0.1A, step 15	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. FRS-0.1A
RESPONSE TO NUCLEAR POWER GENERATION/ATWT	REVISION NO. 8	PAGE 9 OF 30

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15	Isolate Faulted SG(s): • Isolate main feedline. • Isolate AFW flow. • <u>IF</u> SG 1 or 4 faulted, <u>THEN</u> place TDAFW Pump steam supply valve(s) in pull-out. • Isolate blowdown and sample lines. • Ensure SG atmospheric(s) - CLOSED • Ensure main steamline drippot isolation valve(s) - CLOSED	<u>IF</u> SG atmospheric(s) can <u>NOT</u> be closed, <u>THEN</u> dispatch operator to locally close block valve(s). <u>IF</u> other valves can <u>NOT</u> be closed, <u>THEN</u> dispatch operator to locally close valve(s) or block valve(s).

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 EOP-2.0A, Symptoms and Entry Conditions	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-2.0A
FAULTED STEAM GENERATOR ISOLATION	REVISION NO. 8	PAGE 2 OF 14

A. PURPOSE

This procedure provides actions to identify and isolate a faulted steam generator.

B. APPLICABILITY

This procedure is applicable for initiating events occurring in MODES 1, 2, 3 and 4. This procedure assumes RCS temperature is greater than 212°F. 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 with the following symptoms:

   a. Any SG pressure decreasing in an uncontrolled manner, or
   b. Any SG completely depressurized

2) EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT,
   EOP-3.0A, STEAM GENERATOR TUBE RUPTURE,
   ECA-3.1A, SGTR WITH LOSS OF REACTOR COOLANT - SUBCOOLED RECOVERY DESIRED, or,
   ECA-3.2A, SGTR WITH LOSS OF REACTOR COOLANT - SATURATED RECOVERY DESIRED when faulted SG isolation is not verified.

3) FRH-0.5A, RESPONSE TO STEAM GENERATOR LOW LEVEL when the affected SG is diagnosed as faulted.

4) Foldout page of other procedures, whenever a faulted SG is identified.

5) ECA-2.1A Foldout Page if any SG pressure increases.

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).
- 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 lower seal bearing temperature reaches 225°F 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 | Attached w/ Revision # See Comments / Reference
ABN-101, Step 8.3.2
ABN-101, Attachment 1

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

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 Line.
- 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 and classification applies?

A. Hazards, Natural or Destructive Phenomena – Unusual Event

B. Hazards, Fire or Explosion - Unusual Event

C. Hazards, Natural or Destructive Phenomena – Alert

D. Hazards, Fire or Explosion - Alert

Proposed Answer: D

Explanation:

A. Incorrect. Plausible if it believed that a Main Steam Line Break is classified as a Destructive Phenomena. However, Note 9 in the EAL chart directs that a steam line break be classified under the Fire or Explosion EAL. Additionally, this fault would be classified as an Alert due to visible damage to the safeguards building.

B. Incorrect. Plausible because Note 9 in the EAL chart directs that a steam line break be classified under the Fire or Explosion EAL. However, this fault would be classified as an Alert due to visible damage to the safeguards building.

C. Incorrect. Plausible because this fault would be classified as an Alert due to visible damage to the safeguards building. However, Note 9 in the EAL chart directs that a steam line break be classified under the Fire or Explosion EAL.

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. Additionally, this fault would be classified as an Alert due to visible damage to the safeguards building.

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	Natural or destructive phenomena affecting the Protected Area
1 | 2 | 3 | 4 | 5 | 6 | DEF	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	**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)
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	**HU1.2 (Bases Page 145)** Tornado striking within the Protected Area boundary **OR** Sustained high winds > 80 mph
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	**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
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	**HU1.4 (Bases Page 149)** Turbine failure resulting in casing penetration or damage to turbine or generator seals

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.

Which of the following procedures should be implemented?

A. Transition to ECA-3.1A, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired.

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 if believed that the ruptured steam generator is not isolated due to steam generator pressure not rising. Additionally, transition to ECA-3.1A would be correct following actions in EOP-3.0A if the steam generator remained faulted. However, the fact that steam generator pressures are stabilizing indicates steam generator isolation. Additionally, ECA-3.1A cannot be entered directly from ECA-2.1A.

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

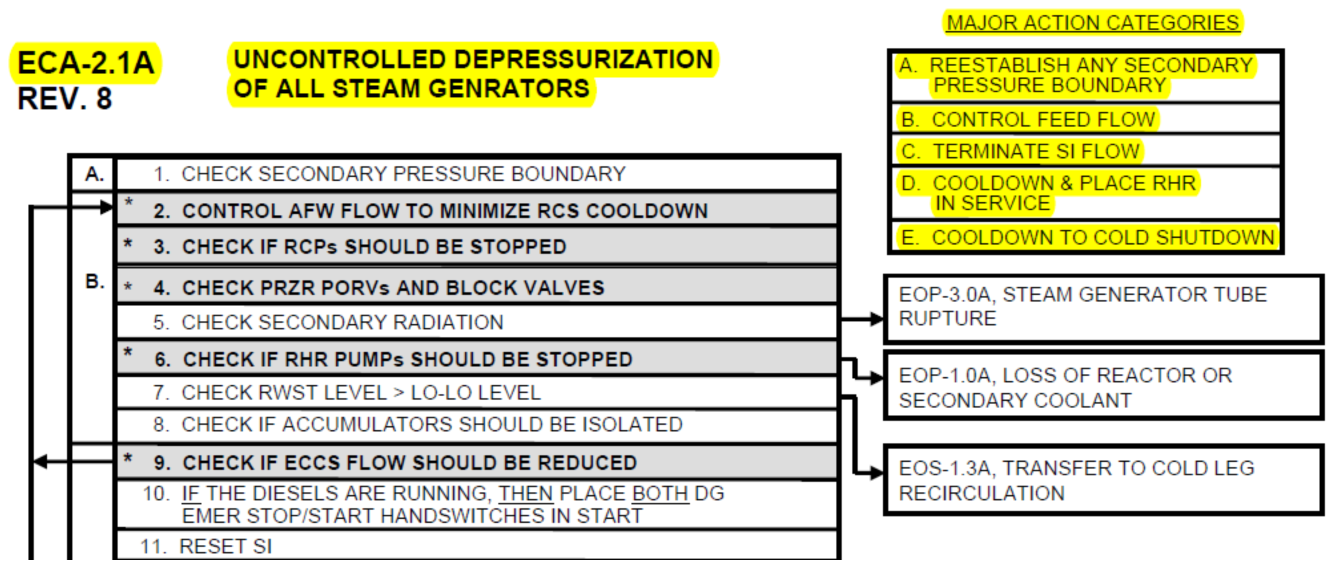
IF any SG level increases in an uncontrolled manner or any SG has abnormal radiation, THEN manually start ECCS pumps as necessary AND go to EOP-3.0A, STEAM GENERATOR TUBE RUPTURE, Step 1.

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

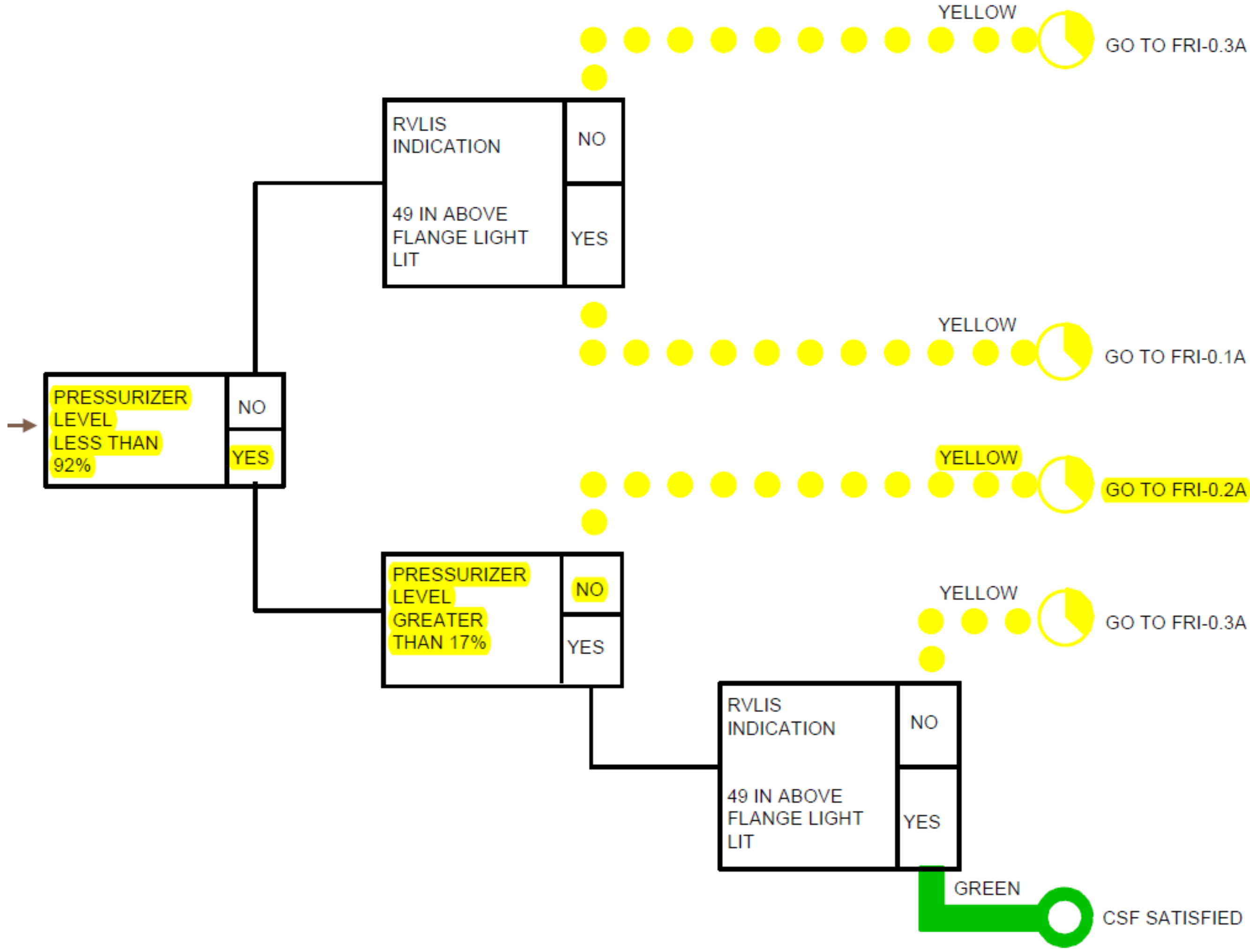
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5	Check Secondary Radiation:	
	a. Request periodic activity samples of all SGs.	
	b. Check any secondary radiation monitor that is NOT isolated - NORMAL: • Condenser off gas radiation (COG-182, 1RE-2959) • Main steamline radiation (MSL-178 through 181, 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

INVENTORY



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:

- ($\geq$40% power) N16 leak rate indication.
- Posted COG correlation graphs by PC-11($\geq$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.
☐ [C] 3 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. Re-actuate Safety Injection and transition to EOP-1.0A, Loss of Reactor or Secondary Coolant.

B. Re-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 normally 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, Steps 1 through 3 of EOS-1.3B must be performed without delay and prior to responding to any FRG which would normally have priority.

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.

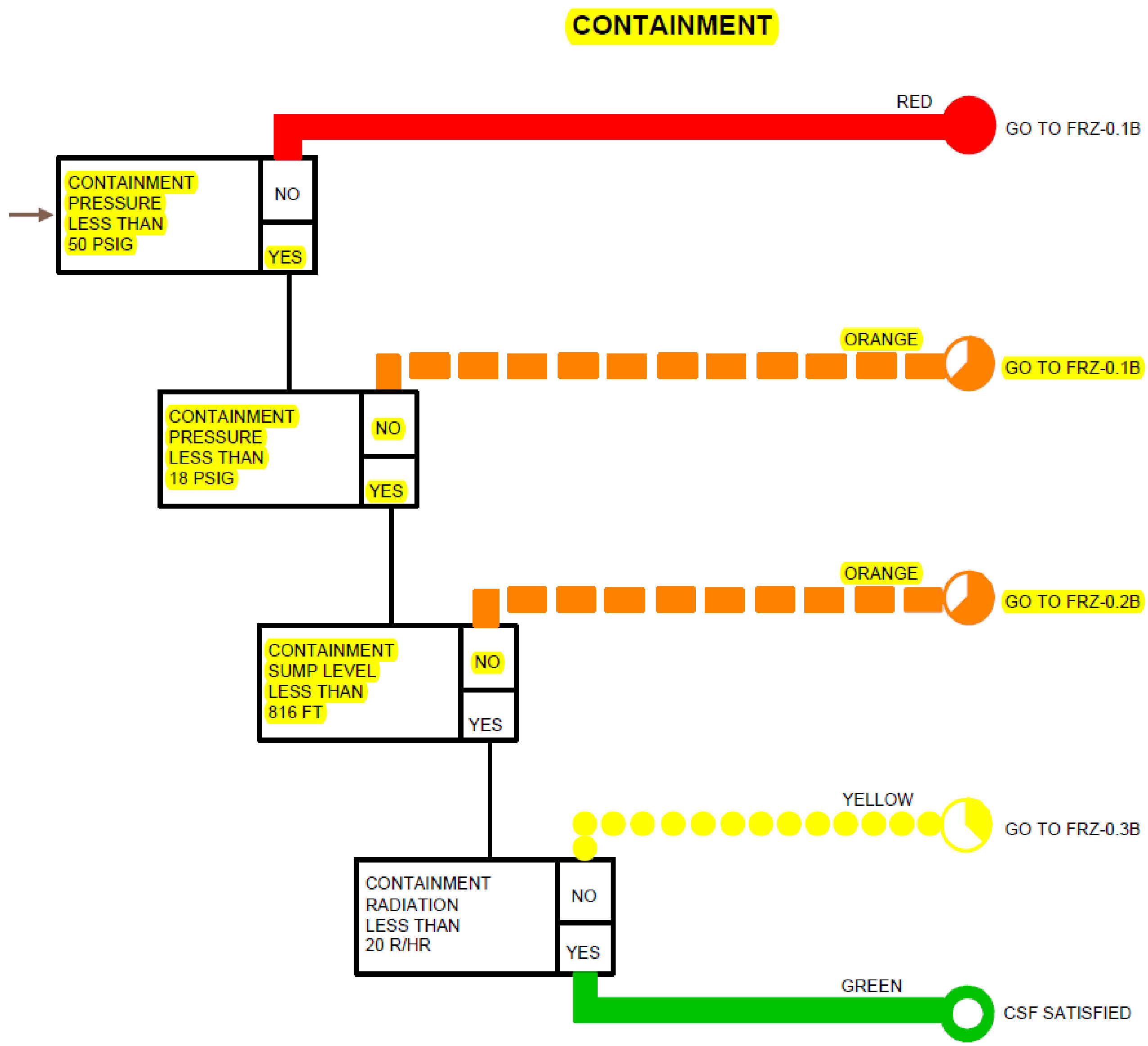
Question Source:
Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

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 35% power and ramping.
- Chemistry reports Steam Generator 2-01 primary to secondary leakage indicates a 120 gpd tube leak.

What indication is used to confirm the Chemistry sample and what is the required action?

A. 2-RE-2959 (COG-282) and the unit should be in MODE 3 ≤ 24 hours.

B. 2-RE-2325A (N16-274) and the unit should be in MODE 3 ≤ 24 hours.

C. 2-RE-2959 (COG-282) and the unit should be in MODE 3 ≤ 2 hours.

D. 2-RE-2325A (N16-274) and the unit should be in MODE 3 ≤ 2 hours.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible as COG-282 is correct for confirmation that a Steam Generator Tube Leak exists, however, in accordance with ABN-106, the unit should be placed in MODE 3 in ≤ 2 hours.
B. Incorrect. Plausible as N16-274 is normally the most accurate indications, however, below 40% power the N16 radiation monitors are removed from poll. Further in accordance with ABN-106, the unit should be placed in MODE 3 in ≤ 2 hours.
C. Correct. Plausible as COG-282 is correct for confirmation that a Steam Generator Tube Leak exists and in accordance with ABN-106, the unit should be placed in MODE 3 in ≤ 2 hours.
D. Incorrect. Plausible as N16-274 is normally the most accurate indications, however, below 40% power the N16 radiation monitors are removed from poll. In accordance with ABN-106, the unit should be placed in MODE 3 in ≤ 2 hours.

Technical Reference(s): IPO-003A, step 5.6.16; ABN-106, Steps 3.3.2 & 3.3.3 — 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 IPO-003A, step 5.6.16	Revision 28

CPNPP INTEGRATED PLANT OPERATING PROCEDURES MANUAL	UNIT 1	PROCEDURE NO. IPO-003A
POWER OPERATIONS	REVISION NO. 28 **CONTINUOUS USE**	PAGE 108 OF 195

NOTE: At ≥ 40%, Main Steamline N-16 Radiation monitors satisfy STA-732 primary-to-secondary leakage detection requirements. The Condenser Off-Gas Radiation monitor provides a reliable backup indication for primary-to-secondary leakage. During Unit shutdown when the Condenser Off-Gas Radiation monitor is operable, Main Steamline N-16 Radiation monitor channel alarms may be disabled at a power level convenient to support manpower availability.

5.6.16 Disable the channel alarms for all Main Steamline N-16 Radiation monitors per SOP-706:

- ☐ N16-174 (1-RE-2325A, MSL 1-01 SG LEAK RATE MONITOR DETECTOR)
- ☐ N16-175 (1-RE-2326A, MSL 1-02 SG LEAK RATE MONITOR DETECTOR)
- ☐ N16-176 (1-RE-2327A, MSL 1-03 SG LEAK RATE MONITOR DETECTOR)
- ☐ N16-177 (1-RE-2328A, MSL 1-04 SG LEAK RATE MONITOR DETECTOR)

_____ / _____
Initials Date

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. Additionally, analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only two charging pumps are actuated. Thus, the LCO allows only two charging pumps OPERABLE during the LTOP MODES.

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 $\geq$ 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.

Comments / Reference: From Technical Specification SR 3.4.12.1 Bases	Revision 67

BASES

APPLICABLE SAFETY ANALYSES (continued)

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all safety injection pumps and one charging pump incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Precluding start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

The analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only two charging pumps are actuated. Thus, the LCO allows only two charging pumps OPERABLE during the LTOP MODES. Since neither one RCS relief valve nor the RCS vent can handle the pressure transient from accumulator injection, when RCS temperature is low, the LCO also requires accumulator isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		2
Group #		2
K/A #	061 G2.2.40	
Importance Rating		4.7

Auxiliary Feedwater: Ability to apply Technical Specifications for a system.

Proposed Question: SRO 89

Given the following conditions:

- Unit 1 at 2% power holding for Main Feedwater Pump 1A to be available.
- Twenty-four hours ago, the Turbine Driven Auxiliary Feedwater Pump failed Surveillance Requirement 3.7.5.2 and is being disassembled.
- Motor Driven Auxiliary Feedwater Pump (MDAFWP) 1-02 trips on overcurrent.
- The Unit Supervisor directs cross-connecting MDAFWP 1-01 to feed all four Steam Generators per ABN-305, Auxilary Feedwater System Malfunctions.

What is(are) the Technical Specification Required Actions?

A. Place Unit 1 in MODE 3 within 7 hours and MODE 4 within 13 hours.

B. Place Unit 1 in MODE 3 within 6 hours and MODE 4 within 18 hours.

C. Place Unit 1 in MODE 3 within 6 hours and restore one AFW train to OPERABLE within 48 hours.

D. Immediately initiate action to restore one AFW train to OPERABLE.

Proposed Answer: D

Explanation:

A. Incorrect. Plausible with the TDAFWP inoperable and one MDAFWP feeding all four Steam Generators, all three AFW pumps are inoperable. This would be the correct action if believed the LCO 3.0.3 were entered for this Condition. However, LCO 3.7.5 Condition D NOTE states that LCO 3.0.3 should not be entered.
B. Incorrect. Plausible if believed that the TDAFWP and MDAFWP 1-02 were the only inoperable pumps as this is the correct actions for two trains of AFW inoperable. However, cross-connecting the MDAFWP 1-01 to feed all four Steam Generators makes MDAFWP 1-01 inoperable as well. This would be the correct action for LCO 3.7.5 Condition C.
C. Incorrect. Plausible as this action would be indicative of a misunderstanding that the TDAFWP is not required OPERABLE in MODE 3 and thus placing the unit in MODE 3 exits the two trains inoperable LCO 3.7.5 Condition C and therefore 48 hours would remain of the 72 hours for Condition B for only one train inoperable. This is not correct, as the allowance for testing the TDAFWP in MODE 3 is for testing purposes only and does not exclude the pump from OPERABILITY requirements in MODE 3.
D. Correct. With the TDAFWP inoperable, MDAFWP 1-02 tripped and MDAFWP 1-01 feeding all four Steam Generators, all three AFW trains are inoperable in accordance with ABN-305 Note. LCO 3.7.5 Condition D is correct for all three trains inoperable.

Technical Reference(s): ABN-305; Technical Specification 3.0.3; Technical Specification 3.7.5 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: LCO 3.7.5

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 7, 10
- 55.43 2

Comments / Reference: From ABN-305, Attachment 3 note	Revision 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-305
AUXILIARY FEEDWATER SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 83 OF 92

ATTACHMENT 3
PAGE 1 OF 3

MD AFW TRAIN CROSS-CONNECTION WITH MD AFWP OPERATING

CAUTION: Operating with the Motor Driven Auxiliary Feedwater trains cross-connected in MODES 1, 2 or 3 violates the train independence of Tech. Spec. 3.7.5.

Comments / Reference: From ABN-305, step 3.3.3	Revision 7

CPNPP ABNORMAL CONDITIONS PROCEDURES MANUAL	UNIT 1 AND 2	PROCEDURE NO. ABN-305
AUXILIARY FEEDWATER SYSTEM MALFUNCTION	REVISION NO. 7	PAGE 47 OF 92

3.3 Operator Actions

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
☐ 3 Verify Steam Generator levels - NORMAL	IF the TD AFW Pump in available, THEN start the TD AFW Pump AND feed the two steam generators NOT being supplied by the MD AFW Pump. If the TD AFW Pump is NOT available, THEN cross connect AFW trains per Attachment 2 or 3 as appropriate.

Comments / Reference: From Technical Specification 3.0.3	Revision 158

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.3

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

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

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

Comments / Reference: From Technical Specification 3.7.5, LCO B	Revision 158

AFW System
3.7.5

3.7 PLANT SYSTEMS

3.7.5 Auxiliary Feedwater (AFW) System

LCO 3.7.5 Three AFW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One steam supply to turbine driven AFW pump inoperable.	A.1 Restore steam supply to OPERABLE status.	7 days
B. One AFW train inoperable for reasons other than Condition A.	B.1 Restore AFW train to OPERABLE status.	72 hours

Comments / Reference: From Technical Specification 3.7.5, LCO C & D	Revision 158

AFW System
3.7.5

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met. OR Two AFW trains inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 18 hours
D. Three AFW trains inoperable.	D.1 -------------------NOTE------------------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status. --- Initiate action to restore one AFW train to OPERABLE status.	Immediately

Comments / Reference: From Technical Specification 3.7.5, SR 3.7.5.2	Revision 158

AFW System
3.7.5

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.5.1	---------------------------------NOTE--------------------------------- AFW train(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. --- Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	In accordance with the Surveillance Frequency Control Program.
SR 3.7.5.2	---------------------------------NOTE--------------------------------- Not required to be performed for the turbine driven AFW pump until 24 hours after $\geq$ 532 psig in the steam generator. --- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice testing Program

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 an 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.

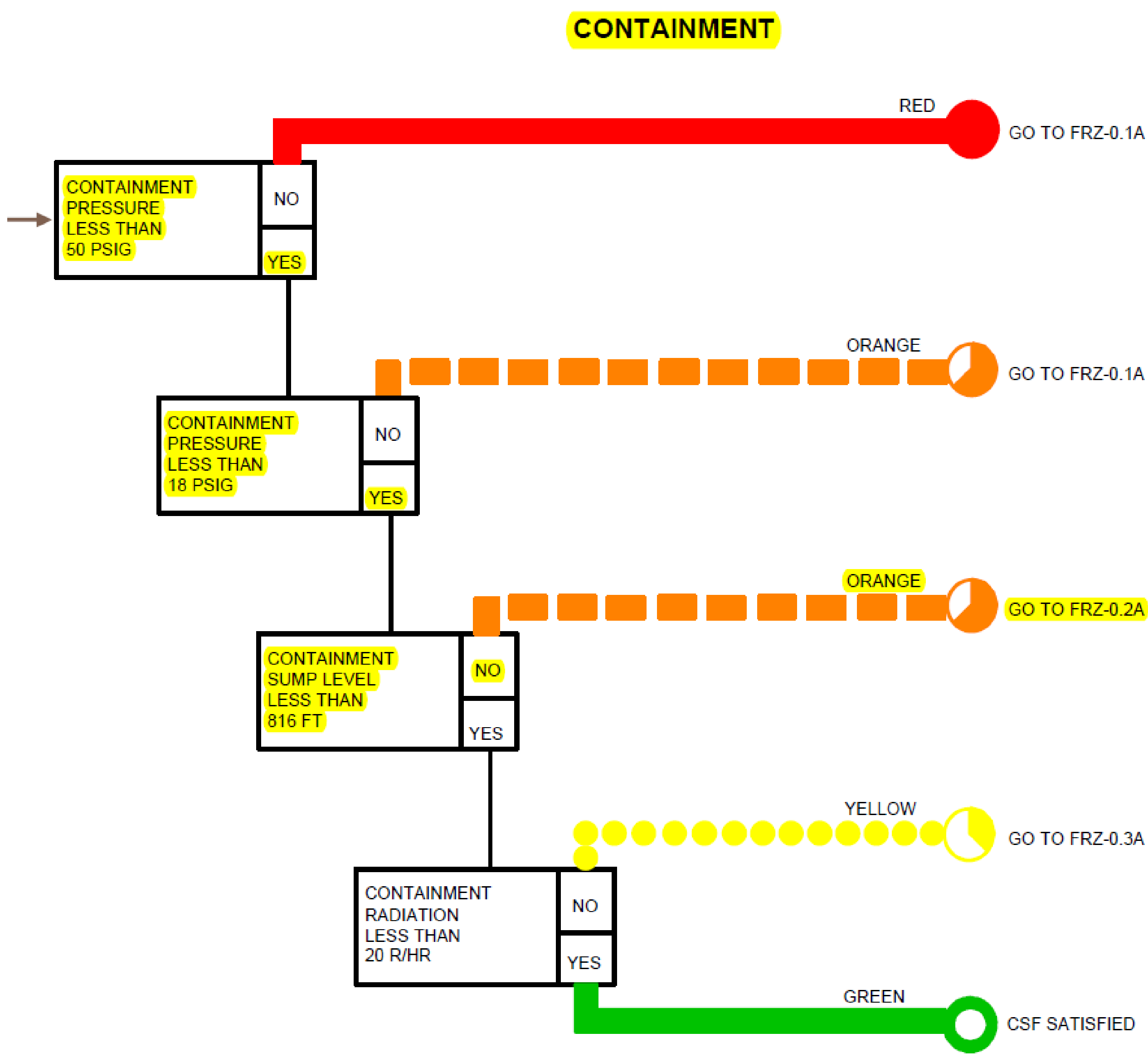
Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History: Last NRC Exam ____

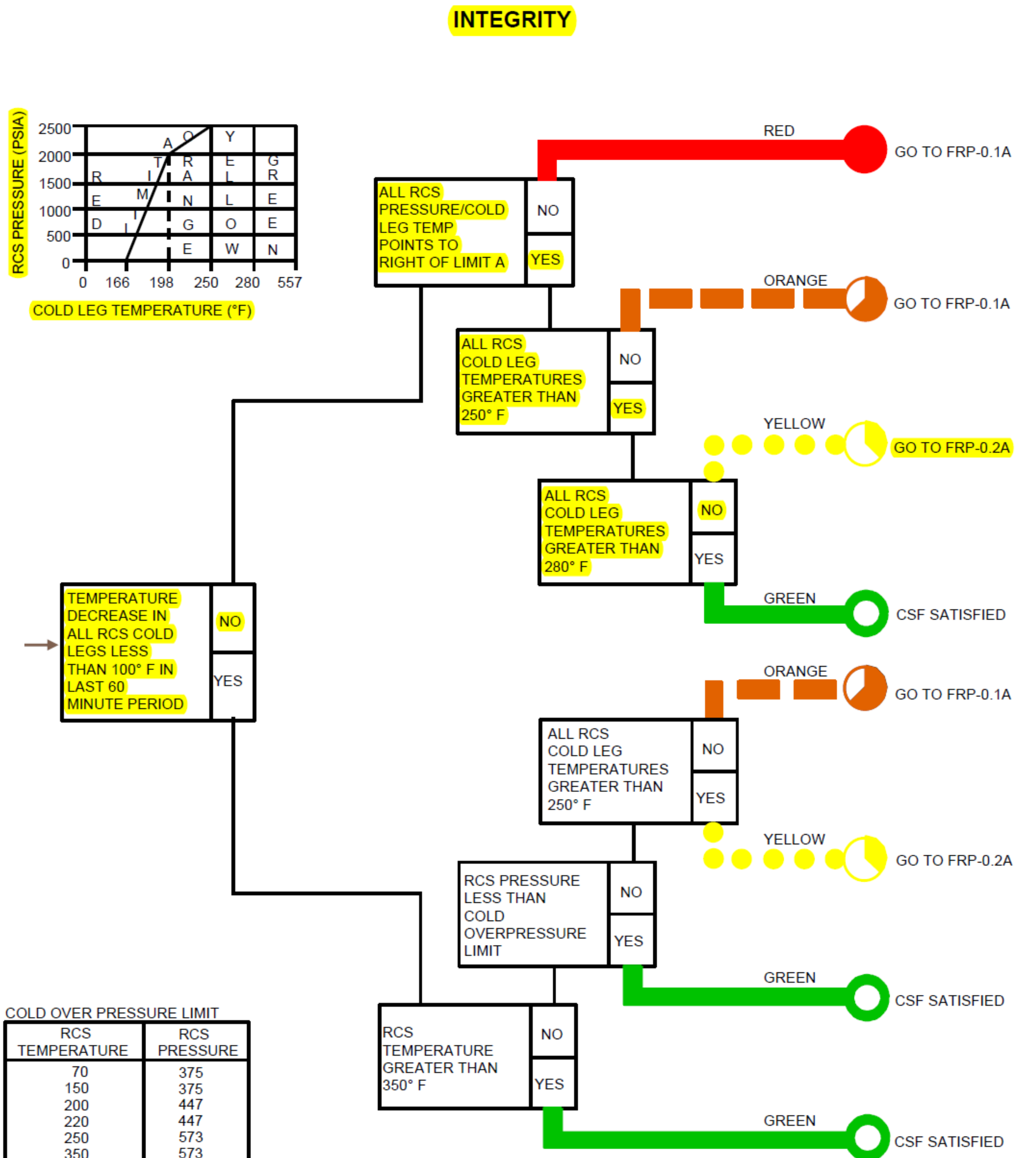
Question Cognitive Level: Memory or Fundamental Knowledge ____
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 ____
55.43 5

Comments / Reference: From FRZ-0.2A, CSFST	Revision # 8



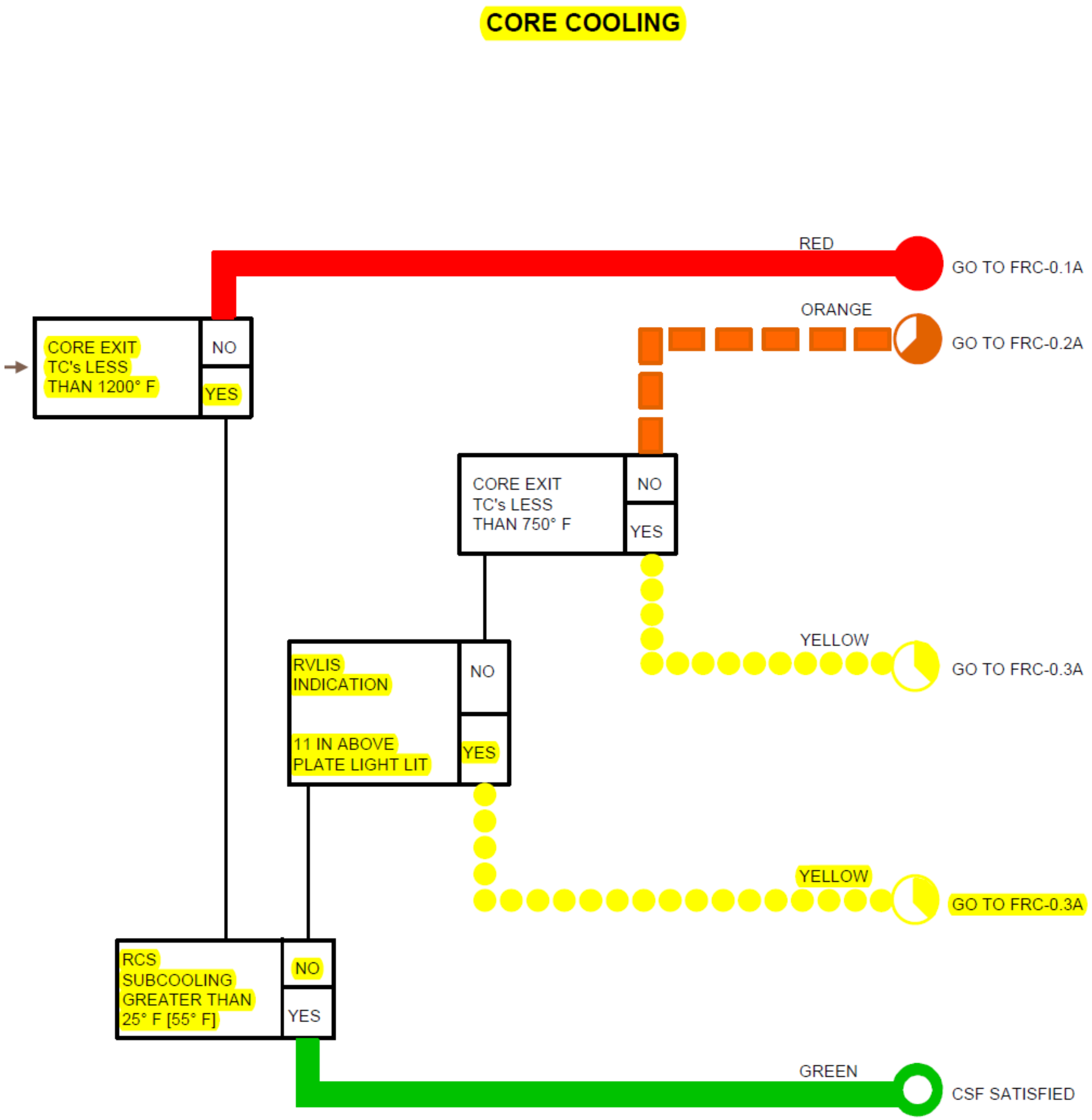
Comments / Reference: From FRP-0.2A, CSFST	Revision # 8



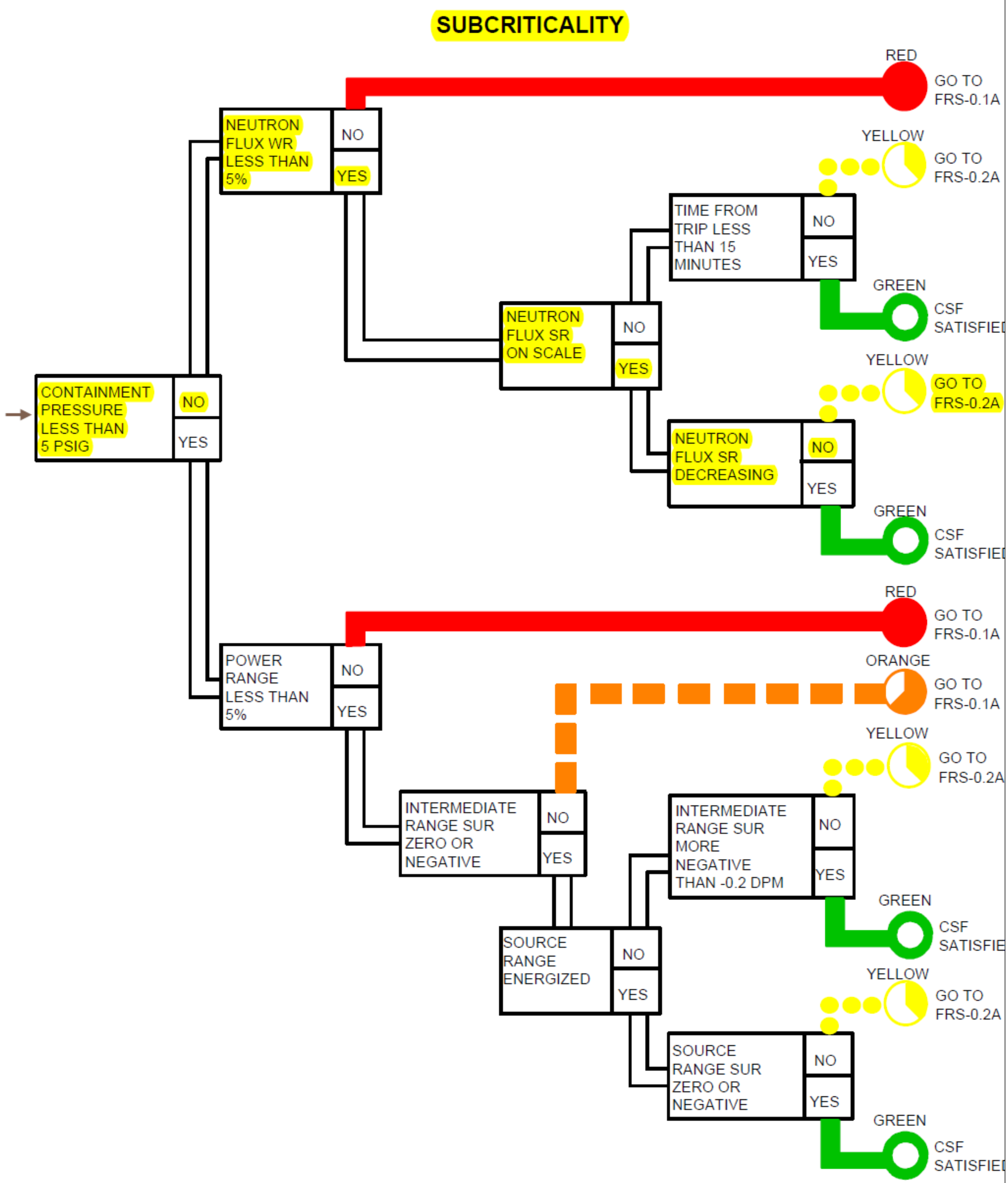
COLD OVER PRESSURE LIMIT

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 #	015 A2.05	
Importance Rating		3.8

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

Proposed Question: SRO 91

Given the following conditions:

- Unit 1 has experienced a Large Break Loss of Coolant Accident during a reactor startup.
- All equipment functioned as designed.
- The crew has transitioned from EOP-0.0A, Reactor Trip or Safety Injection, to EOP-1.0A, Loss of Reactor or Secondary Coolant.
- EOP-0.0A, Attachment 2, Safety Injection Actuation Alignment, has been completed.

Which of the following describes the IMMEDIATE result that voiding in the downcomer region would have on source range instrumentation and the procedure used to mitigate these plant conditions?

A. The displacement of downcomer water would increase the neutron leakage and result in a higher source range count rate.
Continue in EOP-1.0A rather than transition to FRS-0.2A, Response to Loss of Core Shutdown.

B. A decrease in downcomer water density would reduce fission and result in a lower source range count rate.
Continue in EOP-1.0A rather than transition to FRS-0.2A, Response to Loss of Core Shutdown.

C. The displacement of downcomer water would increase the neutron leakage and result in a higher source range count rate.
Transition to FRS-0.2A, Response to Loss of Core Shutdown, rather than continue in EOP-1.0A.

D. A decrease in downcomer water density would reduce fission and result in a lower source range count rate.
Transition to FRS-0.2A, Response to Loss of Core Shutdown, rather than continue in EOP-1.0A.

Proposed Answer: A

Explanation:

A. Correct. Void formation in the downcomer will cause more neutron leakage outside the vessel and neutron flux SR indication will increase. Continuing with EOP-1.0A is the correct action.
B. Incorrect. Plausible because continuing with EOP-1.0A is the correct action, however, void formation in the downcomer will cause more neutron leakage outside the vessel and neutron flux SR indication will increase.
C. Incorrect. Plausible because void formation in the downcomer will cause more neutron leakage outside the vessel and neutron flux SR indication will increase. However, with the conditions listed, the highest transition to FRS-0.2A would be via a yellow path which does not take precedence over EOP-1.0A actions.
D. Incorrect. Plausible if believed that void formation would cause source range neutron instrumentation level to increase due to an increased fission rate, however, void formation in the downcomer will cause more neutron leakage outside the vessel and neutron flux SR indication will increase. Additionally, with the conditions listed, the highest transition to FRS-0.2A would be via a yellow path which does not take precedence over EOP-1.0A actions.

Technical Reference(s) LO21.MCO.MI8 Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: Describe the effects of core uncover on core kinetics and relate the Excore Nuclear System response to voiding.

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
55.43 5

Comments / Reference: From LO21.MCO.MI8	Revision 3/30/04

LO21.MCO.MI8

DOWNCOMER

The water in the downcomer is the only moderator or reflector between the uncovered region of the core and the Excore detector. If the downcomer liquid level is low enough or if the water in the downcomer is highly voided, the neutron shine from the uncovered core can stream unshielded to the Excore detector. In such a situation, the transmission ratio increases by several orders of magnitude.

During an accident in which the core is progressively uncovered, the water level in the downcomer is very likely to be lower than the level of the two-phase mixture in the core. The water in the downcomer is more likely to be a subcooled liquid or at the very least have a much lower void fraction than the two-phase mixture in the core. Thus the downcomer liquid is denser resulting in a lower water column for the same static pressure. Under no flow conditions the two columns will attempt to maintain a hydrostatic equilibrium. The neutron shine is free to stream from the uncovered portion of the core through the voided downcomer to the Excore detector. This means that even before the core mixture level falls to the elevation corresponding to the axial position of the Excore detector, the detector can become exposed to the neutron shine form the uncovered core region. As a result of this phenomenon, the Excore detector will elevate by orders of magnitude early on in core uncovery. If core uncovery progresses, the detector response will stabilize due to the trade-off between a smaller

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	086 A2.02	
	Importance Rating		3.3

Fire Protection System: Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low FPS header pressure

Proposed Question: SRO 92

Given the following conditions:

- X-07, Jockey Fire Pump, was placed in RUN to avoid problems with the pump coupling.
- Fire water header pressure has slowly lowered to 141 psig over several days.
- X-04, Electric Motor Driven Fire Pump, failed to automatically start.
- When X-HS-4091B, ELEC FIRE PMP START pushbutton is depressed, the Electric Motor Driven Fire Pump STARTS and then immediately STOPS when pressure reaches 155 psig.
- X-04, Electric Motor Driven Fire Pump was disabled from AUTO starting and stopped per SOP-904, Fire Protection Main Water Supply and Fire Pumps System.
- X-05, Diesel Fire Pump, has been placed in RUN.

Which of the following describes the impact on the Fire Suppression Water System and what Compensatory Measures that should be implemented?

A. One Fire Pump is inoperable.
Establish a backup fire suppression water supply within 14 days per STA-738, Fire Protection Systems/Equipment Impairments.

B. Two Fire Pumps are inoperable.
Establish a backup fire suppression water supply within 14 days per STA-738, Fire Protection Systems/Equipment Impairments.

C. One Fire Pump is inoperable.
Restore the Electric Fire Pump to OPERABLE status within 7 days or provide an alternate backup pump or supply per STA-738, Fire Protection Systems/Equipment Impairments.

D. Two Fire Pumps are inoperable.
Restore the Electric Fire Pump to OPERABLE status within 7 days or provide an alternate backup pump or supply per STA-738, Fire Protection Systems/Equipment Impairments.

Proposed Answer: C

Explanation:

A. Incorrect. Plausible because one fire pump is inoperable and a backup fire suppression water supply would satisfy STA-738 requirements, however, this must be done within 24 hours.
B. Incorrect. Plausible because establishing a backup fire suppression water supply would satisfy STA-738 requirements, however, placing a diesel driven fire pump in run does not render the pump inoperable. Only one fire pump is inoperable.
C. Correct. As outlined in SOP-904 and STA-738.
D. Incorrect. Plausible if thought that the Jockey Fire Pump was also considered inoperable or that running the Diesel Fire Pump rendered it inoperable. Otherwise, the action is correct.

Technical Reference(s) STA-738, Attachment 8.A — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **COMPREHEND** the administrative requirements of abnormal operations of the Fire Protection 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
55.43 5

Comments / Reference: From STA-738, Attachment 8.A	Revision 6

CPSES STATION ADMINISTRATION MANUAL		PROCEDURE NO. STA-738
FIRE PROTECTION SYSTEMS/EQUIPMENT IMPAIRMENTS	REVISION NO. 6	PAGE 13 OF 45

ATTACHMENT 8.A
PAGE 1 OF 6

GUIDELINES FOR COMPENSATORY MEASURES

THE FOLLOWING COMPENSATORY MEASURES SHALL BE IMPLEMENTED WHEN FIRE PROTECTION SYSTEMS/EQUIPMENT ARE DETERMINED IMPAIRED OR INOPERABLE AS DESCRIBED BY THIS PROCEDURE.

1) FIRE SUPPRESSION WATER SYSTEM

a) With one pump and/or one water supply tank inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply.

[C] 2) FIRE DETECTION

a) With any, but not more than one-half the total in any fire zone Function A fire detection instruments shown in Attachment 8.B inoperable, the inoperable instrument(s) shall be restored to an operable status within 14 days or within the next 1 hour a fire watch patrol shall be established to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside Containment or within Zone V radiation areas outside Containment, then that zone shall be inspected at least once per 8 hours or monitor the containment air temperature at least once per hour by the Control Room Indicators.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	002 G 2.4.18	
	Importance Rating		4.0

Reactor Coolant System: Knowledge of the specific bases EOPs.

Proposed Question: SRO 93

Given the following conditions on Unit 1:

- 1-PCV-455B, RC LOOP 1 PRZR SPR VLV has failed opened.
- An automatic Reactor Trip occurred on Pressurizer Pressure Low prior to performing ABN-705, Pressurizer Pressure Malfunction.
- Safety Injection actuated on Pressurizer Pressure Low at 1820 psig.
- EOP-0.0A, Reactor Trip or Safety Injection is in progress.
- Pressurizer Level is 25% and stable.

What action is required per EOP-0.0A Step 10 when 1-PCV-455B cannot be closed?

A. Stop Reactor Coolant Pumps 1-01 and 1-04.

B. Stop Reactor Coolant Pump 1-01 only.

C. Stop Reactor Coolant Pumps 1-04 and 1-02.

D. Stop Reactor Coolant Pump 1-04 only.

Proposed Answer: A

Explanation:

A. Correct. In accordance with EOP-0.0A Bases Table 1 stopping RCPs 1-01 and 1-04 will essentially stop spray flow through loop 1 spray valve. With pressurizer level at 25% spray flow will be negligible from RCPs 1-02 and 1-03.
B. Incorrect. Plausible as RCP 1-01 is the primary driving head for spray flow through 1-PCV-455B; however the EOP-0.0A Bases Table 1 shows that RCP 1-04 will still supply spray flow and therefore RCP 1-04 should also be stopped.
C. Incorrect. Plausible as RCP 1-04 is known to produce the greatest spray flow and therefore is logical to stop. Additionally stopping RCP 1-02 would lower the driving head farther. However, since 1-PCV-455B is primarily driven from the discharge of RCP 1-01, spray flow is maintained with RCPs 1-01 and 1-03 running.
D. Incorrect. Plausible as RCP 1-04 is known to produce the greatest spray flow and therefore is logical to stop. However, since 1-PCV-455B is primarily driven from the discharge of RCP 1-01, spray flow is maintained with RCPs 1-01, 1-02 and 1-03 running.

Technical Reference(s) EOP-0.0A, step 10 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective: **DISCUSS** EOP-0.0, Reactor Trip or Safety Injection including the purpose, applicability, symptoms/entry conditions, operator actions, bases, foldout pages and attachments.

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 EOP-0.0A, step 10	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 9 OF 115

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10	Check PRZR Valve Status:	
	a. PRZR Safeties - CLOSED	a. Go to EOP-1.0A, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	b. Normal PRZR spray valves - CLOSED	b. IF PRZR pressure less than 2235 psig, THEN manually close valve(s) as necessary. IF valve(s) can NOT be closed, THEN stop RCP(s) as necessary to stop spray flow.

Comments / Reference: From EOP-0.0A, Attachment 10 Bases	Revision 8

CPSES EMERGENCY RESPONSE GUIDELINES	UNIT 1	PROCEDURE NO. EOP-0.0A
REACTOR TRIP OR SAFETY INJECTION	REVISION NO. 8	PAGE 88 OF 115

ATTACHMENT 10
PAGE 9 OF 36

BASES

If the PRZR pressure is below 2235 psig and any normal PRZR spray valve cannot be closed, the operator is instructed to stop RCPs as necessary to terminate the spray flow. It may be necessary to stop two (or more) RCPs in order to minimize the RCS depressurization, depending on which spray valve is failed open, the availability of pressurizer heaters and the existing pressurizer level.

If RCP 4 (loop with PRZR surge line) can be started, then it alone should be sufficient to provide normal PRZR spray. However, if RCP 4 is unavailable, it will likely be necessary to start more than one RCP to provide normal PRZR spray. Analysis performed for RCP operation and spray flow results provides the following conclusions (Reference RCP TRIP/RESTART in the Generic Issues section of the Executive Volume) (See Table 1). Spray flow with any combination of RCPs operating will be more effective with a high PRZR level. Additionally, operating experience has shown that RCP vibration may be higher than normal when only one RCP is running, and that vibration is reduced when a second RCP is started.

TABLE 1

RCP(s) Running	Is Spray Flow Produced?	
	Spray Valve Loop 1 OPEN	Spray Valve Loop 4 OPEN
4	YES	YES
1	YES (1)	NO
1 AND 2 AND 3	YES	YES
1 AND 2	YES	MAYBE (1)
1 AND 3	YES	MAYBE (1)
2 AND 3	MAYBE (1)	MAYBE (1)

(1) Small amount of spray flow is produced when PRZR level is high (e.g., 90%)

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		1
K/A #	G 2.1.4	
Importance Rating		3.8

Conduct of Operations: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: SRO 94

Given the following conditions:

- A Senior Reactor Operator (SRO) is completing the Shift Operation’s Watch Bill for the oncoming crew.
- The SRO is reviewing four Staff Reactor Operator’s (RO) work history to determine if each has met the MINIMUM requirements for maintaining an Active License status in accordance with ODA-315, Licensed Operator Maintenance Tracking.

Which of the following meets the MINIMUM requirements for ensuring that the ROs have satisfied the ODA-315 requirements?

The operator completed five 12-hour shifts...

A. ...during the previous quarter including turnovers, with one four hour absence for makeup of a missed training simulator scenario.

B. ...during the previous quarter including turnovers, during each shift the operator had to utilize Short Term Relief to attend a daily meeting.

C. ...with the fifth shift beginning at 1800 on the last day of the calendar quarter including turnovers.

D. ...during the previous quarter, during one shift the operator had a family emergency and was excused from end of shift turnover.

Proposed Answer: B

Explanation:

A. Incorrect. Plausible as this is 5 12-hour shifts including turnover. However, the four hour absence is in excess of the allowed Short Term Relief as allowed per OWI-107 and thus that shift would not count.

B. Correct. Per ODA-315, 5 12-hour shifts including turnover are the minimum to maintain an Active License status. Allowances for Short Term Relief per OWI-107 are allowed when completing the shifts per ODA-315.

C. Incorrect. Plausible as this is 5 12-hour shifts. However, in accordance with ODA-315, they must all be completed in the previous quarter.

D. Incorrect. Plausible as this is 5 12-hour shifts. However, both turnovers must be included per ODA-315 in order to be counted.

Technical Reference(s) ODA-315, section 6.2; OWI-107, section 6.1.4 — Attached w/ Revision # See Comments / Reference

Proposed references to be provided during examination: None

Learning Objective:

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

Question History: Last NRC Exam CPNPP 2012

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

10 CFR Part 55 Content: 55.41 10
55.43 2

Comments / Reference: From ODA-315, section 6.2	Revision 6

CPNPP OPERATIONS DEPARTMENT ADMINISTRATION MANUAL		PROCEDURE NO. ODA-315
LICENSED OPERATOR MAINTENANCE TRACKING	REVISION NO. 6 INFORMATION USE	PAGE 8 OF 14

6.2 Maintaining an Active License Status

NOTE:	Completion of the requirements of this section are documented on ODA-315-1, “Active License Status Form”.

6.2.1 The requirements by which the Operator maintains an Active License are as follows:

- The Operator shall maintain “Current” license status in accordance with TRA-203 and TRA-204, and
- The Operator shall complete at least five (5) 12-hour shifts from shift turnover to shift turnover each calendar quarter. The Operator must complete each watch in a position that requires an Active License commensurate with the license held by the individual. Positions which require an Active License are described in Attachment 8.A.

EXAMPLES

An Operator that completes five (5) 12-hour shifts from Beginning of Shift turnover to End of Shift turnover per OWI-107 in the same calender quarter has satisfied the minimum requirements to maintain an Active License status.

IF an Operator performs a turnover to another Operator per OWI-107 during a 12-hour shift (anytime between Beginning of Shift turnover to End of Shift turnover), THEN the shift CAN NOT be counted towards one (1) of the five (5) 12-hour shifts for that calender quarter.

IF an Operator performs a Short Term Relief per OWI-107 during a 12-hour shift (anytime between Beginning of Shift turnover to End of Shift turnover), THEN the shift CAN be counted towards one (1) of the five (5) 12-hour shifts for that calender quarter.

IF an Operator begins a shift at 1800 on the final day of the calender quarter, THEN the shift CAN NOT be counted towards one (1) of the five (5) 12-hour shifts for the current quarter OR towards the next quarter since the shift is split between the current and new quarter.

Comments / Reference: From OWI-107, section 6.1.4	Revision 8

CPNPP OPERATIONS DEPARTMENT WORK INSTRUCTIONS		PROCEDURE NO. OWI-107
OPERATIONS DEPARTMENT TURNOVER AND BRIEFING INSTRUCTIONS	REVISION NO. 8 INFORMATION USE	PAGE 10 OF 15

6.1.4 Short Term Relief

Short term reliefs should be used whenever shift operating personnel will be out of the "at the controls" area for periods projected to last for ≤ 60 minutes. This is not required for brief absences such as trips to the rest room, kitchen, CPC, SM office, etc. Short term relief may occur only if all of the following conditions are met:

- The on-coming person is qualified for the position.
- The on-coming person is knowledgeable of all pertinent activities in progress.
- A joint board walkdown is performed, as applicable, and
- The SM grants permission if a Supervisor requires relief. The Unit Supervisor grants permission if the RO or BOP assigned to their Unit requires relief.

Reliefs expected to last for a longer duration should be performed by completion of the associate relief checklist for that position and should be documented in the Narrative Log Module.

Examination Outline Cross-reference:

Level	RO	SRO
Tier #		3
Category #		2
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 #		2
	K/A #	G 2.2.35	
	Importance Rating		4.5

Equipment Control: Ability to determine Technical Specification Mode of Operation

Proposed Question: SRO 96

Unit 1 is commencing a core reload from a full offload.

Which of the following describes the time at which core offload to MODE 6 is declared per RFO-102, Refueling Operation?

A. When Containment Closure is established in preparation for core reload.

B. When the first new fuel assembly is placed in the Containment Storage Rack.

C. When the first fuel assembly is engaged by the Refueling Machine Main Hoist in preparation for placement in the core.

D. When the first fuel assembly is engaged by the Refueling Machine Main Hoist is placed in its designated position in the core.

Proposed Answer: C

Explanation:
A. Incorrect. Plausible because Containment Closure is established per OPT-408A prior to MODE 6 entry in RFO-102.
B. Incorrect. Plausible because it is specifically excluded as a MODE 6 entry criteria by RFO-102.
C. Correct. Per Step 5.5.4.A of RFO-102 when the first assembly is engaged in the Main Hoist MODE 6 is declared.
D. Incorrect. Plausible if thought that MODE 6 is not entered until an assembly is placed in the core.

Technical Reference(s) RFO-102 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 RFO-102	Revision 13

CPNPP STATION REFUELING MANUAL	UNIT COMMON	PROCEDURE NO. RFO-102
REFUELING OPERATION	REVISION NO. 13 CONTINUOUS USE	PAGE 61 OF 96

5.5.3 Prior to conducting CORE ALTERATIONS, PERFORM the following: (ODA-308-3.9.0 is also available)

A. ENSURE the Refueling Machine Main Hoist has been tested within 100 hours prior to initiating fuel movement per RFO-501 (TRS 13.9.33.1). ____

B. ENSURE the Fuel Building Bridge Crane has been load tested within 7 days prior to initiating Fuel Building fuel transfer operations per RFO-402 (TRS 13.9.34.1). ____

C. VERIFY Containment Closure is established per OPT-408A/B (SR 3.9.4.1, 3.9.4.2, 3.9.4.3). ____

NOTE:	Placing new fuel assemblies in the In Containment Storage Rack does NOT constitute Mode 6 entry.

5.5.4 IF the core has been off loaded,
THEN
PERFORM the following:

A. WHEN the first fuel assembly is engaged by the Refueling Machine Main Hoist in preparation for placement in the core,
THEN
LOG the time MODE 6 is entered.

Comments / Reference: From Technical Specification LCO 3.0.4	Amendment # 150

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 #		3
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
ECA-1.2A, Entry Conditions
EOS-0.0A, Entry Conditions
EOS-1.1A, Attachment 1.A

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 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 #		3
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. 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 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
	<u>AND</u>	
	B.1.2 Enter applicable Conditions and Required Actions for one CREFS train made inoperable by inoperable CREFS actuation instrumentation	Immediately
	<u>OR</u>	
	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 #		4
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 Operations 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. Approving shift schedules that support long-term emergency response.

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 4.1.1 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 & 4.2 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

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-109
DUTIES AND RESPONSIBILITIES OF THE EMERGENCY COORDINATOR/RECOVERY MANAGER	REVISION NO. 14 INFORMATION USE	PAGE 3 OF 8

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.
- Approve shift schedules that support long-term emergency response.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Category #		4
	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 conditions:

- At 0200, the Shift Manager (acting as Emergency Coordinator) declared an ALERT based on RCS leakage of 200 gpm.
- At 0211, initial notifications to Offsite Agencies were completed.
- At 0300, the Unit 1 Reactor was tripped and Safety Injection actuated due to increased RCS leakage.
- At 0309, the RCS became saturated and all RVLIS lights indicated DARK.
- At 0327, the Shift Manager (acting as the Emergency Coordinator) declared an escalation of the event to a SITE AREA EMERGENCY.
- At 0339, notification of Emergency Classification escalation to Offsite Agencies was completed.

Which of the following statements is correct regarding the escalation and escalation notification?

A. The escalation was NOT timely.
The notification was timely.

B. The escalation was timely.
The notification was NOT timely.

C. The escalation was timely.
The notification was timely.

D. The escalation was NOT timely.
The notification was NOT timely.

Proposed Answer: A

Explanation:

A. Correct. The escalation was made in 18 minutes which is NOT timely and the notification was made in 12 minutes which is timely.

B. Incorrect. Plausible if thought that escalation of an event has more time available than initial classification and that notification needs to be immediate as is required for an attack against the station (NRC Operations Center).

C. Incorrect. Plausible if thought that escalation of an event has more time available than initial classification and the notification was made in less than 15 minutes and if thought that at notification needs to be immediate as is required for an attack against the station (NRC Operations Center).

D. Incorrect. Plausible because the escalation was not timely (18 minutes) and if thought that notification needs to be immediate as is required for an attack against the station (NRC Operations Center).

Technical Reference(s) EPP-203, Step 4.1.2.1 & 4.1.5 Attached w/ Revision # See Comments / Reference
EPP-201, Step 4.3

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 4.1.2.1	Revision 16

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-203
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4.0 INSTRUCTIONS

4.1 General Information [C-05708]

4.1.1 Following declaration of any emergency classification, identify at least one Plant Page Party System line to transmit information during the emergency.

4.1.2 Notify Somervell County, Hood County, and DPS of any emergency at CPNPP using the dedicated ring-down telephone system.

4.1.2.1 Notification shall be made **within 15 minutes** if any of the following conditions occur:

- Initial emergency classification;
- Escalation of emergency classification;

Comments / Reference: From EPP-203, Step 4.1.5	Revision 16

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-203
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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.

Comments / Reference: From EPP-201, Step 4.3	Revision 12

CPNPP EMERGENCY PLAN MANUAL		PROCEDURE NO. EPP-201
ASSESSMENT OF EMERGENCY ACTION LEVELS EMERGENCY CLASSIFICATION AND PLAN ACTIVATION	REVISION NO. 12 INFORMATION USE	PAGE 8 OF 16

4.3 Emergency Classification Initial Actions [C-08621]

NOTE: Once indication of an abnormal condition is available, classification declaration must be made within 15 minutes. This time is available to ensure that the classification and subsequent actions associated with the classification, if warranted, are appropriate. It does not allow a delay of 15 minutes if the classification is recognized to be necessary.

It is meant to provide sufficient time to accurately assess the emergency conditions and then evaluate the need for an emergency classification based on the assessment performed. The decision to terminate the event or enter Recovery is NOT time independent.

NOTE: IF a higher classification is made prior to transmitting an event notification, THEN notification for the higher classification can supersede the event notification, provided that it can be performed within the 15-minute timeframe of the previous event. IF the notification of the higher classification cannot be performed within the 15-minute timeframe of the previous event classification, THEN the previous event notification is required within its 15-minute timeframe, and the subsequent event notification is required within its 15-minute timeframe.