Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/22/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	003 K5.05	
Level of Difficulty: 2	Importance Rating	2.8	

Reactor Coolant Pump System: Knowledge of the operational implications of the following concepts as they apply to the RCPS: The dependency of RCS flow rates upon the number of operating RCPs

Question: 1

Which of the following is the reason for limiting Reactor Coolant System flow when less than 500°F per OI-RC-9, Reactor Coolant Pump Operation?

A. Prevent deflection of the lower core support plate.

B. Minimize running amperage on the associated 4160 V Bus.

C. Prevent fuel damage from excessive fuel assembly axial stress and core lift.

D. Minimize core support barrel vertical deflection.

Answer: C

K/A Match:
Applicant must know the operational implication of starting the 4th Reactor Coolant Pump (increased RCS flow with higher coolant density) and the reason why this limitation exists (core lift).

Explanation:

A. Incorrect. Plausible because fluid density when less than 500°F is the condition that causes core lift. The concern is not the lower core plate but rather the entire lifting of fuel assemblies and the resultant axial stress.

B. Incorrect. Plausible because fluid density when less than 500°F will result in higher starting and running currents. One Reactor Coolant Pump resides on each of the 4160 V Buses, and starting a reactor coolant pump does result in higher current on the associated bus, but incorrect because higher current is not the limiting factor, as it would take significantly more dense fluid to create running current problems on the bus.

C. **Correct**. As described in Precaution 13 of OI-RC-9. This limitation is imposed when starting the 4th Reactor Coolant Pump. RCS flow rates associated with 4 Reactor Coolant Pumps operating could cause fuel damage due to the lifting associated with this higher density fluid.

D. Incorrect. Plausible because core support barrel vertical deflection can occur due to the distribution of RCP discharge around the reactor vessel due to fluid density, but does not result in core lift or fuel assembly axial stress.

Technical Reference: OI-RC-9, Precaution 13, Rev. 78
(Attach if not previously provided including revision number) LP 7-11-20, Slide #49, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-11-20, Reactor Coolant System-Licensed Operator EO 1.5 - **DISCUSS** the varying flow rate combinations, due to vessel differential pressure, in the RCS as percentages of normal one RCP flow.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	2
	55.43	

Precaution 13 from OI-RC-9

13. No more than three RCP's shall be operated when the Reactor Coolant temperature is less than 500°F to prevent fuel damage from excessive fuel assembly axial stress and core lift.

EO 1.5 *Detailed Component Description* **(Slide #49)**

The vessel differential pressure causes uneven flow through the system during reduced pump availability.

Flow numbers are given as percentages of one pump normal flow, assuming that four pumps are running.

(1) Normal four pump flow

(a) Only allowed above 500ºF due to excessive core uplift.

(2) Three-pump flow

(a) RC-3D is taking a suction on RC-3C through the steam generator.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	004 A3.03	
Level of Difficulty: 3	Importance Rating	2.9	

Chemical and Volume Control System: Ability to monitor automatic operation of the CVCS, including: Ion exchange bypass

Question: 2

The MAXIMUM letdown fluid temperature out of the ___(1)___ Heat Exchanger when TCV-211-2, Ion Exchanger Bypass Valve, receives a closed signal is ___(2)___.

A. (1) Regenerative
(2) 130°F

B. (1) Letdown
(2) 130°F

C. (1) Regenerative
(2) 140°F

D. (1) Letdown
(2) 140°F

Answer: D

K/A Match:

Applicant must know the flow path of Letdown as well as various temperature setpoints that initiate actions in the CVCS.

Explanation:

A. Incorrect. Plausible because a VCT high temperature alarm comes in at 130°F and part of the Annunciator Response Procedure is to place TCV-211-2, Ion Exchanger Bypass Valve, is in BYPASS IX position.

B. Incorrect. Plausible because the Letdown Heat Exchanger is correct, a VCT high temperature alarm comes in at 130°F.

C. Incorrect. Plausible because the isolation temperature is correct but it is Letdown at the outlet of the Letdown Heat Exchanger not the Regenerative Heat Exchanger.

D. **Correct**. When Letdown temperature out of the Letdown Heat Exchanger reaches 140°F, TCV-211-2 will automatically close to preserve ion exchange resin.

Technical Reference: LP 7-11-12, Slides #96, #97, #98, & #128, Rev. 2
(Attach if not previously provided including revision number) ARP-CB-1/2/3/A2, Window A-2U – VCT TEMP HI, Rev. 42a

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-12, Chemical and Volume Control System-Licensed Operator
EO 1.2 - **EXPLAIN** the manual and automatic functions of control valve in the CVCS.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

EO *1.2, 5.1, *5.2 **(Slide #96)**

Major Component Description

Letdown Temperature Indicating Controller (TIC-211)

TIC-211 provides temperature indication on CB-1/2/3.

(LC) Actuates "LETDOWN HEAT EXCH TUBE OUTLET TEMP HI" alarm on annunciator A2 at 140ºF.

Causes TCV-211-2 to shift to bypass the ion exchangers at 140ºF.

Excess flow or inadequate cooling will result in high letdown temperatures.

EO *1.2, 5.1 **(Slide #98)**

Major Component Description

Ion Exchanger Bypass Valve (TCV-211-2)

TCV-211-2 is normally controlled on CB-1/2/3 by a three-way switch (BYP IX – TO IX – RESET).

Automatically bypasses the ion exchangers (at 140ºF) to protect the resin from high temperature.

It must be manually reset when temperature returns to <140ºF to establish flow through the ion exchangers.

The BYP IX (bypass) position diverts the flow around all of the ion exchangers to protect the resin.

EO 5.1, *5.2 **(Slide #128)**

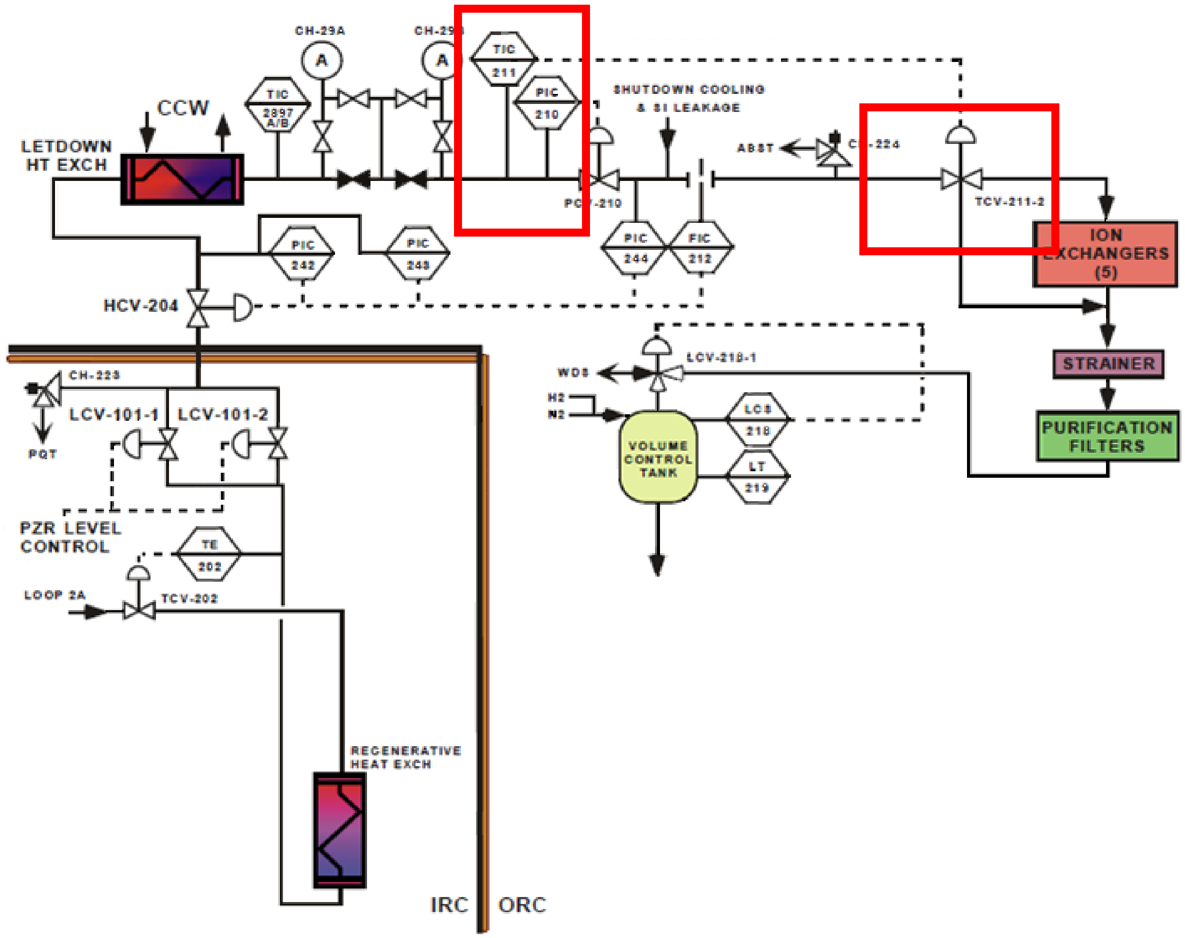
Major Component Description

Volume Control Tank (VCT) (CH-14)

TIA-221 (temperature indication & alarm) provides temperature indication on CB-1/2/3 (50-200°F).

Provides the VCT TEMP HI alarm on annunciator A2. (130ºF)

(Slide #97)


CH-29A
TIC 211
PIC 210
SHUTDOWN COOLING & SI LEAKAGE
CCW
TIC 2897 A/B
LETDOWN HT EXCH
ABST
TCV-211-2
PIC 244
FIC 212
PIC 242
PIC 243
ION EXCHANGERS (5)
HCV-204
STRAINER
LCV-218-1
WDS
H2
N2
CH-223
LCV-101-1
LCV-101-2
LCS 218
PURIFICATION FILTERS
PQT
VOLUME CONTROL TANK
LT 219
PZR LEVEL CONTROL
TE 202
LOOP 2A
TCV-202
REGENERATIVE HEAT EXCH
IRC
ORC

Panel: **CB-1/2/3**	Annunciator: **A2**	Window: **A-2U**

VOLUME CONTROL TANK HIGH TEMPERATURE

SAFETY RELATED

VOLUME CONTROL TANK TEMP HI

Tech Spec References: 2.2

Initiating Device TIA-221 Setpoint >130ºF Power AI-42A

OPERATOR ACTIONS

14. Check Volume Control Tank temperature on the following indicators:

VCT Temperature	TIA-221 ERF T221

15. Ensure TCV-211-2, Ion Exchanger Bypass Valve, is in BYPASS IX.

16. Check the Letdown Temperature (TIC-211).

    16.1 IF Letdown Temperature is high, THEN control letdown temperature per OI-CH-1, Attachment 12.

17. Check Regenerative Heat Exchanger Outlet Temperature (TIC-202).

    17.1 IF Regen HX Outlet Temperature is high, THEN balance Charging and Letdown Flows per OI-CH-1.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	004 K2.01	
Level of Difficulty: 3	Importance Rating	2.9	

Chemical and Volume Control System: Knowledge of bus power supplies to the following: Boric acid makeup pumps

Question: 3

Which of the following identifies the power supplies to the Boric Acid Pumps?

	CH-4A	CH-4B
A.	MCC-3C2	MCC-4A2
B.	MCC-3B2	MCC-4B2
C.	MCC-3B2	MCC-4A2
D.	MCC-3C2	MCC-4B2

Answer: A

K/A Match:

Applicant is asked the power supply to the Boric Acid Pumps.

Explanation:

A. **Correct**. The Boric Acid Pumps are powered from 480 VAC MCCs located in Corridor 26 of the Auxiliary Building.
B. Incorrect. Plausible because these are both 480 V MCCs.
C. Incorrect. Plausible because the power supply for CH-4B is correct.
D. Incorrect. Plausible because the power supply for CH-4A is correct.

Technical Reference: LP 7-11-12, Slide #211, Rev. 2
(Attach if not previously provided including revision number) AOP-32, Attachment B, Steps 21, 24, 28, & 31, Rev. 21

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-12, Chemical and Volume Control System-Licensed Operator EO 1.3 - **EXPLAIN** the automatic and manual controls associated with the charging pumps and boric acid pumps.

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

Question History:	Last NRC Exam	None
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	6
	55.43	

AOP-32, Attachment B, Steps 21, 24, 28, & 31

24. MCC-3C2

AC-13A	Canal Drain Pump
CH-12	Boric Acid Batching Tank Heaters
CH-4A	Boric Acid Pump
DW-41A	Primary Water Booster Pump

28. MCC-4A2

Small Roll-up Door	
AI-102/103	Gas Stripper And Waste Evaporation Panel
CH-4B	Boric Acid Pump
DW-41B	Primary Water Booster Pump

21. MCC-3B2

AC-9A	Bearing Water Pump
CF-4	Amine Feed Pump
CF-7A	Secondary Boric Acid Pump
CF-5	Hydrazine Feed Pump

31. MCC-4B2

CF-6	Amine Or Hydrazine Standby Feed Pump
FP-5	Jockey Fire Pump
FW-30B	FW-4B Lube Oil Pump
HCV-1150B	FW-4B Discharge Valve

EO *1.3 **(Slide #211)**

Major Component Description

Boric Acid Pumps (CH-4A/B)

The boric acid pumps are powered from 480 VAC MCCs located in Corridor 26 of the Auxiliary Building.

CH-4A is powered from MCC-3C2 and CH-4B is powered from MCC-4A2.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/25/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	005 A4.02	
Level of Difficulty: 3	Importance Rating	3.4	

Residual Heat Removal System: Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control.

Question: 4

Which of the following alignments results in FCV-326, Shutdown Cooling (SDC) Heat Exchanger Bypass Control Valve, being in its initial SDC warm-up position of 20% open?

Place FCV-326 Override Switch in _____(1)_____ and the FCV-326 Flow Controller set at _____(2)_____.

A. (1) OPEN
(2) 20%

B. (1) MAN
(2) 20%

C. (1) MAN
(2) 80%

D. (1) OPEN
(2) 80%

Answer: C

K/A Match:
Applicant must be familiar with controls associated with FCV-326, Shutdown Cooling Heat Exchanger Bypass Flow Control Valve. This includes the Override Switch and Flow Controller and how their position affects FCV-326 position.

Explanation:

A. Incorrect. Plausible if thought that 20% open on the controller meant that the valve was also 20% open and that OPEN on the override switch allowed the valve to be open. When the Override Switch is taken to OPEN, the valve goes full open.
B. Incorrect. Plausible because the MAN position is correct but in this condition the Bypass Flow Control Valve would be 80% open.
C. **Correct**. Placing the Override Switch in MAN makes the Flow Controller operable. Positioning the Flow Controller at 80% allows the valve to be 20% open. See attached picture.
D. Incorrect. Plausible because the Flow Controller position is correct but the Override Switch must be in MAN.

Technical Reference: LP 7-11-22, Slides #107, #263, & #265, Rev. 3
(Attach if not previously provided including revision number) OI-SC-1, Attachment 1, Step 24, Rev. 67

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-22, SI/CS/Shutdown Cooling System-Licensed Operator
EO 1.3 - **EXPLAIN** the indications located in the Control Room associated with ECCS.
EO 1.4 - **EXPLAIN** the operation of controls located in the Control Room associated with ECCS.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 3
55.43

EO *1.2b **(Slide #263)**

Major Component Description

Shutdown Heat Exchanger Bypass Control (FCV-326)

Used to control total shutdown cooling flow.

FCV-326 is a variable position ball valve controlled from position controller FIC-326 located on CB-2.

EO *1.3, *1.4 **(Slide #265)**

Major Component Description

Shutdown Heat Exchanger Bypass Control (FCV-326)

Two-position keyswitch (MAN/OPEN) labeled LPSI/SHTDN CLG FLOW CNTRLR FCV-326 OVERRIDE SWITCH is located on panel CB-2.

The keyswitch must be in the MAN position for the position controller to be operable.

Green/closed and red/open valve position indicating lights are provided above the keyswitch.

OI-SC-1, Attachment 1, Step 24

NOTE

Heatup of the Shutdown Cooling System is necessary to reduce thermal shock to the system when hot Reactor Coolant flow is established.

24. IF initiating SDC with fuel in the vessel,
THEN warm up the SDC piping by performing the following:

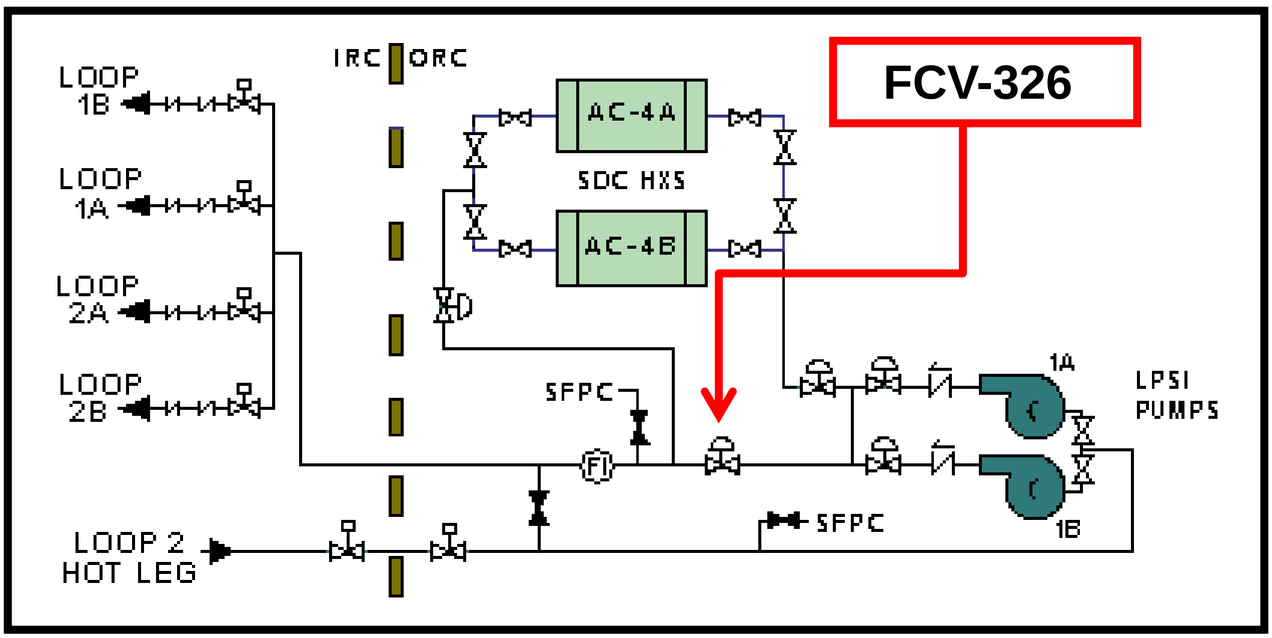
24.1 Place HCV-341, Shtdn HT Exch Valve Flow Cntrlr Ovrd SW, Key Switch in MAN.

24.2 Throttle HCV-341 10% open.

24.3 Place HC-326, LPSI/Shtdn Clg Flow Cntrlr FCV-326 Override Switch in MAN.

24.4 Throttle FCV-326, Shutdown Clg HT Exchs AC-4A & 4B LPSI Bypass Flow Control Valve, 20% open (Controller output at 80%).

(Slide # 107)




LPSI/SHTDN CLG
FLOW CNTRLR FCV-326
OVERRIDE SWITCH 2
MAN
OPEN


LPSI/SHTDN CLG
FLOW CONTROLLER
FCV-326 12
300
250
200
150
100
0
GPM X 10
X100
16
14
X100
12
O
←CNTRLR OUTPUT→
C
0 20 40 60 80 100
C
M
A

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/25/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	005 K3.07	
Level of Difficulty: 3	Importance Rating	3.2	

Residual Heat Removal System: Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: Refueling operations

Question: 5

Given the following conditions:

- The Reactor Vessel Head and Upper Guide Structure are removed.
- Core offload is in progress.
- One train of Shutdown Cooling (SDC) is OPERABLE and in service.
- The other train of SDC is AVAILABLE.
- Refueling Cavity level is 1037 ft.

Which of the following requirements extends the time to boil in the Refueling Cavity in case the running SDC Pump trips?

A. Both trains of SDC are OPERABLE during refueling.

B. Maintain greater than 23 feet of water above the Reactor Vessel Flange.

C. Raise Steam Generator levels to wet layup conditions after Shutdown Cooling is initiated.

D. Starting the second train of Spent Fuel Pool Cooling prior to the start of Core offload.

Answer: B

K/A Match:

Applicant must know the reason for maintaining 23 feet of water above the Reactor Vessel Flange for a loss or malfunction of the Shutdown Cooling System.

Explanation:

A. Incorrect. Plausible because both trains of SDC are required to be operable if refueling cavity level is less than 23 feet.

B. **Correct**. Refueling cavity water level > 23 feet will extend the time to boil when Shutdown Cooling (SDC) is lost. With > 23 feet, only one SDC loop is required for decay heat removal.

C. Incorrect. Plausible because Steam Generator (SG) levels are raised to wet layup conditions during the approach to Shutdown Cooling initiation. Raising these levels before SDC is initiated is what provides the final cooldown needed to reach SDC entry conditions. This is incorrect because waiting to raise SG levels until after SDC is in service will not improve RCS cooldown conditions, since SDC is now the sink for heat transfer instead of the SGs. Sufficient flow does not exist through the SGs in order to remove heat once SDC is in service.

D. Incorrect. Plausible because this will lower Spent Fuel Pool temperature, incorrect because it will not prevent or extend the time to core boiling if fuel is in the vessel, in a loss of Shutdown Cooling.

Technical Reference: (Attach if not previously provided including revision number)
OP-12, Attachment 1, Prerequisite 7, Rev. 70
Technical Specification LCO 2.8.2(2), Amendment #281
SO-O-21, Shutdown Operations Protection Plan

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:
Lesson Plan 4-4-10, Fuel Handling-Auxiliary Operator Nuclear
EO 3.1 - **DESCRIBE** the prerequisites and precautions followed before operating fuel handling equipment.

Lesson Plan 7-11-13, Fuel Handling-Licensed Operator
EO 3.1 - **LIST** the parameters monitored in the Control Room during refueling and explain why each is monitored.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

From OP-12, Attachment 1, Prerequisite 7

7. The Reactor Cavity, Transfer Canal and Spent Fuel Pool have been filled to a level at least 23 feet above the top of the Reactor Vessel Flange.
   - Normal filling is to Elevation 1037’6" (7" below the lighting fixture support bracket).
   - Normal Refueling level is 1036’0" (minimum) to 1037’6".
   - For in-mast sipping the Reactor Cavity shall be filled to at least 1037'3".

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.2 Refueling Operations - Containment

2.8.2(2) Refueling Water Level

Applicability

Applies to the refueling water level during CORE ALTERATIONS, and during REFUELING OPERATIONS inside of containment

Objective

To minimize the consequences of a fuel handling accident during CORE ALTERATIONS and REFUELING OPERATIONS inside of the containment that could affect public health and safety.

Specification

The refueling water level shall be ≥ 23 ft. above the top of the reactor vessel flange.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/25/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	006 K1.02	
Level of Difficulty: 3	Importance Rating	4.3	

Emergency Core Cooling System: Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: ESFAS.

Question: 6

Given the following conditions:

- A Loss of Coolant Accident occurred during a plant cooldown.
- Reactor Coolant System pressure is 1550 psia.

What would be the consequences of initiating Engineered Safeguards Features using the "EMERGENCY OPERATE THINK SE-A & SE-B" switches to mitigate this accident? (Assume no other Operator action is taken).

A. The Diesel Generators would NOT get a start signal.

B. A Recirculation Actuation Signal would NOT be generated.

C. The Sequencers would NOT start the HPSI and LPSI Pumps.

D. A Ventilation Isolation Actuation Signal would NOT be generated.

Answer: B

K/A Match:
Applicant must have knowledge of the logic diagram and signals required for initiating Engineered Safeguards Features equipment and controls.

Explanation:

A. Incorrect. Plausible because this signal is normally generated from a Pressurizer Pressure Low Signal or Containment Pressure High Signal. Incorrect because the THINK switch will generate an independent signal to start the Diesel Generators.

B. **Correct**. A Pressurizer Pressure Low Signal or Containment Pressure High Signal in concert with a Safety Injection Refueling Water Tank Low Signal generates a signal to create a Recirculation Actuation Signal. This signal is blocked by the THINK switches.

C. Incorrect. Plausible because this signal is normally generated from a Pressurizer Pressure Low Signal or Containment Pressure High Signal. Incorrect because the THINK switches will generate an independent signal to fire the Sequencers.

D. Incorrect. Plausible because this signal is not directly generated by the THINK switches. Incorrect because the THINK switches will generate a Safety Injection Actuation Signal which in turn feeds the logic for a Ventilation Isolation Actuation Signal.

Technical Reference: LP 7-12-14, Slide # 51, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Controls System-LO EO 2.2 - **EXPLAIN** the operation/function of ESC switches and controls located in the Control Room.

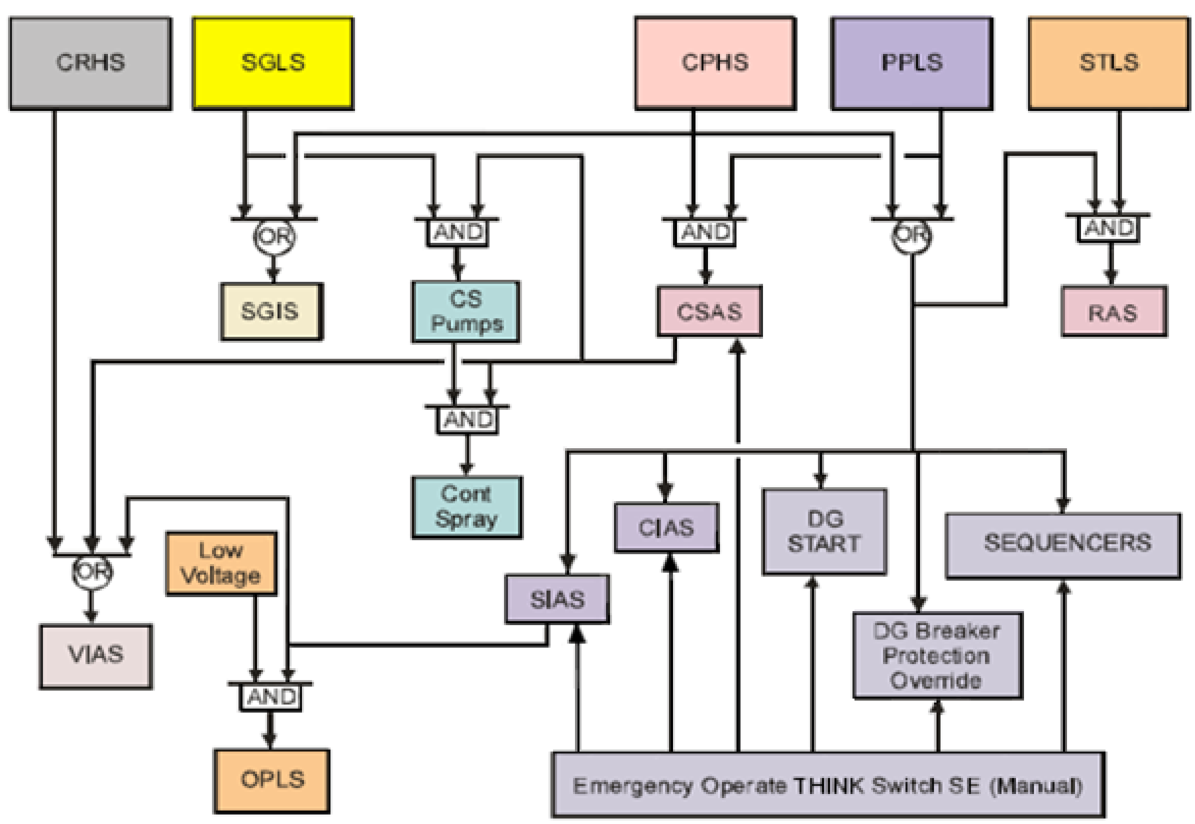
Question Source:
- Bank # X
- Modified Bank # (Note changes or attach parent)
- New

Question History: Last NRC Exam 2004 NRC Exam

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

10 CFR Part 55 Content:
- 55.41 7
- 55.43

(Slide #51)



EO 2.2 **(Slide #51)**

Engineered Safeguards Control Panels AI-30A and AI-30B

Emergency Operate THINK Switches (SE-A & SE-B)

Each emergency operate THINK switch trips prime and backup actuation relays (listed below) directly without tripping the initiation relays:

(1) SIAS

(2) CSAS

(3) CIAS

(4) Load Sequencers

(5) Diesel Start

(6) Diesel Breaker Protection Override

The switch on AI-30A actuates only the A system of safeguards while the switch on AI-30B actuates only the B system of safeguards.

Other actions will still occur as a result of these actuations but are not directly affected by the SE switches (i.e., VIAS).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/22/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	006 K3.03	
Level of Difficulty: 3	Importance Rating	4.2	

Emergency Core Cooling System: Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Containment.

Question: 7

Given the following conditions:

- A design basis Loss of Coolant Accident has occurred.
- VA-3A and VA-3B, Containment Cooling and Filtering Fans, tripped on overcurrent when the Containment Pressure High Signal actuated.

Which of the following is an effect of the loss of Containment Cooling Fans?

A. Indicated Containment pressure greater than actual pressure.

B. Steam Generator indicated levels lower than actual levels.

C. Containment Cooling Fan Heat Exchangers become vapor bound.

D. Containment design pressure of 60 psig will be exceeded.

Answer: D

K/A Match:
Applicant must know the effect of a loss of ECCS Containment Cooling on Containment.
Explanation:

A. Incorrect. Plausible because rising Containment pressure could impact indicated Containment pressure. Incorrect because the FCS Containment pressure detectors do not have an open reference leg.
B. Incorrect. Plausible because a loss of Containment Cooling Fans would disrupt the airflow that is channeled to Containment instrumentation. Incorrect because SG indicated levels would be higher than actual level due to reference leg heating.
C. Incorrect. Plausible if thought that CCW flow was disrupted when the fans trip.
D. **Correct**. Overcurrent trip signal to VA-3A and VA-3B must be specified during a LOCA because the undervoltage trip is overridden during an ESF Actuation. Containment Spray does not actuate at Fort Calhoun Station on a high Containment pressure signal unless it is accompanied by a Steam Generator Low Pressure Signal (SGLS). Containment Pressure High Signal together with a Pressurizer Pressure Low Signal will generate a Containment Spray Actuation Signal (CSAS). The CSAS then sends a permissive signal to the Containment Spray Pumps and Containment Spray Isolation Valves. If a SGLS is present then Containment Spray is actuated.

Technical Reference: LP 7-14-2, Slides #6, #7 & #18, Rev. 1
(Attach if not previously provided including revision number) LP 7-12-14, Slide # 51, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-22, SI, CS, & SDC Systems-Licensed Operator
EO 1.0 - Given specific plant conditions, **APPLY** the principles of operation of the Containment Air Cooling and Filtering System to diagnose system response.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	9
	55.43	

TO 1.0 **(Slide #6)**

System Purposes

(LC) Maintains a continuous flow or recirculating air throughout the containment to prevent the accumulation of hydrogen pockets.

Removes heat from the containment atmosphere during a Design Basis Accident (DBA) to the extent necessary to maintain the structure below design pressure (60 psig).

Reduces the fission product inventory in the containment atmosphere by filtration following a DBA. (This is not credited in the Radiological Consequences analysis)

Reduces the temperature and pressure during the first few seconds of a Main Steam Line Break (MSLB) inside Containment, as credited in Section 14.16 of the USAR, during the time it takes HCV-1385 and HCV-1386 to close upon generation of SGIS. (The safety analysis assumes the valves will close in 40 seconds.)

NOTE: The DBA is a double ended rupture of the largest reactor coolant pipe coincident with a loss of normal and offsite electrical power.

EO 1.4 **(Slide #18)**

Major Component Description

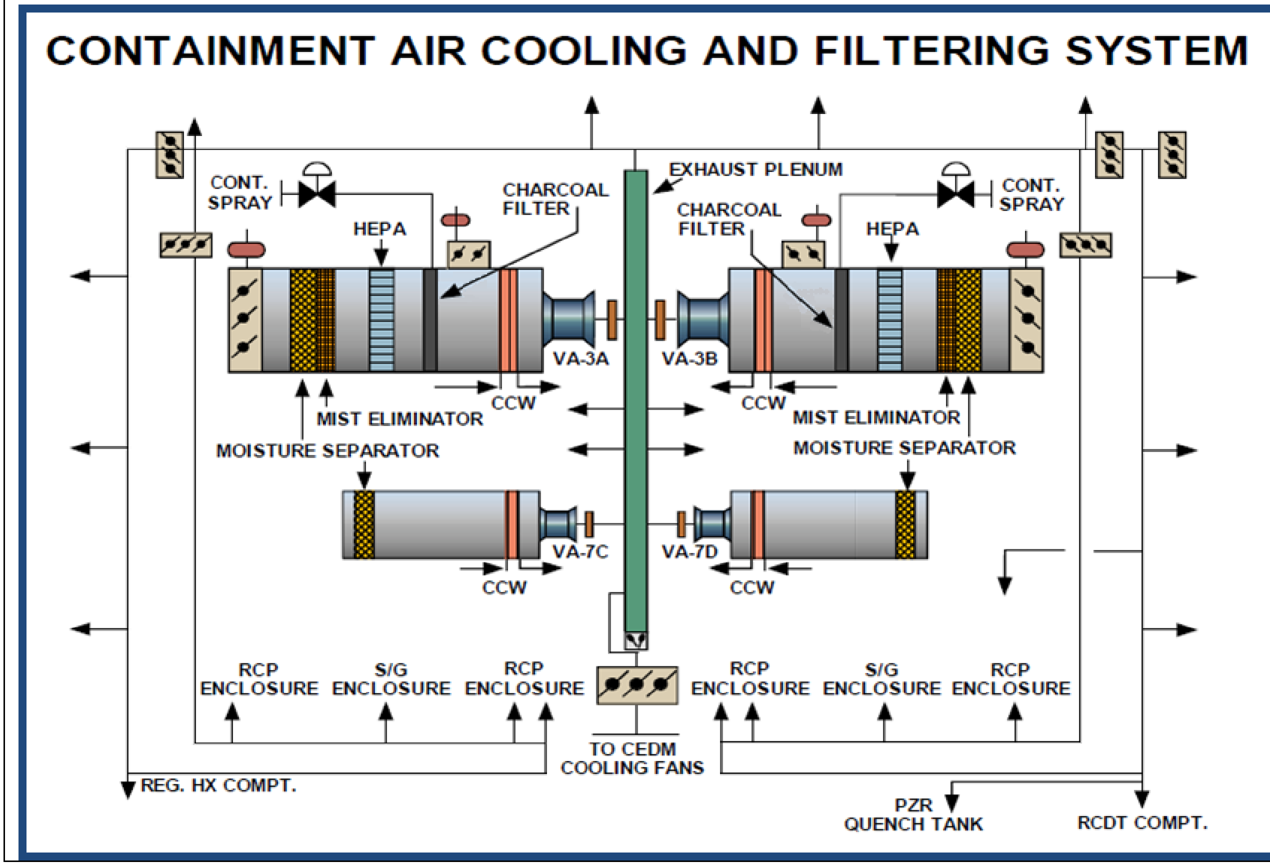
Cooling and Filtering Unit Fans (VA-3A/B)

Both fans are automatically started by engineered safeguards sequencers upon receipt of PPLS or CPHS.

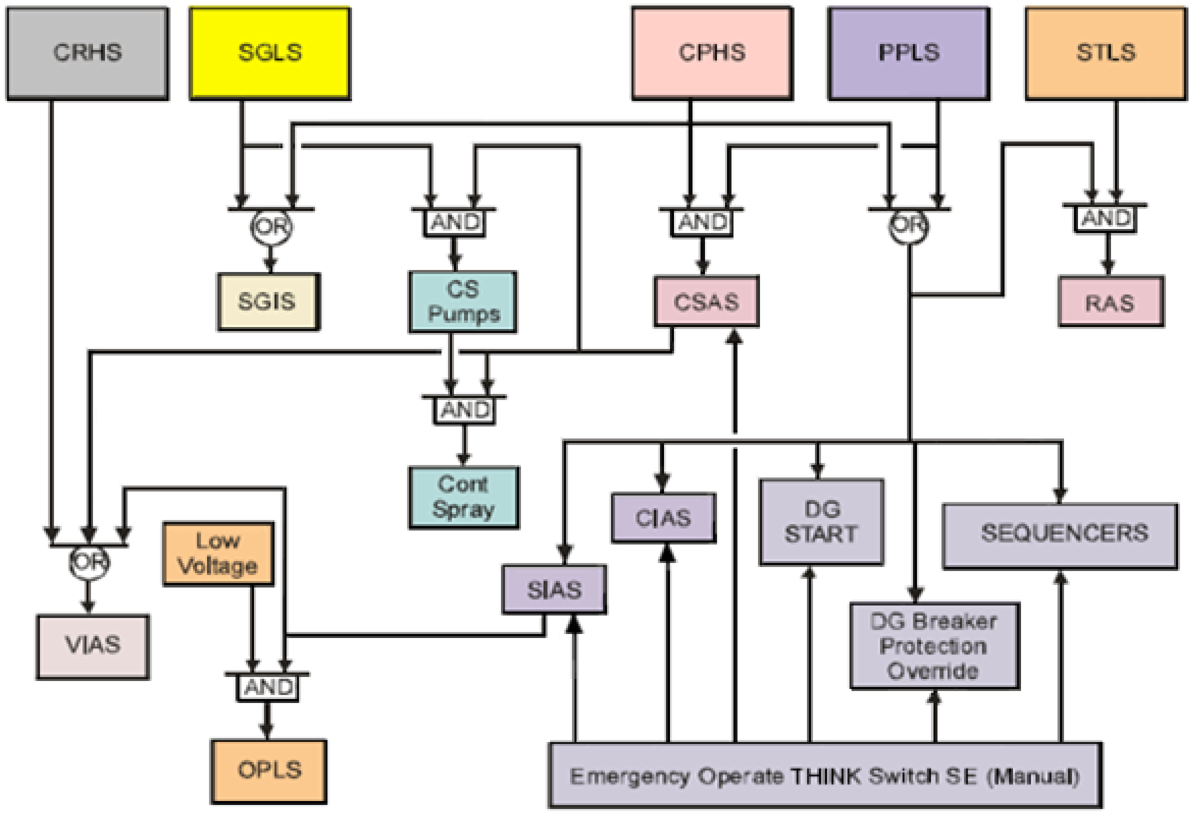
The undervoltage trip of VA-3A/B is overridden during an ESF actuation.

Therefore, the fan would not trip off due to the actuation of the undervoltage relay during a LOCA.

(Slide #7)


CONTAINMENT AIR COOLING AND FILTERING SYSTEM
CONT. SPRAY
HEPA
CHARCOAL FILTER
EXHAUST PLENUM
CHARCOAL FILTER
HEPA
CONT. SPRAY
VA-3A
VA-3B
CCW
CCW
MIST ELIMINATOR
MIST ELIMINATOR
MOISTURE SEPARATOR
MOISTURE SEPARATOR
VA-7C
VA-7D
CCW
CCW
RCP ENCLOSURE
S/G ENCLOSURE
RCP ENCLOSURE
RCP ENCLOSURE
S/G ENCLOSURE
RCP ENCLOSURE
TO CEDM COOLING FANS
REG. HX COMPT.
PZR QUENCH TANK
RCDT COMPT.

(Slide #51)


CRHS
SGLS
CPHS
PPLS
STLS
OR
AND
AND
OR
AND
SGIS
CS Pumps
CSAS
RAS
AND
Cont Spray
CIAS
DG START
SEQUENCERS
OR
Low Voltage
SIAS
VIAS
DG Breaker Protection Override
AND
OPLS
Emergency Operate THINK Switch SE (Manual)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	007 A2.02	
Level of Difficulty: 3	Importance Rating	2.6	

Pressure Relief/Quench Tank System: Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT.

Question: 8

Given the following conditions:

- Plant is at 90% power.
- PCV-102-1, Power Operated Relief Valve (PORV), spuriously opened, then immediately closed, but now is leaking by.
- Quench Tank level is 70% and slowly rising.
- Quench Tank pressure is 8 psig and slowly rising.
- Quench Tank temperature is 122°F and slowly rising.

Which of the following results from the spurious PORV Actuation, and what restores normal Quench Tank temperature per OI-RC-6, Pressurizer Quench Tank Normal Operation?

The MINIMUM Quench Tank pressure where an automatic action will occur to lower Quench Tank pressure is ____(1)____ .
Restore Quench Tank temperature by draining to the ____(2)_____.

A. (1) 70 psig.
   (2) Reactor Coolant Drain Tank and refilling with Deaerated Water

B. (1) 75 psig.
   (2) Containment Sump and refilling with Potable Water

C. (1) 70 psig.
   (2) Containment Sump and refilling with Potable Water

D. (1) 75 psig.
   (2) Reactor Coolant Drain Tank and refilling with Deaerated Water

Answer: A

K/A Match:

Applicant must know response to rising pressure and alignment necessary to cool the Quench Tank.

Explanation:

A. **Correct**. The Quench Tank Relief Valve will eventually lift at 70 psig as the Vent Valve is manually operated. Cooling of the Quench Tank is performed by draining and refilling via the RCDT and Deaerated Water.

B. Incorrect. Plausible because at 75 psig, the Quench Tank rupture disk will relieve pressure to containment. Incorrect because the Quench Tank Relief Valve will lift first. Plausible because the cooling mechanism for the Quench Tank in accordance with OI-RC-6 is via the RCDT and Deaerated Water.

C. Incorrect. Plausible because the Quench Tank Relief Valve will lift at 70 psig. Incorrect because temperature control for the Quench Tank is via the RCDT and Deaerated Water.

D. Incorrect. Plausible because the mechanism to cool the Quench Tank is correct. Incorrect because the Quench Tank Relief Valve will lift before the Quench Tank Rupture disk relieves pressure.

Technical Reference: OI-RC-6, Precautions 1, 2, and 4, & Attachment 5, Rev. 13

(Attach if not previously provided including revision number) LP 7-11-20, Slide #247 & #248, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-20, Reactor Coolant System-Licensed Operator

EO 3.4 - **LIST** the major steps for proper operation of the quench tank per OI-RC-6.

EO 3.4a - **DISCUSS** the prerequisites and precautions for operating the quench tank.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	
55.41	3
55.43	

OI-RC-6, Pressurizer Quench Tank Normal Operation

PRECAUTIONS

1. Pressurizer Quench Tank, RC-5, pressure shall not exceed 10 psig during normal Reactor operation.

2. Pressurizer Quench Tank temperature shall not exceed 120°F during normal Reactor operation.

3. Pressurizer Quench Tank level shall be maintained at 67% to 79%.

4. The Pressurizer Quench Tank Safety Relief Valve, RC-125, will lift at 70 psig and the Quench Tank Rupture disc will relieve at 75 psig.

EO 1.9 **(Slide #247)**

Detailed Component Description

Quench Tank (RC-5)

The QT is equipped with its own pressure relief valve (RC-125) and a rupture disc.

The relief valve (set at 70 psig) relieves to the Waste Disposal System via a floor drain.

The rupture disc (set at 75 psig) relieves pressure to the containment atmosphere.

The rupture disc is designed to handle the discharge capacity of all four pressurizer reliefs.

EO 1.9, *4.4 **(Slide #248)**

Detailed Component Description

Quench Tank (RC-5)

Gases from the top of the tank are processed by the Waste Gas Disposal System (containment vent header) through HCV-155.

HCV-155 three-position control switch (CLOSE/NOR/OPEN) is located on CB-3.

NOTE: MR-FC-92-008 removed the internals from check valve WD-817 resulting in the RCDT pressure to “float” on vent header pressure.

Notes/Cautions were added to OI-RC-6 because vent header pressure may be affected by pressurizing and purging the quench tank.

Attachment 5 - Quench Tank Temperature Control following Safety or Relief Valve Discharge

PROCEDURE

NOTE
Quench Tank temperature may also be read on ERF Computer Point T133.

1. Open the following valves (CB-11):

   - HCV-1560A, Deaerated Water Header Isolation Valve
   - HCV-1560B, Deaerated Water Header Isolation Valve

2. Open HCV-153, Quench Tank Drain Valve, as necessary to maintain level at 67% to 79% (normally 73%) on LIA-132, Pressurizer Quench Tank level (CB-1/2/3).

NOTE
Operation of WD-2A, RCDT WD-1 Outlet Pump, in AUTO mode should satisfy the following step.

3. Pump WD-1, Reactor Coolant Drain Tank, as necessary, to maintain the following (AI-100):

   - LIC-501, RC Drain Tank Level Indicator, level less than 39 inches
   - PIC-503, Indicating Pressure Controller, less than 2 psig

4. Open and close the following valves as necessary to maintain 67% to 79% level in the Quench Tank:

   - HCV-153
   - HCV-1560A
   - HCV-1560B

5. WHEN Quench Tank temperature is less than 120°F on TIA-133, Pressurizer Quench Tank Temperature (CB-1/2/3),
   THEN close the following valves:

   - HCV-153
   - HCV-1560A
   - HCV-1560B

From OI-RC-6, Attachment 5

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	008 K4.09	
Level of Difficulty: 2	Importance Rating	2.7	

Component Cooling Water System: Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: The "standby" feature for the CCW pumps.

Question: 9

Given the following conditions:

- The plant is operating at 50% power with Component Cooling Water (CCW) Pump AC-3A operating.
- The control switches for CCW Pumps AC-3B and AC-3C are in the AFTER-STOP position.

What is the expected plant response to an overcurrent trip of CCW Pump AC-3A?

A. Only AC-3C immediately starts. AC-3B remains in STANDBY.

B. Both AC-3B and AC-3C immediately start.

C. AC-3B receives a start signal. AC-3C starts 30 seconds later if AC-3B fails to start.

D. AC-3C receives a start signal. AC-3B starts 30 seconds later if AC-3C fails to start.

Answer: B

K/A Match:

Applicant must understand the standby feature of the CCW Pumps and the interlock associated with being in the AFTER-STOP position.

Explanation:

A. Incorrect. Plausible because both 480 V Buses feeding AC-3A (1B3B) and AC-3C (1B3C-4C) are fed from 4160 V Safeguards Bus 1A3. It could be thought that like powered Buses start to maintain equalized transformer loads.
B. **Correct**. Standby CCW Pumps in AFTER-STOP will automatically start when the running pump trips.
C. Incorrect. Plausible if thought that the standby CCW Pump starts were tied to a sequencer, which in this case has not actuated.
D. Incorrect. Plausible if thought that the standby CCW Pump starts were tied to a sequencer.

Technical Reference: LP 7-11-6, Slides #25, #26, & #35, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-6, Component Cooling Water System-Licensed Operator
EO 1.4 - **EXPLAIN** standby operation of the CCW pumps in terms of switch positions and automatic actions.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

EO *1.4, 4.1 **(Slide #25)**

Major Component Description

Component Cooling Water Pumps (AC-3A, B & C)

Pumps are operated from the Control Room panel CB-1,2,3.

Handswitch positions: START/STOP/AUTO/PULL-TO-LOCK.

Normally one pump is running; two pumps are in standby.

EO *1.4, *1.5 **(Slide #26)**

Major Component Description

Component Cooling Water Pumps (AC-3A, B & C)

Non-running pumps are in standby any time:

(1) The circuit breaker is racked in.

(2) The 69-permissive switch is red-flagged.

(3) The control switch is not in PULL-TO-LOCK.

Standby pumps will start if the running pump trips, initiated by circuit breaker trip contacts.

EO 4.2 **(Slide #35)**

Major Component Description

Component Cooling Water Pumps (AC-3A, B & C)

<u>NOTE</u>: Use current revision of ARPs to review operator actions.

The following alarms are found on annunciator panel A2 in the Control Room:

CCW PUMPS AC-3A/B/C STANDBY START

(a) Actuates when any pump starts as a result of a standby start signal

Operator actions are:

(a) Note that both standby pumps will probably be running.

(b) Determine if all running pumps are needed.

(c) Secure unneeded pump(s).

(d) Determine cause of pump trip.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/11/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	010 G 2.2.22	
Level of Difficulty: 3	Importance Rating	4.0	

Pressurizer Pressure Control System: Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Question: 10

(1) Which of the following is the MINIMUM Pressurizer pressure which violates a Safety Limit when in MODE 2, and
(2) What Technical Specification action is required?

A. (1) 2600 psia.
(2) Restore compliance within five minutes.

B. (1) 2800 psia.
(2) Restore compliance within five minutes.

C. (1) 2600 psia.
(2) Restore compliance and be in HOT SHUTDOWN within one hour.

D. (1) 2800 psia.
(2) Restore compliance and be in HOT SHUTDOWN within one hour.

Answer: D

K/A Match:
Applicant must be able to analyze the event in progress and evaluate its impact on Technical Specification Safety Limits and Limiting Conditions for Operation.

A. Incorrect. Plausible because restoring compliance within five minutes is required if the Reactor is in MODES 3, 4, or 5. Incorrect setpoint.
B. Incorrect. Plausible because the Safety Limit is 2750 psia (110% of the 2500 psia design pressure). Incorrect because restoring compliance within five minutes is required if the Reactor is in MODES 3, 4, or 5.
C. Incorrect. Plausible because compliance must be restored and HOT SHUTDOWN achieved within one hour. Incorrect setpoint.
D. **Correct**. 2800 psia exceeds the Safety Limit setpoint of 2750 psia (110% of the 2500 psia design pressure). Compliance must be restored and HOT SHUTDOWN achieved within one hour per Technical Specification 1.1.

Technical Reference: LP 7-11-20, Slide #174, Rev. 1
(Attach if not previously provided including revision number) Technical Specification 1.1, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-08, Technical Specifications-Licensed Operator
EO 3.0 - **STATE** the three (3) safety limits and the basis for each.
EO 4.0 - **STATE** the immediate action for a safety limit violation.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

TECHNICAL SPECIFICATIONS

1.0 SAFETY LIMITS

1.1 Safety Limits (SLs)

1.1.1 Reactor Core SLs

Applicability

This specification applies to the limiting combinations of reactor power and reactor coolant system flow, temperature and pressure during operation.

Objective

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant.

Specifications

(a) The reactor power level shall not exceed the allowable limit for the pressurizer pressure and the cold leg temperatures as shown in Figure 1-1 for 4-pump operation. The safety limit is exceeded if the point defined by the combination of reactor coolant cold leg temperature and power level is at any time above the appropriate pressurizer pressure line.

(b) Peak fuel centerline temperature shall be maintained at < 5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56(P)(A), Revision 1, Supplement 1.

1.1.2 Reactor Coolant System Pressure SL

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity to the containment.

Specification

The reactor coolant system pressure shall not exceed 2750 psia when fuel assemblies are located within the reactor vessel.

TECHNICAL SPECIFICATIONS

1.0 SAFETY LIMITS
1.1 Safety Limits (SLs) (continued)

Flow maldistribution effects for operation under less than full reactor coolant flow have been evaluated via model test.(2) The flow model data established the maldistribution factors and hot channel inlet temperature for the thermal analyses that were used to establish the safe operating envelopes presented in Figure 1-1. The reactor protective system is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the minimum DNBR limit.(1)

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

The reactor coolant system serves as a barrier to prevent radionuclides in the reactor coolant from reaching the containment atmosphere.(3) In the event of a fuel cladding failure, the reactor coolant system is the primary barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of the reactor coolant system and fuel cladding. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME Code, section III, is 110% of design pressure. The maximum transient pressure allowable in the reactor coolant system piping, valves and fittings under USAS section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.

The settings and capacity of the main steam safety valves (1000 - 1050 psia)(5), the reactor high-pressure trip (≤ 2400 psia) and the reactor coolant system safety valves (2500-2545 psia)(6) have been established to assure never reaching the reactor coolant system pressure safety limit. The initial hydrostatic test pressure was conducted at 3125 psia (125% of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the pressurizer power-operated relief valves, consistent with the reactor high pressure trip, and opening the steam system steam dump and bypass valves upon receipt of a turbine trip signal.(7)

TECHNICAL SPECIFICATIONS

1.0 SAFETY LIMITS

1.2 Safety Limit Violations

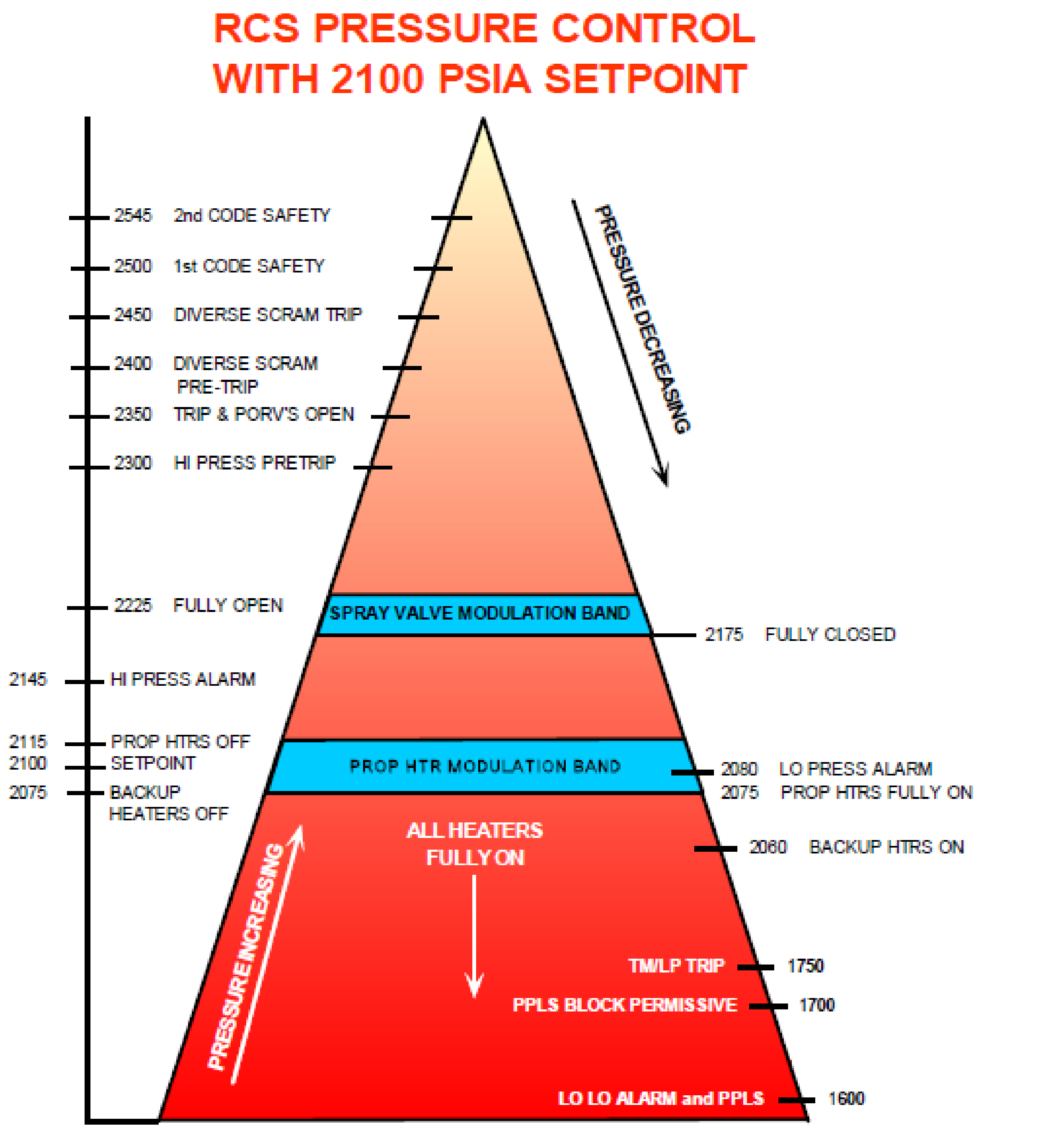
1.2.1 If Safety Limit 1.1.1 is violated, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2 If Safety Limit 1.1.2 is violated:

1.2.2.1 In MODE 1 or 2, restore compliance and be in at least HOT SHUTDOWN within 1 hour.

1.2.2.2 In MODES 3, 4, or 5, restore compliance within 5 minutes.

(Slide #174)


RCS PRESSURE CONTROL
WITH 2100 PSIA SETPOINT
2545 2nd CODE SAFETY
2500 1st CODE SAFETY
2450 DIVERSE SCRAM TRIP
2400 DIVERSE SCRAM PRE-TRIP
2350 TRIP & PORV'S OPEN
2300 HI PRESS PRETRIP
PRESSURE DECREASING
2225 FULLY OPEN
SPRAY VALVE MODULATION BAND
2175 FULLY CLOSED
2145 HI PRESS ALARM
2115 PROP HTRS OFF
2100 SETPOINT
2075 BACKUP HEATERS OFF
PROP HTR MODULATION BAND
2080 LO PRESS ALARM
2075 PROP HTRS FULLY ON
ALL HEATERS FULLY ON
2060 BACKUP HTRS ON
PRESSURE INCREASING
TM/LP TRIP 1750
PPLS BLOCK PERMISSIVE 1700
LO LO ALARM and PPLS 1600

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	012 A1.01	
Level of Difficulty: 3	Importance Rating	2.9	

Reactor Protection System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPS controls including: Trip setpoint adjustment.

Question: 11

Given the following conditions:

- Power ascension is in progress.
- Reactor power has just reached 90%.
- Window C-7 – VARIABLE OVERPOWER RESET DEMAND has just alarmed on CB-4.

Which of the following describes where the Variable Overpower Trip (VOPT) can be reset and what adjustment is made to the VOPT setpoint once reset?

VOPT is reset _____(1)_____ and the new VOPT setpoint becomes _____(2)_____.

A. (1) only at CB-4
(2) 99%

B. (1) only at CB-4
(2) 109%

C. (1) at CB-4 or RPSCIP Panel
(2) 99%

D. (1) at CB-4 or RPSCIP Panel
(2) 109%

Answer: C

K/A Match:
Applicant must predict changes to the VOPT setpoint including where it is performed.
Explanation:
A. Incorrect. Plausible because the VOPT setpoint is correct but this trip can be reset at CB-4 or the RPSCIP Panel.
B. Incorrect. Plausible because the VOPT reset pushbuttons are located on CB-4 to reset VOPT. 109% is the maximum value for VOPT but given the initial power level it can only rise ~10%.
C. **Correct**. The VOPT setpoint calculator generates a trip signal (Qtr) which cannot be more than 10% above existing power level. Depressing either pushbutton increases Qtr to not more than 10% (9% actual) above existing power when depressed.
D. Incorrect. Plausible because the reset locations are correct but given the initial power level the VOPT setpoint can only rise ~10%.

Technical Reference: LP 7-12-25,Slides #89, #90, #91, & #250, Rev. 0
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-25, Reactor Protective System and DSS-Licensed Operator
EO 1.3 - **LIST** the reactor trips and trip setpoints provided by the RPS and **STATE** the source of the signal(s) supplied to each trip.
EO 3.2 - Given a copy of OI-RPS-1 and a drawing of the RPS cabinets, **DISCUSS** the indications provided by the RPS and **STATE** the function of each switch and its associated position.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6
55.43

EO 1.3 **(Slide #89)**

Major Component Description

High Power Level Trip (TU-1)

The VOPT setpoint calculator has fixed minimum and maximum setpoints (19.1% and 109% power respectively by T.S.).

The VOPT setpoint calculator generates a trip signal (Qtr) which cannot be more than 10% above existing power level for power between the minimum and maximum.

EO 1.3 **(Slide #90)**

Major Component Description

High Power Level Trip (TU-1)

The Qtr calculation is only performed when the operator depresses the channel (A, B, C or D) VOPT reset pushbutton on CB-4 or the individual RPSCIP drawer.

(LC) Both pushbuttons illuminate and the VARIABLE OVER POWER RESET DEMAND alarm on CB-4 (A20) annunciates when power is within 3% of Qtr.

EO 1.3 **(Slide #91)**

Major Component Description

High Power Level Trip (TU-1)

Depressing either pushbutton increases Qtr to not more than 10% (9% actual) above existing power when depressed.

Maximum Qtr is 109% (actual 108.6%).

The pretrip setpoint (2% below Qtr) is recalculated with each change in Qtr.

NOTE: The VOPT calculator is enabled at 9.1% power. The first trip setpoint would be 19.1%, pretrip would be 17.1% and the first "Reset Demand" would be at 16.1%.

When power is lowered, the calculator automatically ramps down Qtr such that Qtr is 10% or less above existing power.

Minimum Qtr is 19.1% (actual 18.9%).

EO 3.2 **(Slide #250)**

Normal System Operation

RPSCIP

The variable high power trip reset (VOPT RESET) button and light.

Illuminates when the existing power is within 4% of the variable setpoint.

When the button is depressed, a new trip setpoint is generated which is approximately 10% above the existing power level.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/25/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	012 A2.04	
Level of Difficulty: 4	Importance Rating	3.1	

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: Erratic power supply operation.

Question: 12

Given the following condition:

- High voltage DC has dropped to 600 VDC on Channel B Power Range Safety Nuclear Instrument Drawer.

Which of the following identifies the impact on Channel B Reactor Protection System and the required action?

The Channel B Linear Non-Op light is _____(1)_____.
Place the Trip Units for _____(2)_____ in BYPASS in accordance with AOP-15, Loss of Flux Indication or Flow Streaming.

A. (1) lit
(2) High Power, High Startup Rate, & Axial Power Distribution

B. (1) NOT lit
(2) High Power, Thermal Margin Low Pressure, & Axial Power Distribution

C. (1) NOT lit
(2) High Power, High Startup Rate, & Axial Power Distribution

D. (1) lit
(2) High Power, Thermal Margin Low Pressure, & Axial Power Distribution

Answer: D

K/A Match:

Normal power supply to Linear NI Channel is between 700 and 800 VDC. Voltage < 650 VDC implies erratic power supply operation and the applicant must know the impact of this loss of voltage. The mitigative actions are to bypass the selected Trip Units.

Explanation:

A. Incorrect. Plausible because the PTTI relay is de-energized and 2 of 3 Trip Units are correct. High startup rate is associated with the Wide Range Nuclear Instrumentation.

B. Incorrect. Plausible because these Trip Units need to be bypassed. When voltage is less than 650 VDC the PPTI relay is de-energized and the non-op light is lit.

C. Incorrect. Plausible because these Trip Units all use an NI signal, however, the High Startup Rate is from the Wide Range NI.

D. **Correct**. The Power Trip/Test Interlock is used to ensure that valid Reactor power signals are provided to those Trip Units using Linear NI Power inputs. Trip Units 1, 9, and 12 are the affected channels. The PPTI relay is also deenergized when the RPSCIP TM/LP calculator mode switch is not in OPERATE and a Linear NI channel is in the TEST mode, which generates a non-op light on the drawer.

Technical Reference: LP 7-12-25,Slide #63 & #64, Rev. 0
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-25, Reactor Protection System & DSS-Licensed Operator
EO 1.3 - **LIST** the reactor trips and trip set points provided by the RPS and **STATE** the source of the signal(s) supplied to each trip.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6
55.43 ______

EO 1.3 **(Slide #63)**

General System Description

RPS testing

A power trip/test interlock (PTTI) is provided to ensure the high power trip unit, the APD trip unit, the TM/LP trip unit and the SCEAPIS are receiving valid reactor power input signals.

PTTI is activated (relay de-energized) on any of the following:

(1) Either the HV or LV power supplies in the linear NI drawer is outside of designated voltages. NOTE: Normal power source to channel detectors is 700-800 VDC. Loss of high voltage occurs at <650 VDC.

(2) Linear NI channel in TEST mode (the Test Enable Switch out of OPERATE and the Test Select switch out OFF).

(3) RPSCIP TM/LP calculator mode switch not in OPERATE.

(4) Loss of 24 VDC to relay K-31.

EO 1.3 **(Slide #64)**

General System Description

RPS testing

Activating PTTI results in the following:

(1) Trips the High Power trip unit (T.U. #1).

(2) Trips the TM/LP trip unit (T.U. #9).

(3) Trips the APD trip unit T.U. #12).

(4) Removes Q power input signal to SCEAPIS maximum select circuit.

Bank Question:
Power Trip/Test Interlock [PTTI] ensures that certain RPS trip units are receiving valid inputs. If PTTI is initiated on "A" channel of the RPS, which RPS trip units would trip?

A. High Power [T.U.#1], TM/LP [T.U.#9], and APD [T.U.#12]

B. High Start-up Rate [T.U.#2], ASGT [T.U.#7], and APD [T.U.#12]

C. Low Flow [T.U.#3], Low S/G Pressure [T.U.#6], and TM/LP [T.U.#9]

D. High Power [T.U.#1], High Start-up Rate [T.U.#2], and APD [T.U.#12]

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	013 A4.02	
Level of Difficulty: 3	Importance Rating	4.3	

Engineered Safety Features Actuation System: Ability to manually operate and/or monitor in the control room: Reset of ESFAS channels.

Question: 13

Given the following conditions:

- A Loss of Offsite Power occurred during a Loss of Coolant Accident.
- Offsite Power was subsequently restored.

Which of the following describes reset of Offsite Power Low Signal (OPLS) 86A/OPLS?

Place CHAN "A" TEST AND BYPASS SW TS-A/OPLS in _____(1)_____.
Reset the tripped 86A/OPLS relay.
Observe the amber Supervisory lights _____(2)_____.

A. (1) TEST
(2) ON

B. (1) BYPASS
(2) ON

C. (1) BYPASS
(2) OFF

D. (1) TEST
(2) OFF

Answer: C

K/A Match:

Applicant must know the purpose of the Test and Bypass Switch as well as the indications of a RESET ESFAS relay.

Explanation:

A. Incorrect. Plausible because the amber RESET light will be lit if OPLS is not tripped, however, the Test and Bypass Switch must be in the BYPASS position, and when the switch is in bypass, the amber supervisory lights will not be lit.

B. Incorrect. The ESFAS channel is reset by placing the Test and Bypass Switch in the BYPASS position. This would be correct if the supervisory lights were dark. This is plausible because the lights are not lit even when the bus is energized when the test switch is in Bypass.

C. **Correct**. The Test and Bypass Switch must be in BYPASS position to reset OPLS. When the test switch is in bypass, the supervisory and relay reset lights are not lit.

D. Incorrect. Selecting this answer implies a misunderstanding of the purpose of the Test and Bypass Switch. While in the TEST position, it keeps the matrix relays energized which could be misconstrued as allowing the relay to be RESET.

Technical Reference: (Attach if not previously provided including revision number)
AOP-23, Section V, Steps 9 & 10, Rev. 11a
AOP-23, Section V, Step 2.0.B.2), Rev. 11a
LP 7-12-14,Slide #35, #255, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, ESFAS-Licensed Operator
EO 1.5 - **EXPLAIN** the functions performed by each Engineered Safeguards Control signal.
EO 1.8 - **EXPLAIN** the features that allow testing of ESC signals.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

AOP-23, Section V

9. **IF** 86A/OPLS has tripped,
THEN place "CHAN "A" TEST AND BYPASS SW TS-A/OPLS" in "BYPASS".

9.1 **IF** 86B/OPLS has tripped,
THEN place "CHAN "B" TEST AND BYPASS SW TS-B/OPLS" in "BYPASS".

10. Reset the OPLS relay by performing step a or b:

a. Reset the tripped OPLS relay.

b. Perform Floating Step II, Disabling Safeguards Relays.

10.1 **IF** the OPLS relay will not reset, **THEN** restore the ability to load Vital 4160 Buses by performing step a or b:

a. Initiate PPLS using the PPLS test switches.

b. Initiate CPHS using the CPHS test switches.

Section V - Reset of Offsite Power Low Signal (OPLS)

1.0 PURPOSE

This procedure section provides guidance for restoration of the plant following an inadvertent actuation of Offsite Power Low Signal (OPLS).

2.0 ENTRY CONDITIONS

A. Plant conditions and other evidence indicate that OPLS has inadvertently actuated.

B. One or more of the following indications may be present:

1) ERF Safety Actuation Matrix Display Alarms.

2) OPLS Relay tripped and Amber Light off.

3) ERF Printouts showing OPLS Relay tripped.

EO *1.5 **(Slide #35)**

Actuation Signals

ESC is comprised of the following actuation signals:

(1) Diesel Generator (DG) Start – starts diesel generators D-1 and D-2.

(2) DG Breaker Protection Override – overrides protective trips for the DG circuit breaker.

(3) Sequencers – sequentially start safeguards pumps and fans.

(4) Safety Injection Actuation Signal (SIAS) – provides emergency core cooling and emergency boration.

(5) Containment Isolation Actuation Signal (CIAS) – isolates unnecessary flowpaths to and from the containment and provides CCW to the containment cooling units to minimize radiological release.

(6) Containment Spray Actuation Signal (CSAS) coincident with a Steam Generator Low Signal (SGLS) – initiates containment spray to reduce containment pressure.

(7) Ventilation Isolation Actuation Signal (VIAS) – isolates containment vent paths.

(8) Recirculation Actuation Signal (RAS) – initiates recirculation mode for long term core cooling.

(9) Steam Generator Isolation Signal (SGIS) – isolates S/G steam and feed flow to terminate main steam leak events.

(10) Offsite Power Low Signal (OPLS) – ensures a reliable source of adequate voltage is provided for safeguards equipment.

EO 1.8 **(Slide #255)**

Offsite Power Low Signal (OPLS)

OPLS Sensor Channels

Pushbutton test switch (TS/OPLS-A(B)), located on the bus/transformer potential compartment, can be used to simulate SIAS actuation for the sensor circuit under test.

While in the TEST position, it will also keep the matrix relays energized, thereby allowing the voltage sensing device and the time delay relays to be tested.

In this configuration, OPLS would be in a two-of-three logic.

When used in conjunction with the undervoltage test switch discussed earlier, it allows testing of the TDDO relay (27-T1/OPLS-A(B/C/D).

The amber light on the potential compartment with the test switches will go OUT when the relay drops out, just like a regular trip.

The difference in this case is that the matrix relays remain energized.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	013 K6.01	
Level of Difficulty: 3	Importance Rating	2.7	

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

Question: 14

Given the following conditions on Pressurizer Pressure Low Signal (PPLS) Matrix lights:

- 2 of 4 PPLS Matrix A and B Supervision amber lights are extinguished on AI-30A/B-ESF.
- The other 2 PPLS Matrix A and B Supervision amber lights are brighter than normal.

Which of the following has occurred to cause this condition?

A. 3 of 4 Channels of PPLS in trip.

B. A/PIA-102Y, Pressurizer Pressure Safety Channel failed low.

C. A Loss of Instrument Bus AI-40A occurred.

D. A/PIA-102X, Pressurizer Pressure Safety Channel Setpoint failed low.

Answer: B

K/A Match:
Applicant must understand the purpose of ESFAS Supervisory light indication because they monitor the sensor inputs used to trip ESFAS relays (PPLS/SIAS/CRHS, etc.).

Explanation:

A. Incorrect. In this condition none of the lights would be lit.

B. **Correct**. This is the supervisory panel indication when one sensor circuit goes to the tripped condition. If 2 sensors go to the tripped condition, all 4 lights will extinguish.

C. Incorrect. Only one light will be deenergized for this condition consistent with Instrument Bus power supplies AI-40A/B/C/D feeding each of the 4 supervisory lights.

D. Incorrect. Plausible because this signal is calculated from PIA-102Y for the TM/LP setpoint and is adjacent on the control boards to PIA-102X; incorrect because this instrument provides input to the TM/LP trip and does not input PPLS.

Technical Reference: LP 7-12-14,Slide #26, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Control System-Licensed Operator
EO 2.1 - Given the control boards or simulator, **EXPLAIN** the Control Room indications associated with ESC.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

EO 2.1 **(Slide #26)**

Supervisory Systems

Matrix supervisory lights are provided for each two-of-four coincidence matrix.

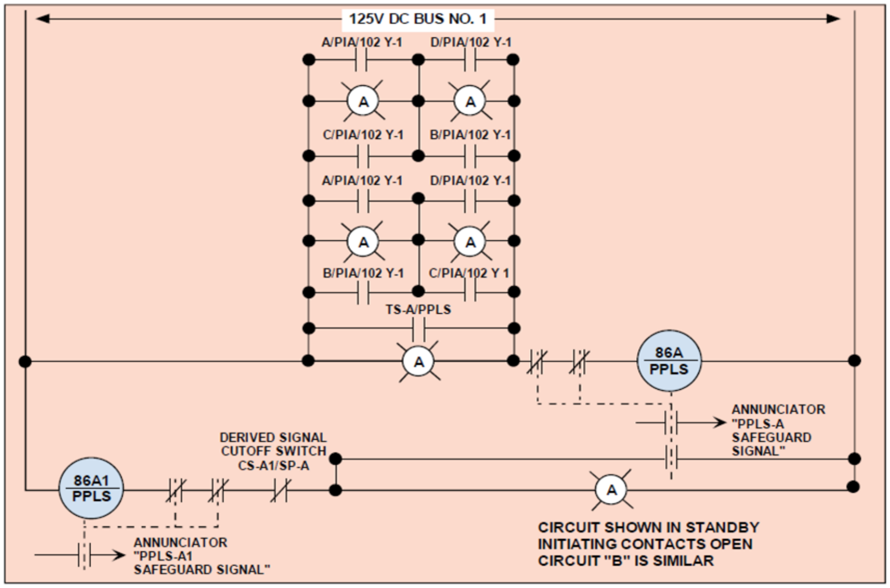
Four amber lights, connected in parallel with the sensor contacts, are normally lit.

Matrix supervisory lights can detect a loss of power, sensor contact closure, grounds, shorts or open circuits.

If one sensor circuit goes to the tripped condition, two lights will extinguish and two lights will get brighter.

If two sensor circuits go to the tripped condition, the lockout relay trips, or power is lost, all four lights will extinguish.

(Slide #26)


PPLS 2/4 MATRIX AND SUPERVISION
125V DC BUS NO. 1
A/PIA/102 Y-1
D/PIA/102 Y-1
C/PIA/102 Y-1
B/PIA/102 Y-1
A/PIA/102 Y-1
D/PIA/102 Y-1
B/PIA/102 Y-1
C/PIA/102 Y 1
TS-A/PPLS
86A PPLS
ANNUNCIATOR "PPLS-A SAFEGUARD SIGNAL"
DERIVED SIGNAL CUTOFF SWITCH CS-A1/SP-A
86A1 PPLS
ANNUNCIATOR "PPLS-A1 SAFEGUARD SIGNAL"
CIRCUIT SHOWN IN STANDBY
INITIATING CONTACTS OPEN
CIRCUIT "B" IS SIMILAR

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	022 K1.01	
Level of Difficulty: 3	Importance Rating	3.5	

Containment Cooling System: Knowledge of the physical connections and/or cause effect relationships between the CCS and the following systems: SWS/cooling system.

Question: 15

Given the following conditions:

- The plant was previously at 100% power during the summer.
- A Loss of Component Cooling Water has occurred.
- Raw Water has been aligned to the Containment Cooling Coils.
- River water level is at 980' and slowly lowering.
- Containment temperature is 150°F.

Which of the following is the effect of these conditions?

A. Flashing and potential water hammer in the Containment Cooling Coils.

B. Raw Water Pumps are cavitating due to insufficient Net Positive Suction Head.

C. Containment Cooling Coils become plugged due to sediment.

D. Raw Water Pump seals become damaged unless backup cooling is aligned.

Answer: A

K/A Match:
Applicant must understand the implications of aligning the Raw Water System to the Containment Cooling Coils when river level is less than 983.5'.

Explanation:

A. **Correct**. Lowering River level affects the discharge pressure of the Raw Water Pump. This in turn impacts the cooling water flow available to the Containment Cooling Coils.

B. Incorrect. Plausible because River level does affect NPSH to the Raw Water Pump but that does not occur until 973' 9" (Raw Water Pump minimum suction elevation). Additionally, use of the Missouri River is limited to a minimum level of 973' 9" when RCS temperature is greater than 210°F.

C. Incorrect. Plausible because sediment in the Raw Water System is a concern and the Raw Water Strainers have a limit on hydraulic resistance but the strainers are designed to keep sediment out of Raw Water System components.

D. Incorrect. Plausible because pump seals do get damaged in this configuration but they are pumps that are being provided Raw Water. See CAUTION 3

Technical Reference: AOP-11, Step 13.c CAUTION, Rev. 16
(Attach if not previously provided including revision number) OI-RW-1, Precautions 1 & 3, Rev. 108

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 4-43-16, Containment Air Cooling and Filtering-Licensed Operator
EO 1.2c - **STATE** the functional relationship between the Containment Air Cooling and Filtering System and the following: Raw Water System.
EO-1.8 - **EXPLAIN** the principles of Emergency operation of the Containment Air Cooling and Filtering System in terms of flowpaths, major parameters (temperature, pressure, flowrate, etc.), alarms and control devices.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 8
55.43 ____

Error! Reference source not found.. (continued)

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CAUTIONS

1. To prevent Raw Water Pump damage, maintain flowrates between 1700 and 8500 gpm per pump.

2. Backup cooling should not be used on Containment Cooling Coils with river level less than 983.5 ft or Containment temperatures greater than 150°F. Flashing and potential water hammer in the Containment Coils may be present with Raw Water aligned under low river conditions (< 983.5 ft) and at higher temperatures.

3. To prevent seal degradation, it is desirable to align RW to operating HPSI, LPSI and CS Pumps. During Shutdown Cooling operations, it should be aligned within 1 hour with RCS Temperature > 175°F. However, after a RAS, a loss of cooling water can be tolerated with no loss of pump performance.

****

c. IF VA-3A, Containment Vent Fan, is in service,
THEN establish RW Flow to VA-1A by performing the following:

1) *Place "CNTMT CLG COIL VA-1A AC VLVS CONTROL SW HCV-400B/D" in "CLOSE".*

(continue)

PRECAUTIONS

1. Limiting Missouri River parameters for operation at, or above, an RCS temperature of 210°F are:
   - Minimum Level - 976 feet 9 inches
   - Maximum Level - 1,009 feet
   - Maximum Temperature - 87°F
2. If for any reason HCV-2893 or HCV-2894 are closed or the East Raw Water Header is isolated, ensure that at least one of the EFWST backup water supplies listed in AOP-30 are available.
3. Raw Water Pump minimum suction elevation is 973 feet 9 inches.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	026 G 2.1.7	
Level of Difficulty: 3	Importance Rating	4.4	

Containment Spray System: Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Question: 16

Given the following conditions:

- The Plant was at 100% power when an Uncontrolled Heat Extraction event occurred.
- The following conditions exist:
  - Steam Generator RC-2A pressure is 750 psig.
  - Steam Generator RC-2B pressure is 340 psig.
  - Reactor Coolant System pressure is 1235 psia.
  - Containment pressure is 12 psig.
  - SI-3A, Containment Spray Pump is running.
  - SI-3B, Containment Spray Pump breaker failed to close.

What is the status of the HCV-344 and HCV-345, Containment Spray Header Isolation Valves?

HCV-344…

A. …and HCV-345 both open.

B. …opens. HCV-345 remains closed.

C. …remains closed. HCV-345 opens.

D. …and HCV-345 both remain closed.

Answer: C

K/A Match:

Applicant must understand the operating characteristics of the Containment Spray System. Specifically, the relationship between the Containment Spray Pumps and Containment Spray Isolation Valves. Additionally, the applicant must determine that a valid Containment Spray Actuation Signal is present given the Steam Generator pressures listed.

Explanation:

A. Incorrect. Plausible because containment conditions are such that both CS Isolation Valves should be open. These valves receive a permissive signal from their associated CS Pump breaker. Failure of the breaker to close keeps the valve from opening.
B. Incorrect. Plausible if thought that HCV-344 worked in tandem with CS Pump SI-3A.
C. **Correct**. In order for HCV-344 to open, its associated CS Pump motor breaker must be closed. Analysis has shown that if only one CS pump is running with both CS isolation valves open, the pump will operate at runout and the motor may be damaged.
D. Incorrect. Plausible if thought that a Containment Pressure High Signal (CPHS) was not present since a CPHS & PPLS must be present to initiate a CSAS and this in conjunction with an SGLS ultimately starts Containment Spray Pumps and opens CS Isolation Valves.

Technical Reference: LP 7-11-22, Slide #182 to #186, Rev. 3

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-22, Safety Injection, Containment Spray & SDC-LO
EO 1.4 - **EXPLAIN** the operation of controls located in the Control Room associated with ECCS.
EO 1.8b - **EXPLAIN** overall system response to actuation of automatic engineered safeguards signals: Containment Spray Actuation Signal (CSAS)

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

EO *1.4 **(Slide #182)**

Major Component Description

Containment Spray Header Isolation Valves (HCV-344 & HCV-345)

Air-operated isolation valve on each spray header.

Valves fail open on a loss of air or power to the air solenoids

EO *1.4 **(Slide #183)**

Major Component Description

Containment Spray Header Isolation Valves (HCV-344 & HCV-345)

HCV-344 and HCV-345 are controlled from control switches (HC-344 & HC-345) on AI-30A/B (OPEN/AUTO/OVERRIDE).

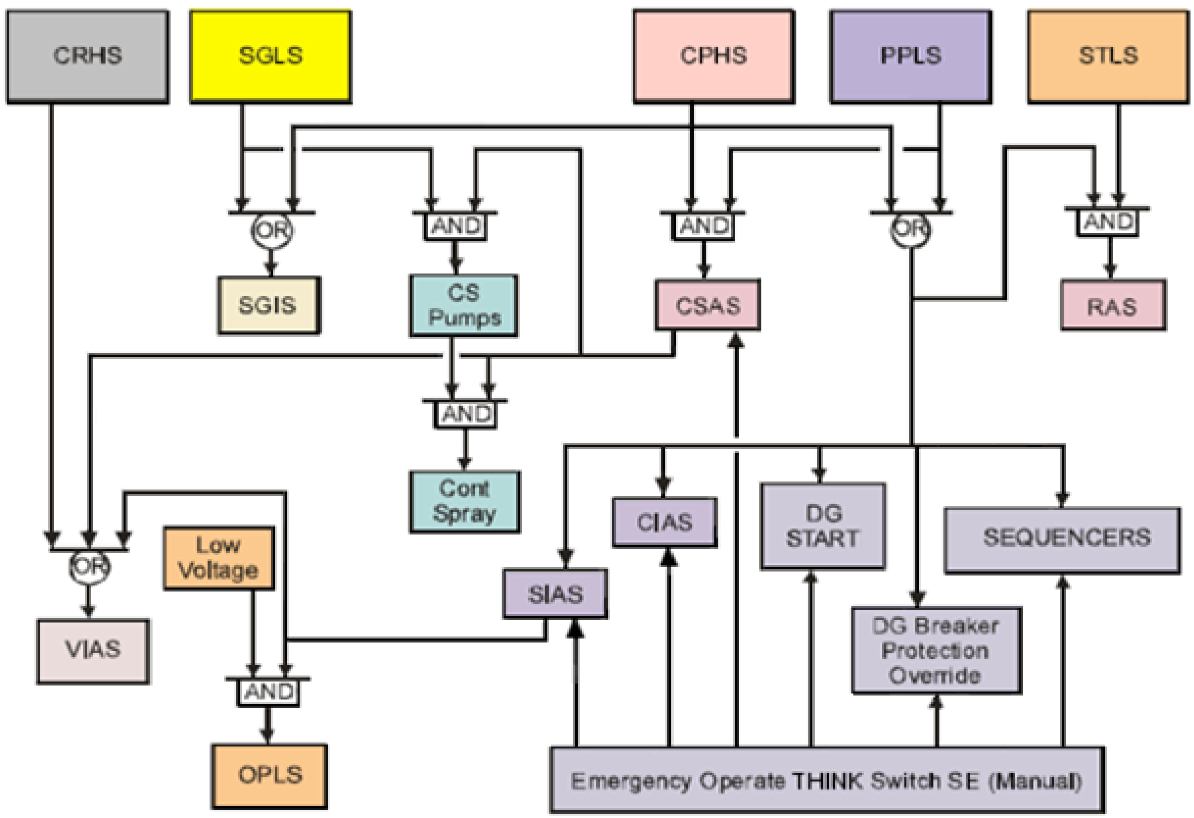
An E/P controller is used to close the valves to isolate spray flow when the control switch is placed in OVERRIDE.

Analysis has shown that if only one CS pump is running with both CS valves open, the pump will operate at run-out and the motor may be damaged.

(a) HCV-344 opens upon receipt of Containment Spray Actuation (CSAS) and SI-3B pump motor breaker closure.

(b) HCV-345 opens upon receipt of CSAS and SI-3A pump motor breaker closure.

(Slide #184)



EO *1.4 **(Slide #185)**

Major Component Description

Containment Spray Header Isolation Valves (HCV-344 & HCV-345)

HCV-344 and HCV-345 fail open on loss of air or electrical power to the air solenoid (S1).

The control air systems are equipped with automatic backup Nitrogen supply.

Normal 2265 psig regulated to 100 psig and then regulated to 80 psig.

Will be provided only if IA pressure is below 70 psig.

Nitrogen backup maintains the valve operable for 4 hours, operation beyond that may be possible but is not credited.

EO *1.4 **(Slide #186)**

Major Component Description

Containment Spray Header Isolation Valves (HCV-344 & HCV-345)

Two solenoid valves (S1 and S2) are in the air supply line to HCV-344 and HCV-345 (powered from opposite DC buses).

Solenoid valve S1 provides the normal air supply path to the spray valve operator (normally energized).

Solenoid valve S2 provides an alternate air supply (when energized, 01-TEST/HC-344(345) test switch in TEST) to hold the containment spray valve closed during spray pump testing (normally de-energized).

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	039 K3.05	
Level of Difficulty: 3	Importance Rating	3.6	

Main and Reheat Steam System: Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: RCS.

Question: 17

Given the following conditions:

- Plant is at 50% power at Beginning-Of-Life.
- Regulating Group 4 CEAs are at 100".
- A transient occurs which causes Reactor Power to lower and Reactor Coolant System T_{COLD} to rise.

Which of the following malfunctions would affect the Reactor Coolant System in this way?

Inadvertent…

A. …CEA insertion

B. …CEA withdrawal.

C. …closing of the Turbine Control Valves.

D. …opening of the Turbine Control Valves.

Answer: C

K/A Match:
Applicant must understand the effect of moderator temperature on the reactor as well as secondary systems that affect reactivity.

Explanation:

A. Incorrect. Plausible because RCS temperature would lower, however, so would power level.
B. Incorrect. Plausible because both power and temperature would rise with an inadvertent CEA withdrawal.
C. **Correct**. At BOC and 50%power, the MTC will be negative. Therefore, a transient that causes Reactor Power to lower and RCS temperature to rise must involve a decrease in secondary heat removal.
D. Incorrect. Plausible because the malfunction involves the Turbine Control Valves. If there were an inadvertent opening power would rise and temperature would lower.

Technical Reference: LP 7-15-12, Pages 11-15, Rev. 4
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-15-12, Transient and Accident Analysis-Licensed Operator
EO 1.2d - **EXPLAIN** the behavior of the primary and secondary plant for the following transients: A negative reactivity addition.

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

Question History: Last NRC Exam 2007 NRC Exam

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

10 CFR Part 55 Content: 55.41 1
55.43

1.1 Plant Response to transients

10. The first transient analyzed will be a 5% increase in steam flow to the main turbine from 90% power.

a. Initial conditions:

Reactor power = 90%
T_{ave} = 566EF
Pressurizer level= 60%
Pressurizer pressure = 2100 psia

b. The first step is to estimate the change in RCS temperature due to the increase in power. This can be done by looking at a reactivity balance. The decrease in RCS temperature will add positive reactivity due to the MTC. The increase in power will add negative reactivity due to the FTC. Typical values for FTC and MTC are:

MTC = $-1.5\times10^{-4}\ \Delta\rho/°F$
FTC = $-1.5\times10^{-4}\ \Delta\rho/\%$ power

These are approximate middle of cycle values.

A 5% increase in reactor power will supply $-7.5\times10^{-4}\ \Delta\rho$.

In order to balance reactivity, the MTC must supply $+7.5\times10^{-4}\ \Delta\rho$. This

means that temperature must change by:

$\Delta T = 7.5x10^{-4}/-1.5x10^{-4} = -5.0°F$

When this transient was run using the computer code CEPAC, RCS average temperature decreased by 5°F.

11. The next case looks at a 10% decrease in steam flow to the main turbine starting from 100% power.

a. The initial conditions for this case are:

Reactor power = 100%
T_{av} = 568EF
Pressurizer level = 60%
Pressurizer pressure = 2100 psia

b. The first step is to perform a reactivity balance to determine the change in T_{av}. Using the values of MTC and FTC above, the power decrease will increase reactivity by $1.5x10^{-3}$ $\Delta\rho$. T_{av} will need to increase by 10EF to balance that reactivity change.

CEPAC predicts a 10°F increase in temperature.

c. A 10EF increase in T_{av} is expected to produce a 10% increase in pressurizer level according to the thumb rule.

CEPAC calculates a 9% increase in pressurizer level.

d. According to the pressure increase thumb rule, pressure will increase by 20 psi for every 1% increase in pressurizer level. This means that pressure is expected to increase by 200 psia.

CEPAC calculates a 240 psia increase.

e. Again notice that CEPAC indicates that power will level out above 90%. This is due to an increase in pressurizer pressure following the increases RCS temperatures.

12. The next transient to be analyzed will be a reactivity increase due to a control rod withdrawal from 90% power. CEA withdrawal will continue until T_{av} increases by 7EF.

a. The initial conditions are the same as for case 1.

b. We already know that T_{av} will increase by 7°F. Using our thumb rules we can calculate the following:

Pressurizer level will increase by 7%.

Pressurizer pressure will increase by 140 psia.

CEPAC calculates a 6% level increase and a 150 psia pressure increase

c. The CEPAC results calculate a steady state increase in reactor power of 4%. This is the result of an increase in steam generator pressure which increased steam flow.

13. The next transient is a decrease in reactivity due to rod insertion. Control rods will be inserted until T_{av} decreases by 13°F.

a. The initial conditions for this transient are the same as for case 2.

b. Using the pressurizer level thumb rule, pressurizer level will be expected to decrease by 1% for each °F or 13%.

CEPAC calculates a 11% level decrease.

c. According to the pressure decrease thumb rule, we would expect pressure to decrease by about 130 psia for this case.

CEPAC calculates a 70 psi decrease due to this transient. The thumb rule appears to overpredict the pressure response for transients initiated by control rod movement. (or else CEPAC underpredicts it.)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	059 A3.02	
Level of Difficulty: 3	Importance Rating	2.9	

Main Feedwater System: Ability to monitor automatic operation of the MFW, including: Programmed levels of the SG.

Question: 18

What is the response of the Feedwater Regulating System to a high narrow range level in the Steam Generator?

The MINIMUM level where Feedwater Regulating Valves receive a closed signal is…

A. …80% level. Controller transfers to MANUAL with 0% output and remains there until HI-HI LEVEL alarm clears.

B. …84% level. Controller transfers to MANUAL with 0% output until the high level resets.

C. …80% level. Controller remains in AUTO with 0% output until the HI-HI LEVEL alarm clears.

D. …84% level. Controller remains in AUTO with 0% output until the high level resets.

Answer: B

K/A Match:

Applicant must be able to predict the response of the Feedwater System to levels that are pre-programmed in the controller.

Explanation:

A. Incorrect. Plausible because the controller will transfer to manual at 0% but that signal will clear at 2% below actuation of 84%. The HI-HI LEVEL alarm comes in at 75%.
B. **Correct**. The setpoint is correct. The controller transfers to MANUAL at 0% out but then resets at 2% below actuation. The Feed Regulating Valve will return to the position it held prior to the high level but remain in MANUAL.
C. Incorrect. Plausible if thought that the controller would remain in AUTO but it shifts to MANUAL with 0% output and remains there until ~82% narrow range level.
D. Incorrect. Plausible because the level setpoint is correct but the controller transfers to MANUAL.

Technical Reference: ARP-CB-4/A8, Window B-5U, Rev. 26
(Attach if not previously provided including revision number) LP 7-11-11, Slide #103, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-11-11, Feedwater and Feedwater Regulating Systems-LO EO 2.6 - **EXPLAIN** the operation of the Feedwater Control System during a Steam Generator High Downcomer Level condition

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	None	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

ARP-CB-4/A8, Window B-5U

Panel: **CB-4**	Annunciator: **A8**	Window: **B-5U**
STEAM GENERATOR A HI-HI LEVEL **SAFETY RELATED**		**FEEDWATER CONTROL STEAM GENERATOR RC-2A LEVEL HI-HI**
Tech Spec References: 2.15		
Initiating Device DCS (LY0903)	Setpoint 75.0%NR	Power IB-3A or MPP-66

OPERATOR ACTIONS

1. Check all available level indication for RC-2A. (A-D/LI-901, DCS Secondary)

1. IF RC-2A level is greater than 89%NR, THEN trip the Reactor and GO TO EOP-00.

2. IF RC-2A level is greater than 84%NR, THEN perform the following:

   2.1 Check FCV-1101, Feedwater Regulating Valve automatically closed.

   2.2 Manually restore level to the normal operating range per OI-FW-3.

3. Check Steam Generator RC-2A level at DCS point LY0903.

   3.1 IF DCS alarms are present,
   THEN refer to ARP-DCS-FW.

4. Dispatch an operator to check RC-2A level on LI-903Y-1 (AI-179).

5. Determine the cause of the Steam Generator HI-HI level.

EO *2.6 **(Slide #103)**

Major Component Description

Major Feedwater Control Components

Feedwater Flow Controller

The Feedwater Flow Controller also receives high S/G level signal to initiate FRV closure on high downcomer level at 84%.

The level controller will be switched to MANUAL and a 0% signal is generated closing the FRV.

The high level will reset at ~2% below actuation.

The FRV will return to the position it held prior to the high level.

Operator action may be required to prevent level from returning to the high level condition again.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	061 K2.02	
Level of Difficulty: 2	Importance Rating	3.7	

Auxiliary/Emergency Feedwater System: Knowledge of bus power supplies to the following: AFW electric drive pumps.

Question: 19

Control power for FW-6, Motor Driven Auxiliary Feedwater Pump is supplied from _____(1)_____ while motor power is supplied from _____(2)_____.

A. (1) DC Bus 1
   (2) Bus 1A3

B. (1) DC Bus 2
   (2) Bus 1A3

C. (1) DC Bus 2
   (2) Bus 1A4

D. (1) DC Bus 1
   (2) Bus 1A4

Answer: A

K/A Match:

Applicant must know source of control power as well as motor power for FW-6, MDAFW Pump.

Explanation:

A. **Correct**. FW-6 receives it starting control power from DC Bus 1 and its motor power from Bus 1A3.
B. Incorrect. Plausible because its motor power is from Bus 1A3 but control power is from DC Bus 1.
C. Incorrect. Plausible if thought that the even-numbered (FW-6) pump matched even-numbered (DC Bus 2 & Bus 1A4) Buses.
D. Incorrect. Plausible because its control power is from DC Bus 1 but motor power is from Bus 1A3.

Technical Reference: (Attach if not previously provided including revision number)
AOP-32, Attachment B, Step 3, Rev. 21
LP 7-11-1, Slide #62, Rev. 3
ARP-AI-66A/A66A, Window 15, Rev. 19

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-1, Auxiliary Feedwater System-Licensed Operator
EO 1.6 - **STATE** the normal and alternate power supplies for each major component of the Auxiliary Feedwater System.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	8
	55.43	

3. Bus 1A3

AC-10A	RW Pump
AC-10C	RW Pump
FW-6	Electric AFW Pump
RC-3C	RC Pump

Panel: **AI-66A**	Annunciator: **A66A**	Window: **15**

AUXILIARY FEEDWATER PUMP FW-6 FAIL
TO START ON AUTOMATIC DEMAND

FW-6
FAILED TO START
ON DEMAND

SAFETY RELATED

Tech Spec References: 2.5

Initiating Device 74-1/FW-6 Setpoint Breaker OPEN + Start Power DC Bus 1

OPERATOR ACTIONS

3.0 Dispatch an operator to inspect FW-6 and its breaker (1A3-16).

1.2 Manually start FW-6 by depressing the Aux Feedwater Pump FW-6 start pushbutton (AI-66A).

2. IF required to provide Aux Feed to Steam Generators, THEN ensure FW-10 is operating.
3. Refer to Technical Specification 2.5.
4. Initiate notification to the Work Week Manager.

EO 1.6 **(Slide #62)**

Major Component Description

Auxiliary Feedwater Pump (FW-6)

FW-6 is driven by a three-phase, 60 cycle induction motor powered from 4160 V bus 1A3.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	062 A4.01	
Level of Difficulty: 3	Importance Rating	3.3	

AC Electrical Distribution System: Ability to manually operate and/or monitor in the control room: All breakers (including available switchyard)

Question: 20

Given the following condition:

- Fort Calhoun Station is in a normal electric plant alignment when 63FPX-1/T1A-3, Transformer T1A3 Sudden Pressure Relay actuates.

Which of the following breakers received a trip signal from the 63FPX-1/T1A-3 relay?

A. Breakers 110 and 111 only.

B. All 161 KV Switchyard Circuit Breakers.

C. Breakers 1A31, 1A33, 1A42, and 1A44 only.

D. Breakers 1A31, 1A33, 1A42, and 1A44, 110 and 111.

Answer: D

K/A Match:

Applicant must know response of switchyard and plant breakers following a transformer fault.

Explanation:

A. Incorrect. Plausible because Breakers 110 and 111 are the feeds from 161 KV Offsite Power and would deenergize Transformers T1A3 and T1A4.
B. Incorrect. Plausible because the 161 KV Line could experience overload but in this condition only the high-voltage side of the Transformer is deenergized.
C. Incorrect. Plausible because opening these breakers prevents a potential back feed however it does not remove power from the Transformer.
D. **Correct**. Tripping of the Sudden Pressure Relay removes incoming power from the 161 KV Line (Breakers 110 and 111) and also isolates the low-voltage side of the Transformers (Breakers 1A31, 1A33, 1A42, and 1A44). This prevents back feeding into the Transformer from continuing to feed the problem that caused the sudden pressure.

Technical Reference: LP 7-13-1,Slide #50, #63, Rev. 2
(Attach if not previously provided including revision number) ARP-CB-20/A17, Window D-2, Rev. 30

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-1, High-Voltage/Grid System-Licensed Operator
EO 1.3b - **IDENTIFY** 161 KV substation components and protective relaying schemes.
EO 1.5c - **EXPLAIN** how the system configuration is manipulated from the Control Room including: Control and protection of system equipment.

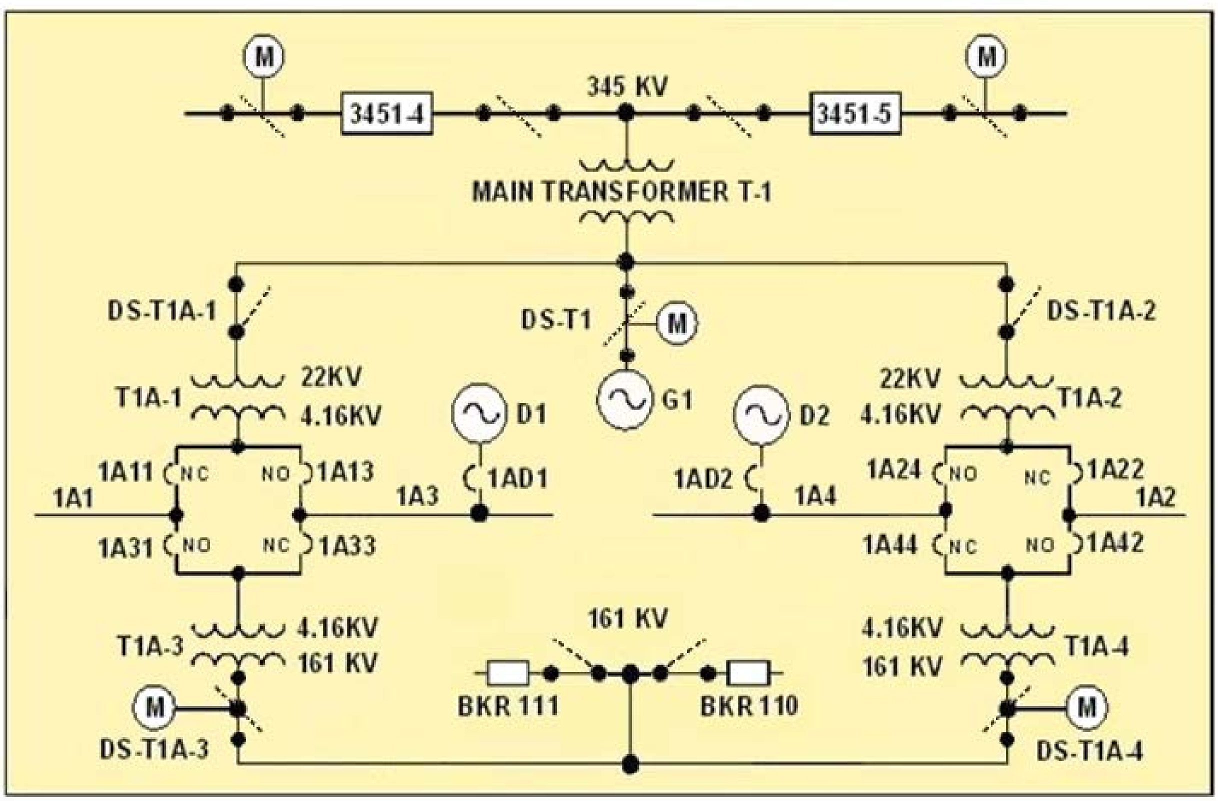
Question Source: Bank # X
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam None

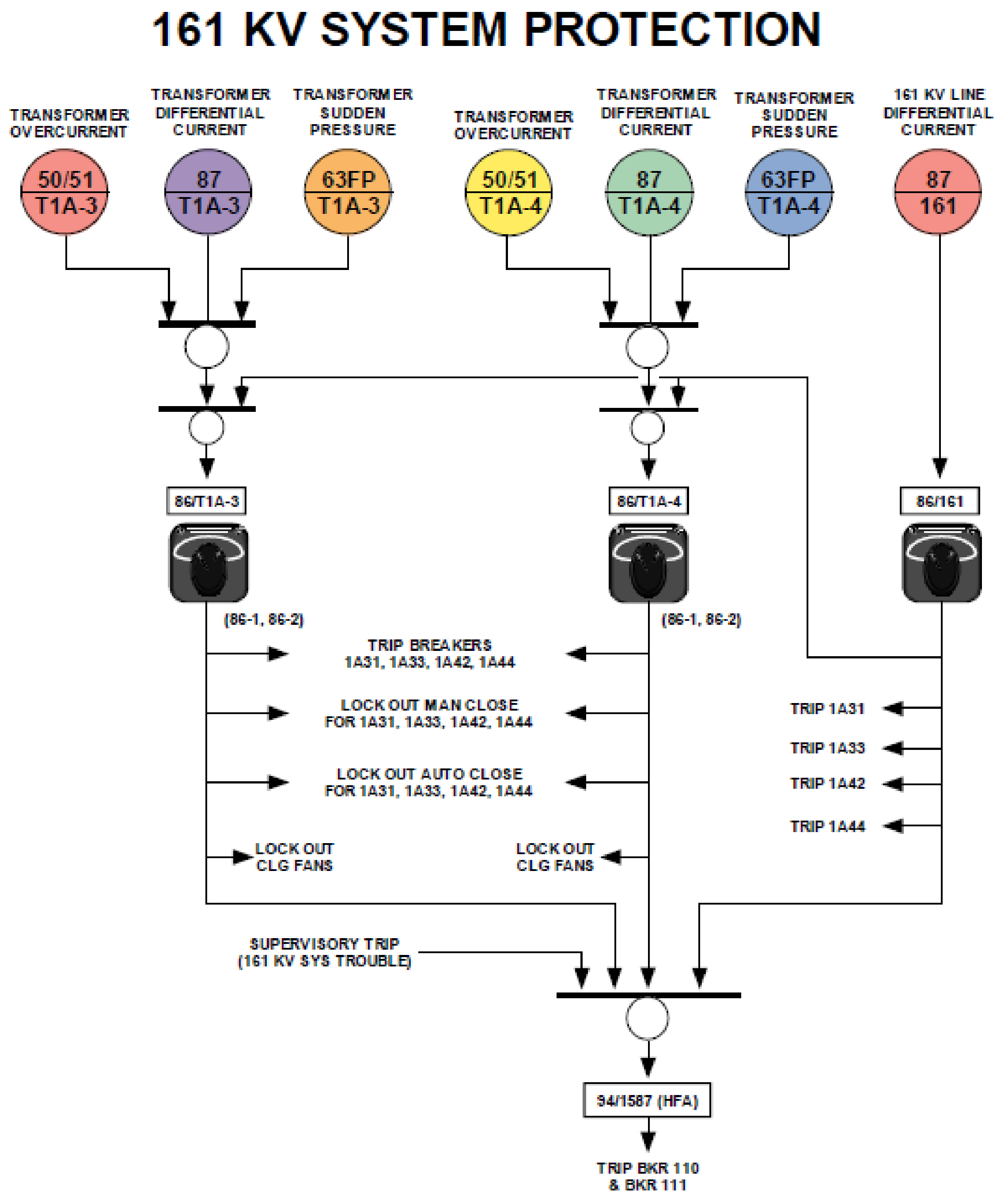
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7
55.43

(Slide #50)



(Slide #63)


161 KV SYSTEM PROTECTION
TRANSFORMER OVERCURRENT
50/51 T1A-3
TRANSFORMER DIFFERENTIAL CURRENT
87 T1A-3
TRANSFORMER SUDDEN PRESSURE
63FP T1A-3
TRANSFORMER OVERCURRENT
50/51 T1A-4
TRANSFORMER DIFFERENTIAL CURRENT
87 T1A-4
TRANSFORMER SUDDEN PRESSURE
63FP T1A-4
161 KV LINE DIFFERENTIAL CURRENT
87 161
86/T1A-3
86/T1A-4
86/161
(86-1, 86-2)
(86-1, 86-2)
TRIP BREAKERS 1A31, 1A33, 1A42, 1A44
LOCK OUT MAN CLOSE FOR 1A31, 1A33, 1A42, 1A44
LOCK OUT AUTO CLOSE FOR 1A31, 1A33, 1A42, 1A44
LOCK OUT CLG FANS
LOCK OUT CLG FANS
TRIP 1A31
TRIP 1A33
TRIP 1A42
TRIP 1A44
SUPERVISORY TRIP (161 KV SYS TROUBLE)
94/1587 (HFA)
TRIP BKR 110 & BKR 111

Panel: **CB-20**	Annunciator: **A17**	Window: **D-2**

HOUSE SERVICE TRANSFORMER T1A-3
LOCKOUT RELAY OPERATED

SAFETY RELATED

TRANS T1A-3 LOCKOUT RELAY OPERATED 86/T1A3

Tech Spec References: 2.7

Initiating Device 86 / T1A-3 Setpoint TRIPPED Power AI-41A

OPERATOR ACTIONS

1. IF Reactor Trip occurs, THEN GO TO EOP-00.

2. Verify the following actions automatically occur:

   - Breakers are tripped and locked out:

     1A31 1A33 1A42 1A44

   - 161KV Breakers 110 and 111 are tripped
   - Breakers 1A13 and 1A24 Fast Transfer to re-energize 1A3 and 1A4 from 345KV
   - IF Bus 1A3 and/or 1A4 not energized, IMPLEMENT AOP-32

3. Notify Shift Manager and initiate notification to the Work Week Manager of lockout relay operation.

PROBABLE CAUSES

- Transformer Differential Relay (87/T1A-3)
- Transformer Overcurrent (50-51/T1A-3)
- Transformer Sudden Pressure Relay actuation (63FP/T1A-3) **[AR 04925]**
- 161KV incoming supply differential (87/161 via 86/161)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	062 K4.01	
Level of Difficulty: 3	Importance Rating	2.6	

AC Electrical Distribution System: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Bus lockouts.

Question: 21

Assuming all Fast Transfer permissives are made up, which of the following results in a Fast Transfer of Bus 1A1 from the 22 KV to the 161 KV source?

A. 86-1/G1, GENERATOR LOCKOUT RELAY actuation.

B. 86-2/T1A-4, TRANSFORMER T1A-4 LOCKOUT RELAY actuation.

C. 86/1A11, 4.16 KV INCOMING BREAKER 1A11 LOCKOUT RELAY actuation.

D. 86-2/BF4, 345 KV BREAKER FAILURE BACKUP RELAY actuation.

Answer: A

K/A Match:

Applicant must be familiar with AC Electrical Distribution System interlocks associated with a Fast Transfer.

Explanation:

A. **Correct**. When Lockout Relay 86-2/G1 actuates numerous automatic actions occur. These include trip and lockout of the 22 KV supply breakers to 4160 V Buses 1A1 (1A11), 1A2 (1A22), 1A3 (1A13), and 1A4 (1A24). With the Fast Transfer permissives satisfied for Bus 1A1 breaker 1A11 will trip and breaker 1A31 will close in ~ 3 cycles.

B. Incorrect. Plausible because this would prevent a fast transfer for Buses 1A2 and 1A4. Bus 1A4 is already aligned to 161KV. Plausible because it only inhibits fast transfer of Bus 1A2 in the normal alignment.

C. Incorrect. Plausible because if this occurred a slow transfer vice fast transfer would be generated.

D. Incorrect. Plausible because this does result in a loss of 345 KV for Bus 1A2 and 1A4. Incorrect because this relay actuation defeats all automatic transfers for Bus 1A11.

Technical Reference: ARP-CB-20/A14, Window D-1, Rev. 46
(Attach if not previously provided including revision number) LP 7-13-2, Slides #42 & #92, Rev. 0

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-2, 4160 V Distribution- Licensed Operator
EO 1.9 - **EXPLAIN** how the system responds automatically to malfunctions.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Panel: **CB-20**	Annunciator: **A14**	Window: **D-1**

MAIN GENERATOR TRIP

NON SAFETY RELATED

GENERATOR LOCKOUT RELAY OPERATED 86/G1

Tech Spec References: None

Initiating Device 86-1/G1 Setpoint Tripped Power AI-41A

OPERATOR ACTIONS

1. IF Reactor Trip occurs, THEN GO TO EOP-00.

2. Verify the following automatic actions occur:

   - Breakers trip & lockout 1A11, 1A13, 1A22, 1A24, Generator Field Breaker, 3451-4 & 3451-5
   - Breakers Fast Transfer 1A31, 1A33, 1A42, 1A44
   - Bus Duct Cooling Fans are tripped and locked out
   - Transformer Cooling tripped and locked out on T1, T1A-1, and T1A-2
   - Stator Cooling Pumps are tripped and locked out
   - Turbine Trip

3. IF backfeed from 345KV was established, THEN GO TO AOP-32.

EO 1.3 **(Slide #92)**

Major Component Description

4160V Supply Breakers (1A11, 1A13, 1A31, 1A33, 1A22, 1A24, 1A42, 1A44)

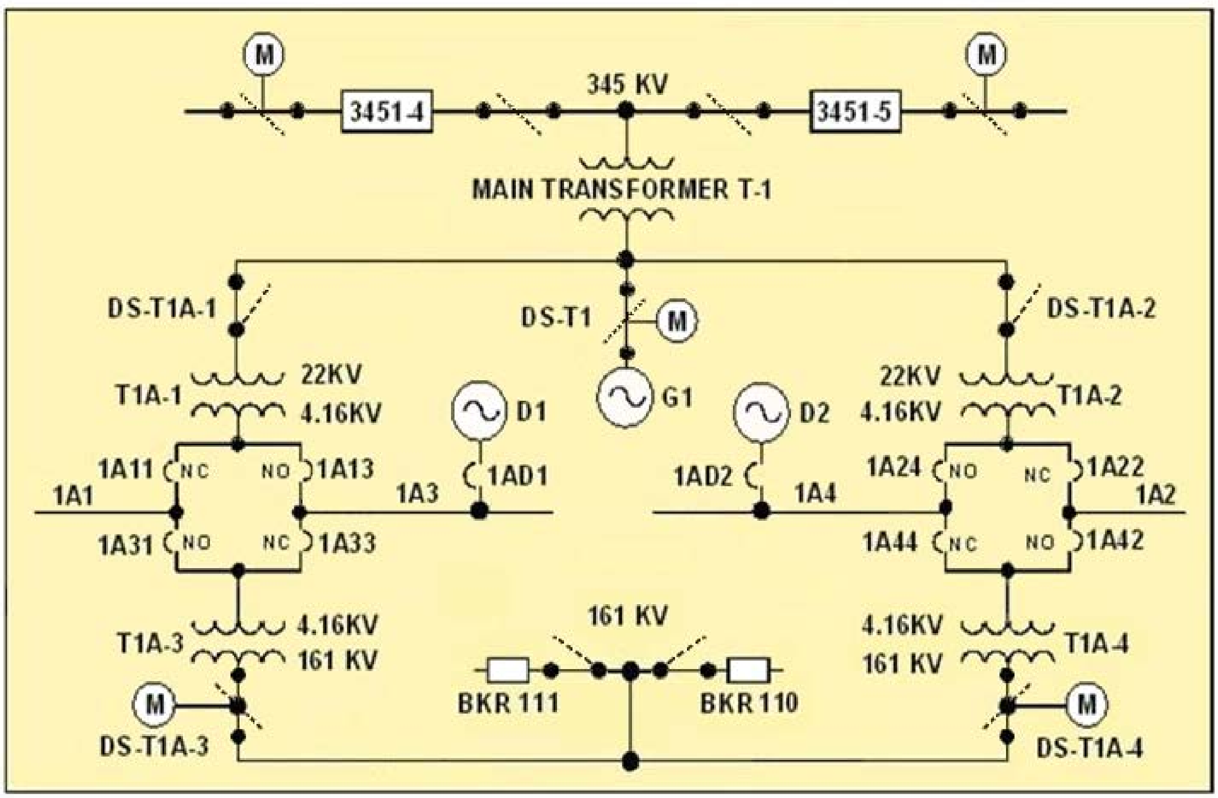
161KV/4160V Feeder Breaker Controls for Breaker 1A33 (1A44 is similar)

Automatic transfer closing (all contacts must be closed to energize closing coil):

NOTE: This is the transfer of bus 1A3 from the 22KV to the 161KV.

(a) 43/1A1-1A3/AUTO – Fast transfer switch on CB-20 in AUTO position.
(b) CS-AT/1A33 – Control switch for 1A33 in AFTER TRIP (green flag)
(c) CS-AC/1A13 – Control switch for 1A13 in AFTER CLOSE (red flag)
(d) 86/1A33 – No Lockout (LO) for breaker 1A33*Major Component Description*
(e) 86/1A13 – No LO for breaker 1A13 (the alternate feeder breaker)
(f) 86-2/T1A3 – No LO on supply transformer
(g) 86-2/T1A4 – No LO on other house service transformer
(h) 27T1X/1A3-13 – No UV on secondary windings of T1A-3
(i) 86A/OPLS – No OPLS present, OPLS blocks auto transfers

(Slide #42)



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/17/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	063 G 2.4.34	
Level of Difficulty: 3	Importance Rating	4.2	

DC Electrical Distribution System: Emergency Procedures/Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Question: 22

Given the following conditions:

- EE-8C, Battery Charger #1 Supply Breaker is open.
- DC Bus #1 is de-energized.
- Preparations are being made to restore power using Battery Charger #3.

Which of the following local actions is required to restore power to DC Bus #1 per OI-EE-3, 125 VDC System Normal Operation?

Ensure the preferred source is available from _____(1)_____.
Prior to energizing DC Bus #1, _____(2)_____ all breakers on 125 VDC Bus #1 Main Distribution Panel.

A. (1) MCC-3C1
(2) close

B. (1) MCC-4B1
(2) close

C. (1) MCC-3C1
(2) open

D. (1) MCC-4B1
(2) open

Answer: C

K/A Match:

Applicant must have knowledge of the correct power supply when using Battery Charger #3. Additionally, they must understand the purpose and use of the Kirk Key interlock.

Explanation:

A. Incorrect. Plausible because this is the correct MCC when powering DC Bus #1. Incorrect because the breakers must be open.

B. Incorrect. Plausible because MCC-4 B1 would be correct when powering DC Bus #2. Incorrect because the breakers must be open.

C. **Correct**. Per OI-EE-3, 125 VDC System Normal Operation, Attachment 4, Energizing a Deenergized DC Bus, power to DC Bus #1 is aligned from MCC-3C1. All input and output breakers in DC Bus #1 are opened prior to aligning the spare charger.

D. Incorrect. Plausible because the input and output breakers in DC Bus #1 are opened prior to aligning the spare charger. Incorrect because MCC-4 B1 is aligned when powering DC Bus #2.

Technical Reference: OI-EE-3, Attachment 4, Rev. 25

(Attach if not previously provided including revision number) LP 7-13-4, Slide #1, #8, #24, & #27, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-4, 125 VDC & 120 VAC Distribution-Licensed Operator
EO 1.2 - **EXPLAIN** the operation of each major component during all modes of operation.
EO 1.3 - **LIST** the primary (preferred) and alternate (if any) power supplies to each bus/component.

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

Question History: Last NRC Exam None

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

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

Attachment 4 - Energizing a Deenergized DC Bus

PROCEDURE (continued) (✓) INITIALS

2. IF energizing EE-8F, 125V DC Bus Number 1 Main Distribution Panel, using EE-8E, 125V DC Battery Charger Number 3,
THEN perform the following:

a. Ensure all input and output breakers on EE-8F, 125V DC Bus Number 1 Main Distribution Panel, are open.
b. Close EE-8F-CB1, Battery Number 1, EE-8A Main Breaker.
c. Ensure the following Battery Charger Number 3 breakers are OFF:

- EE-8E-CB1, Batt. Charger 3 (EE-8E) AC Input Breaker
- EE-8E-CB2, Batt. Charger 3 (EE-8E)DC Output Breaker

d. Ensure EE-8E, 125V DC Battery Charger Number 3, Voltage Selector is in FLOAT.
e. Place EE-114, Battery Charger 3 (EE-8E) AC Input Power Transfer Switch, in MCC-3C1 position.
f. Ensure Breaker MCC-3C1-A2L, EE-8E Battery Charger Number 3, is ON.

g. At Battery Charger Number 3, place 69/EE-8E, Alarm Permissive, in Normal.

h. Place EE-8E-CB1 in ON.
i. Place EE-8E-CB2 in ON.

2.

NOTE

Breaker EE-8F-CB2 operation will require a Kirk Key.

j. Place Breaker EE-8F-CB2, Batt Charger 3, EE-8E, in ON.

EO 1.1 **(Slide #8)**

System Interfaces

480V MCC-3B1 feeds DC Bus #1 through Battery Charger #1.

480V MCC-4A1 feeds DC Bus #2 through Battery Charger #2.

480V MCC-3C1 and MCC-4B1 can feed Battery Charger #3 via a manual transfer switch (EE-114).

(a) Battery Charger #3 can be lined up to feed either DC Bus #1 or DC Bus #2.

(b) The preferred line-up is for Battery Charger #3 to be fed from MCC-3C1 when lined up to DC Bus #1 and fed from MCC-4B1 when lined to DC Bus #2.

(c) This will maintain electrical separation of the DC buses.

EO 1.3 **(Slide #24)**

Major Component Description

Battery Chargers

Power Sources

Charger #1 – MCC-3B1

Charger #2 – MCC-4A1

Charger #3 – MCC-3C1 or MCC-4B1 via a manual transfer switch (EE-114).

NOTE: Modification DCN 2831 adds MCC-4B1 as an additional power source to Battery Charger #3. The change provides greater flexibility with the third charger capable of being fed from either Bus 1A3 (DG-1) or Bus 1A4 (DG-2) via their MCCs. This configuration will also support Tech. Spec. 2.7(2)i. Battery Charger #3 is a swing battery charger with MCC-3C1 being the preferred source when lined-up with DC Bus #1 and MCC-4B1 being the preferred source when lined-up with DC Bus #2.

EO 1.3 **(Slide #27)**

Major Component Description

Batteries

Power Sources – During high current demand the battery acts as a power source; however, under normal plant conditions the battery acts as a load.

Battery #1

(1) Normal supply – Charger #1

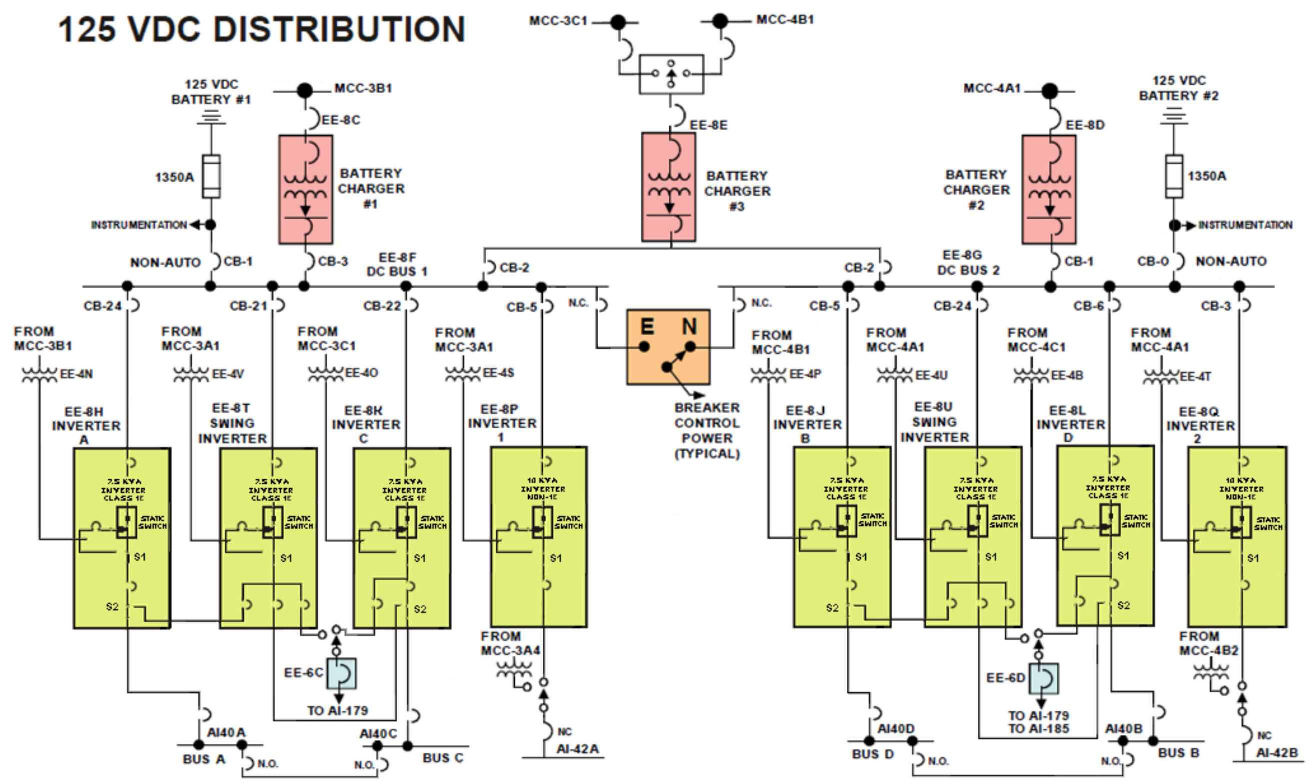
(2) Alternate supply – Charger #3

Battery #2

(1) Normal supply – Charger #2

(2) Alternate supply – Charger #3

(Slide #1)


125 VDC DISTRIBUTION
MCC-3C1
MCC-4B1
125 VDC BATTERY #1
MCC-3B1
EE-8C
EE-8E
MCC-4A1
EE-8D
125 VDC BATTERY #2
1350A
BATTERY CHARGER #1
BATTERY CHARGER #3
BATTERY CHARGER #2
1350A
INSTRUMENTATION
INSTRUMENTATION
NON-AUTO
CB-1
CB-3
EE-8F DC BUS 1
CB-2
CB-2
EE-8G DC BUS 2
CB-1
CB-0
NON-AUTO
CB-24
CB-21
CB-22
CB-5
N.C.
N.C.
CB-5
CB-24
CB-6
CB-3
E N
FROM MCC-3B1
FROM MCC-3A1
FROM MCC-3C1
FROM MCC-3A1
FROM MCC-4B1
FROM MCC-4A1
FROM MCC-4C1
FROM MCC-4A1
EE-4N
EE-4V
EE-4O
EE-4S
EE-4P
EE-4U
EE-4B
EE-4T
EE-8H INVERTER A
EE-8T SWING INVERTER
EE-8K INVERTER C
EE-8P INVERTER 1
BREAKER CONTROL POWER (TYPICAL)
EE-8J INVERTER B
EE-8U SWING INVERTER
EE-8L INVERTER D
EE-8Q INVERTER 2
S1
S2
FROM MCC-3A4
FROM MCC-4B2
EE-6C
TO AI-179
EE-6D
TO AI-179
TO AI-185
AI40A
AI40C
AI40D
AI40B
BUS A
N.O.
N.O.
BUS C
NC
AI-42A
BUS D
N.O.
N.O.
BUS B
NC
AI-42B

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/27/15	Tier #	2	
Change: 0	Group/Category #	1	
	K/A #	064 K1.02	
Level of Difficulty: 2	Importance Rating	3.1	

Emergency Diesel Generator System: Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: D/G cooling water system.

Question: 23

Which of the following Diesel Generator components are served by the Jacket Water subsystem?

1. Engine Crankcase (cylinder liners)
2. Starting Air
3. Scavenging Air
4. Lube Oil
5. Engine Radiator
6. Fuel Oil

A. 1, 2, 5

B. 2, 4, 6

C. 3, 5, 6

D. 1, 3, 4

Answer: D

K/A Match:

Applicant must be knowledgeable of the physical connections between Emergency Diesel Generator subsystems.

Explanation:

A. Incorrect. Plausible because Engine Crankcase and Engine Radiator are correct. Incorrect because Starting Air is not service by the Jacket Water Subsystem.

B. Incorrect. Plausible because Lube Oil is serviced by the Jacket Water Subsystem whereas Starting Air and Fuel Oil are not.

C. Incorrect. Plausible because Scavenging Air and Engine Radiator with the Jacket Water Subsystem. Incorrect because Fuel Oil does not.

D. **Correct**. Engine Crankcase, Scavenging Air, Lube Oil, and Engine Radiator are all served by the Jacket Water Subsystem.

Technical Reference: LP 4-23-7, Slides 128-131, & #156, Rev. 2
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 4-23-7, Diesel Generator Mechanical-AON
EO 4.2 - **STATE** the functional relationship between Jacket Water Subsystem and the following:
EO 4.2a - Scavenging Air Subsystem
EO 4.2b - Lube Oil Subsystem

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 8
55.43

EO 4.1 **(Slide #128)**

Jacket Water Subsystem

Provides cooling during diesel operations and maintains preheated conditions during standby, for faster more reliable starting of the diesel engine for the following:

1. Diesel engine crankcase
2. Scavenging Air Subsystem
3. Lube Oil Subsystem

EO 4.2 **(Slide #129)**

Jacket Water Subsystem

Interfaces

Jacket water flows through the engine crankcase in the discharge manifold and around the engine cylinder liners.

EO 4.2a **(Slide #130)**

Jacket Water Subsystem

Interfaces

Jacket water flows through the scavenging air aftercoolers.

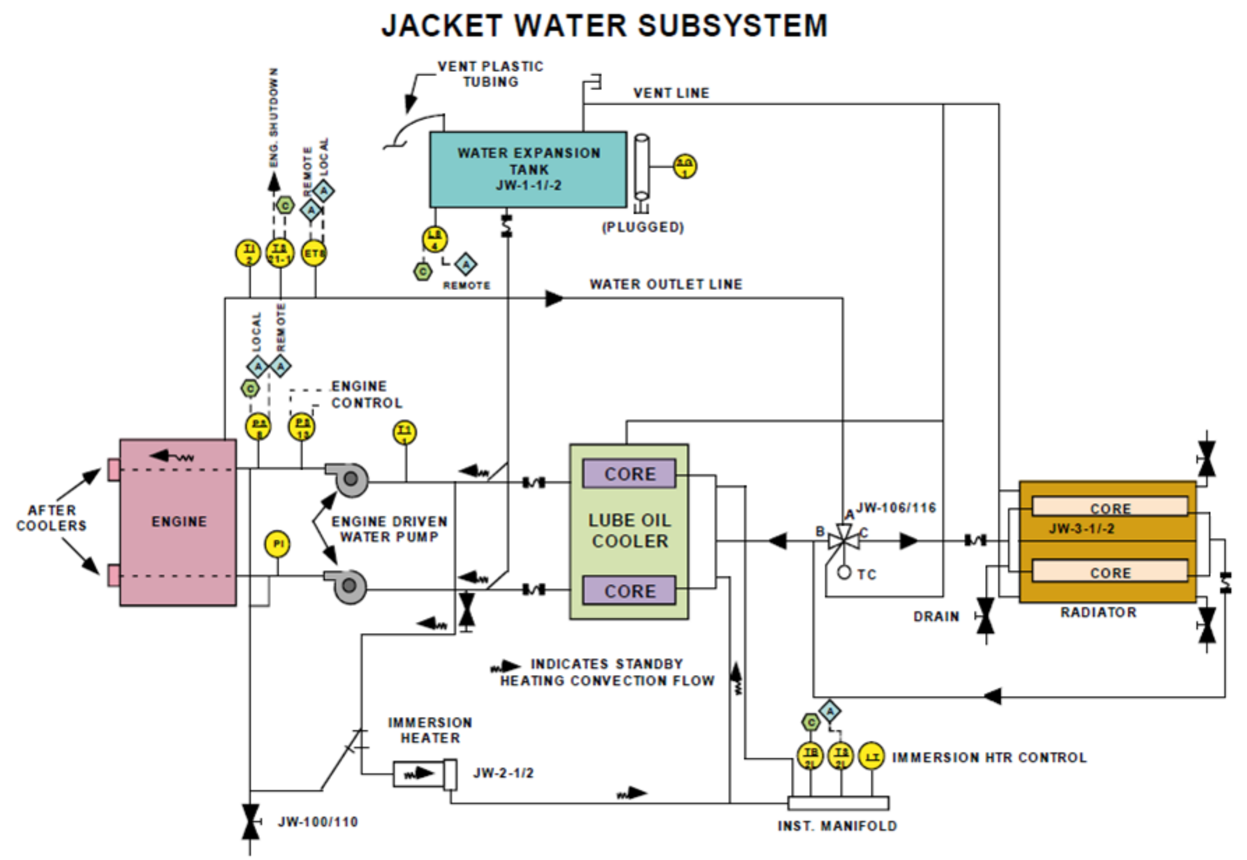
EO 4.2b **(Slide #131)**

Jacket Water Subsystem

Interfaces

Jacket water flows through the lube oil cooler.

(Slide #156)



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	064 A1.03	
Level of Difficulty: 4	Importance Rating	3.2	

Emergency Diesel Generator System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ED/G system controls including: Operating voltages, currents, and temperatures.

Question: 24

Given the following conditions:

- Emergency Diesel Generator (EDG) DG-1 is supplying Bus 1A3 loads.
- DG-1 load is 500 KW.
- T1A3, House Service Transformer is available to be returned to service.
- MVA-13, Restoring Offsite Power to Bus 1A3, is in progress.
- When the synchroscope is energized for Feeder Breaker 1A33, the following is observed:
  - INCOMING voltage is higher than RUNNING voltage.
  - Synchroscope is rotating slowly in the FAST direction.

Which of the following describes the required action after breaker closure?

Place the Governor Control Switch in…

A. …RAISE after breaker closure to prevent overload.

B. …LOWER after breaker closure to prevent reverse power conditions.

C. …LOWER after breaker closure to prevent overload.

D. …RAISE after breaker closure to prevent reverse power conditions.

Answer: D

K/A Match:
Candidate must understand how to avoid a reverse power condition on the Diesel Generator including understanding of synchronizing actions when the Safeguards Bus is in the isochronous mode.

Explanation:

A. Incorrect. Plausible because the Governor Control Switch must be placed in raise for the current condition of the Safeguards Bus but the reason is wrong. In the conditions listed a reverse power would occur. Voltage must also be raised to avoid reactive load transfer.

B. Incorrect. Plausible because the reason for adjusting the Governor Control Switch is correct but the direction is wrong and would result in a reverse power condition. Lowering voltage is incorrect as this would increase the reactive load transfer when the breaker was closed.

C. Incorrect. Plausible if thought that the DG operated as it normally does during surveillance testing.

D. **Correct**. Voltage must be raised because INCOMING (Offsite Power) voltage is currently higher than RUNNING (DG) voltage. Voltages should be approximately equal when paralleling to avoid a transfer in reactive load. Under normal conditions, with the synchroscope rotating slowly in the FAST direction, the DG would pick up load when the Output Breaker is closed. This condition is reversed when the DG is running in the ISOCHRONOUS mode, therefore, placing the Governor Control switch in RAISE would avoid a reverse power condition on the DG.

Technical Reference: MVA-13, Step 7 CAUTION, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: LP 7-13-5, 125 VDC & 120 VAC Distribution-Licensed Operator
EO 1.11 - **EXPLAIN** the operational use of the synchroscope and synchronizing lights.
EO 1.14 - **EXPLAIN** abnormal operation of the EDG.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 5
55.43 ____

(continued)

****

CAUTION

While paralleling, rotation of the synchroscope in the "FAST" direction will result in a reduction of load on the Diesel Generator when off-site power is synchronized to the bus. Reverse power may occur if less than 300 KW is loaded onto 1A3 while synchronizing to off-site power.

****

G. **IF** Diesel is loaded on the Bus, **THEN** set Diesel Generator, DG-1 Governor Droop Dial to the "SCRIBE MARK" (DG-1).

H. Synchronize and close Breaker 1A33.

I. **IF** Diesel Generator load drops below 300 KW, **THEN** open breaker 1AD1.

(continued)

e. Ensure the operable Isolated Bus Duct Cooling Unit is red-flagged.

f. Synchronize and close at least one of the following Generator Output Breakers:

- 3451-4
- 3451-5

g. Check that T1A-1 secondary voltage is greater than or equal to 4160 V.

h. Verify the "TRANS T1A-1 SECONDARY LOW VOLTAGE" alarm is clear (A16, A2).

(continue)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	073 A2.01	
Level of Difficulty: 4	Importance Rating	2.5	

Process Radiation Monitoring System: Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Erratic or failed power supply.

Question: 25

Given the following conditions:

- Plant is in MODE 1.
- RM-050, Containment Particulate Radiation Monitor, is in KEYPAD.
- RM-052, Containment/Auxiliary Building Vent Stack Swing Monitor and RM-062, Auxiliary Building Vent Stack Radiation Monitor, are both aligned to the Auxiliary Building Vent Stack.
- AOP-16, Loss of Instrument Bus, Section V, Loss of Instrument Bus AI-40D, is in progress.
- RM-052 was being powered from Instrument Bus AI-40D and MCC-4C2.

(1) Which of the following is the result of AI-40D deenergizing, and
(2) After RM-052 is re-aligned to its preferred power source, what is required?

A. (1) Only RM-052 is unavailable.
(2) Align RM-052 to monitor Containment.

B. (1) Both RM-052 and RM-062 are unavailable.
(2) Align RM-052 to monitor Containment.

C. (1) Only RM-052 is unavailable.
(2) Continue RM-052 monitoring of the Auxiliary Building Vent Stack.

D. (1) Both RM-052 and RM-062 are unavailable.
(2) Continue RM-052 monitoring of the Auxiliary Building Vent Stack.

Answer: D

K/A Match:

Applicant must be knowledgeable of the normal at-power Radiation Monitoring System alignment. Also must be aware of the multiple power supplies to RM-052. Normal and alternate power alignments for RM-052.

Explanation:

A. Incorrect. Plausible since power from AI-40D is aligned to RM-052, but RM-062 only gets power from AI-40D and MCC-4C2. Realigning RM-052 to monitor Containment is not done because it would leave the Auxiliary Building Vent Stack incapable of generating the Engineered Safeguard Signal for a CRHS. RM-062 must be re-powered before this action could be performed. With RM-050 in KEYPAD the applicant could think that RM-052 must be aligned to Containment, however, only RM-051 provides an input to the Containment Radiation High Signal (CRHS).

B. Incorrect. Plausible because both RM-052 and RM-062 are unavailable. RM-052 cannot be aligned to Containment until power is restored to RM-062. With RM-050 in KEYPAD the applicant could think that RM-052 must be aligned to Containment, however, only RM-051 provides an input to the Containment Radiation High Signal (CRHS).

C. Incorrect. Plausible because RM-052 must continue monitoring the Aux Building Vent Stack, however, both radiation monitors are unavailable due to a loss of power.

D. **Correct**. RM-052 is provided with 2 sets of power supplies (a 480 VAC and 120 VAC). One set is from AI-40C and MCC-3B1 while the other set is from AI-40D and MCC-4C2. When both RM-052 and RM-062 are aligned to the Auxiliary Building Vent Stack, RM-052 would normally be aligned to AI-40C and MCC-3B1. This prevents a loss of both RM-052 and RM-062 should AI-40D or MCC-4C2 deenergize. If RM-052 is the only Aux Building Vent Stack monitor in service, it will be powered from MCC-4C2 and AI-40D. This ensures compliance with Technical Specification 2.15.

Technical Reference: (Attach if not previously provided including revision number)

OI-RM-1, Precaution 3 & Attachment 2, Step 4.g CAUTION, Rev. 68

AOP-16, Section V, Step 13 NOTE, Rev. 20

LP 7-12-3, Slides #19, #138 & #197, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-3, Radiation Monitoring System-Licensed Operator

EO 4.0 - **EXPLAIN** the operations, actuations and applications of the individual radiation monitors.

EO 7.0 - **EXPLAIN** the overall operations of the Radiation Monitoring System using OI-RM-1 as a guide.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
- 55.41 11
- 55.43 ______

OI-RM-1, Attachment 2, Step 4.k CAUTION

CAUTION
Loss of either AI-40D or MCC-4C2 will render both RM-052 and RM-062 inoperable when RM-052 is not aligned to the preferred power supplies (AI-40C and MCC-3B1).

k. IF the preferred power supplies (AI-40C and MCC-3B1) are available, THEN align the following switches: (Rm 69)

- AI-81-SW1, 120 VAC Instrument, to C
- AI-81-SW2, 480V Power Supply Disconnect/ Selector Switch, to MCC-3B1-G1B Channel A

OI-RM-1, Precaution 3

PRECAUTIONS

3. All automatic functions are inoperable when the Control Room ratemeter is placed in KEYPAD. All requirements of the Technical Specifications and the ODCM must be met prior to placing the Control Room ratemeter to KEYPAD.

NOTE

Upon loss of Instrument Bus D, **ALL** of the following instrumentation or equipment associated with the **Containment Integrity Safety Function** is inoperable:

- RM-052 (depending upon plant conditions)
- RM-054B
- RM-055
- RM-062
- RM-063
- All Area Radiation Monitors on AI-33B
- RR-049A
- RR-099

13. Confirm Containment integrity by performing the following:

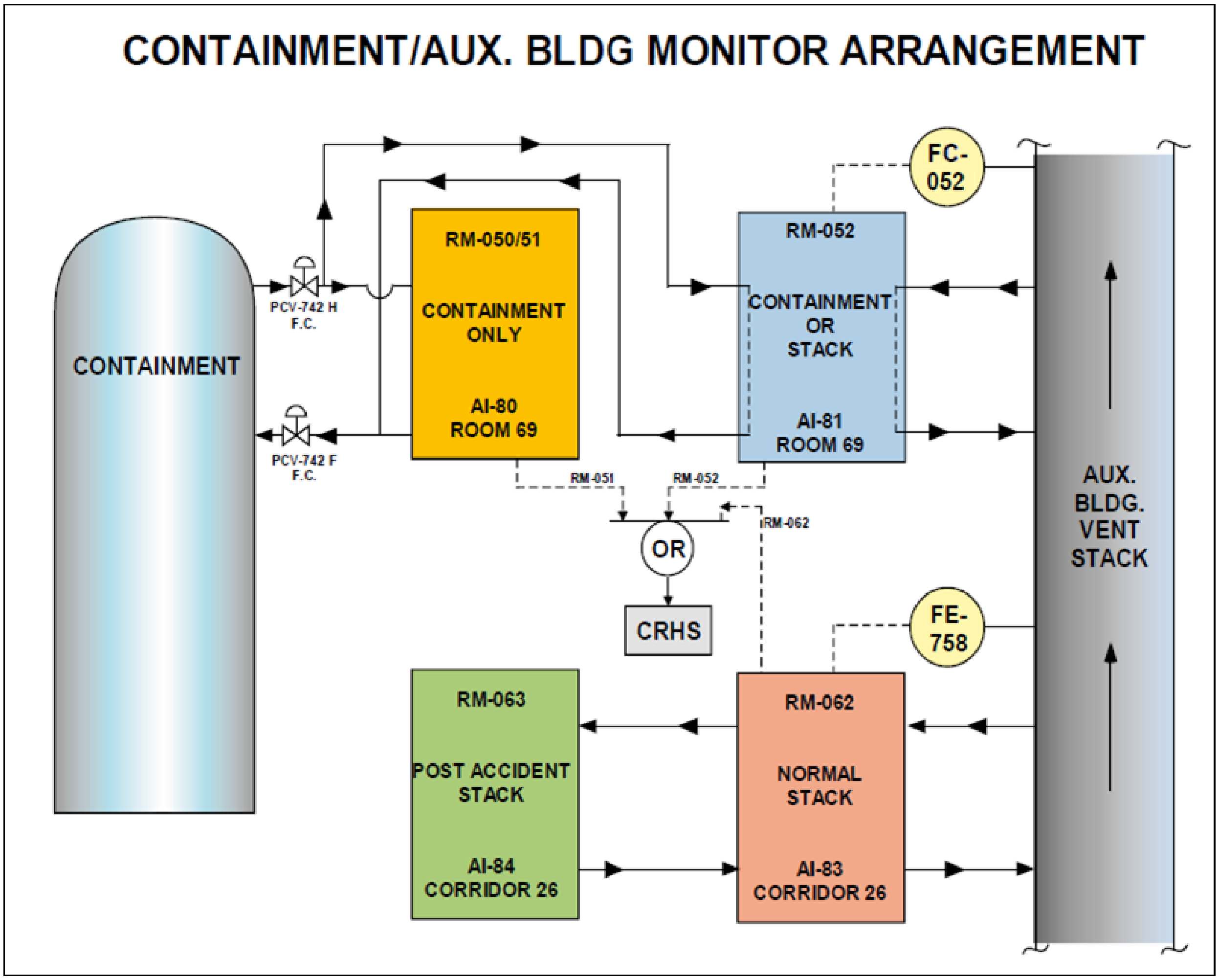
    a. Check for no unexpected rise in Containment Sump level.

15. **IF** RM-052 was powered from AI-40D, **THEN** place "AI-81-SW1" in "C" PER the RM-052 (Stack/CNTMT Gas) Attachment of OI-RM-1, Radiation Monitoring (Room 69).

EO 4.0 **(Slide #197)**
RM-052
Per OI-RM-1:
During normal operation RM-052 will be lined-up to monitor the Aux. Bldg. vent stack along with RM-062.
To ensure that one stack monitor will remain operable upon the loss of a single power supply, RM-052 will be powered from MCC-3B1 and AI-40C.
If RM-052 is the only vent stack monitor in service, it will be powered from MCC-4C2 and AI-40D. This ensures compliance with Technical Specification 2.15.

EO *4.1 **(Slide #138)**
RM-062
RM-062 ratemeters located on the skid and on Control Room panel AI-33A are powered from 120 VAC Instrument Bus AI-40D.
A High Alarm on the Control Room ratemeter will actuate a Containment Radiation High Signal.

(Slide #19)



CONTAINMENT/AUX. BLDG MONITOR ARRANGEMENT
CONTAINMENT
PCV-742 H
F.C.
PCV-742 F
F.C.
RM-050/51
CONTAINMENT ONLY
AI-80
ROOM 69
RM-052
CONTAINMENT OR STACK
AI-81
ROOM 69
FC-052
FE-758
RM-051
RM-052
RM-062
OR
CRHS
RM-063
POST ACCIDENT STACK
AI-84
CORRIDOR 26
RM-062
NORMAL STACK
AI-83
CORRIDOR 26
AUX. BLDG. VENT STACK

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	076 K4.06	
Level of Difficulty: 4	Importance Rating	2.8	

Service Water System: Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Service water train separation.

Question: 26

(1) Which of the following is the normal (at-power) position of HCV-2893 and HCV-2894, Raw Water System Backup Supply Valves, and
(2) How do they provide for Safe Shutdown Analysis concerns?

A. (1) Opened.
(2) Provide emergency backup cooling to the Containment Air Cooling and Filtering Units in order to achieve Cold Shutdown conditions.

B. (1) Opened.
(2) Ensures Cold Shutdown conditions can be achieved within 72 hours in the event of a fire that causes loss of all Component Cooling Water Pumps.

C. (1) Closed.
(2) Provide emergency backup cooling to the Containment Air Cooling and Filtering Units in order to achieve Cold Shutdown conditions.

D. (1) Closed.
(2) Ensures Cold Shutdown conditions can be achieved within 72 hours in the event of a fire that causes loss of all Component Cooling Water Pumps.

Answer: B

K/A Match:

Applicant must be familiar with the position and purpose of Raw Water System valves that are important to safety. This question addresses the Raw Water (service water) system cross-connect valves, HCV-2893 and HCV-2894. These valves, and their position, are part of the Technical Specification and Design Basis for the system and how it meets the design function of the system to be able to provide backup cooling and Emergency Feedwater Storage Tank fill capability. Because this is in the overall system purpose and LCO/USAR applicability, this is RO-level knowledge.

Explanation:

A. Incorrect. Plausible because these valves are maintained open. Incorrect because this is not a function of the Safe Shutdown Analysis.

B. **Correct**. AOP-18, Attachment C, Equipment Isolation associated with the Raw Water System restores HCV-2893 and HCV-2894 to their normal open position if the leak is not on either Raw Water Header. Maintaining these valves in the open position ensures that in the event of a loss of either of the headers the plant will still be able to achieve Cold Shutdown within 72 hours.

C. Incorrect. Plausible because providing emergency backup cooling to the Containment Air Cooling and Filtering Units is one of the purposes of the Raw Water System but this is not a reason for achieving Cold Shutdown. Incorrect because these valves are maintained in the open position.

D. Incorrect. Plausible because the reason is correct. Incorrect because these valves are maintained in the open position during normal operation.

Technical Reference: (Attach if not previously provided including revision number)
- TDB-VIII, Attachment 1, Rev. 64
- TDB-AOP-18, Step 15, Rev. 8a
- LP 7-11-19, Slides #14 & #113, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-19, Raw Water System-Licensed Operator
EO 1.7 - **EXPLAIN** how changes in plant conditions may affect the Raw Water System.

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

Question History: Last NRC Exam None

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

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

15. **IF** the leak is indicated on Raw Water Backup Cooling Header, **THEN** close **BOTH** of the following Raw Water Header Isolation Valves:

- HCV-2893
- HCV-2894

15.1 **IF** leak was **NOT** isolated, **THEN** open any or **ALL** of the following Raw Water Header Isolation Valves:

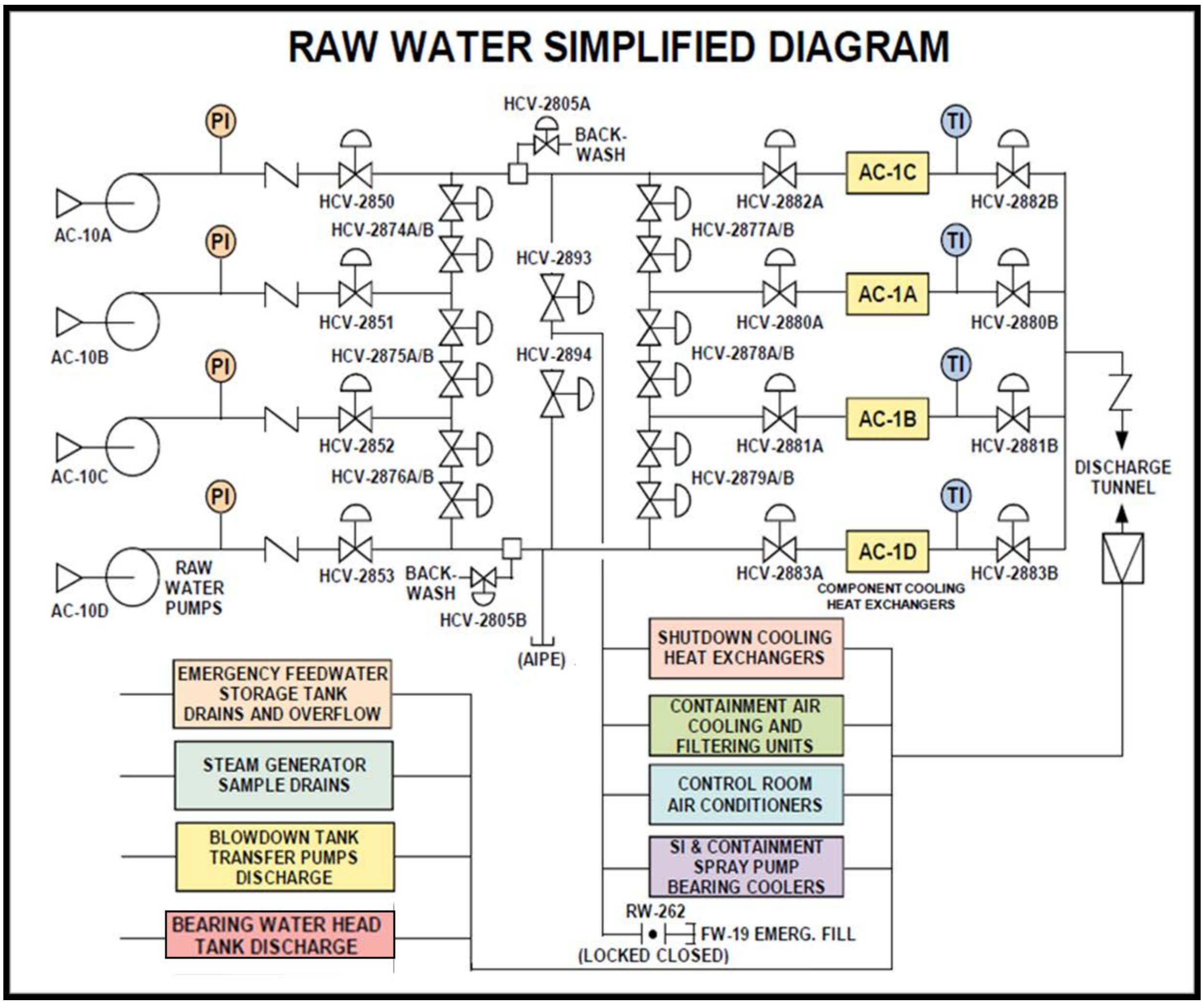
- HCV-2893
- HCV-2894

The determination as to whether certain valves (or components) when found inoperable should be considered as LCO entries for valves was based on the heavy load drop analysis, the Raw Water System calculation, the FCS Design Basis Documents, the Updated Safety Analysis Report (USAR), and Reference 6.

The basic criteria of 10 CFR 50 Appendix R is that cold shutdown must be achievable in 72 hours after a fire event in any fire area (Reference 7). Appendix R also requires that the safe shutdown analysis consider all equipment in a fire area to be destroyed and offsite power to be lost. However, it does not require the postulation of a single failure (Reference 8). The FCS appendix R safe shutdown analysis does not credit the use of CCW to achieve cold shutdown. A fire in Room 69 would "destroy" all three CCW pumps which necessitates the use of RW backup. Since RW backup was required for one event, it was simplest to not credit CCW for any event and always rely on RW backup for all fires.

To achieve cold shutdown in 72 hours post fire, a single flowpath of RW is adequate since there is no fire event identified which would disable a RW header. A single shutdown cooling heat exchanger is adequate since there is no fire event which would disable a shutdown cooling heat exchanger. A fire in an ESF room or DG room could disable one LPSI pump, therefore RW backup to both LPSI pumps must be available to preclude a complete loss of LPSI pumps for shutdown cooling (Reference 9).

(Slide #14)



EO 1.7 **(Slide #113)**

Emergency System Operation

AOP-30, Emergency Fill of the EFWST

NOTE: Use a current revision of AOP-30 for review.

The RW System is available as a back-up fill supply for the EFWST through two 50’ lengths of fire hose and 2 isolable fire hose connectors located in Room 81.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	1	
	K/A #	078 A3.01	
Level of Difficulty: 3	Importance Rating	3.1	

Instrument Air System: Ability to monitor automatic operation of the IAS, including: Air pressure.

Question: 27

PCV-1752, Air Dryers CA-31 & CA-12 Bypass Valve, automatically opens when ___(1)___ system pressure reaches a SETPOINT pressure of ___(2)___.

A. (1) Service Air
(2) 78 psig

B. (1) Service Air
(2) 96 psig

C. (1) Instrument Air
(2) 78 psig

D. (1) Instrument Air
(2) 96 psig

Answer: C

K/A Match:
Requires knowledge of the automatic operation of the Instrument Air System including sensing point and system pressure.

Explanation:

A. Incorrect. Plausible because the setpoint is correct but the sensing location is wrong.
B. Incorrect. Plausible because this is the setpoint for the Plant Air low pressure alarm but it is sensed from the Instrument Air System.
C. **Correct**. PCV-1752, Air Dryers CA-31 & CA-12 Bypass Valve, is opened when Instrument Air pressure reaches 78 psig as sensed by the Instrument Air System and is a backup to the closure of PCV-1753.
D. Incorrect. Plausible because the sensing location is correct. Incorrect because this is the setpoint for the Plant Air low pressure alarm as sensed from the Instrument Air System.

Technical Reference: LP 4-23-5, Slide #33, #108, & #161, Rev. 3
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 4-23-5, Compressed Air System-Auxiliary Operator nuclear That EO 1.8 - **EXPLAIN** the principles of abnormal operation of the Compressed Air System in terms of flowpaths, major parameters, (temperature, pressure, flowrate, etc.), alarms and control devices.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

EO *1.8 **(Slide #161)**

Abnormal System Operation

Loss of Instrument Air

If instrument air pressure is less than 78 psig, ensure PCV-1752, air dryer bypass valve, is open.

If valve is not open, then open the manual bypass valve CA-197.

EO 1.4b **(Slide #108)**

Major Component Description

Air Receivers (CA-3A/B)

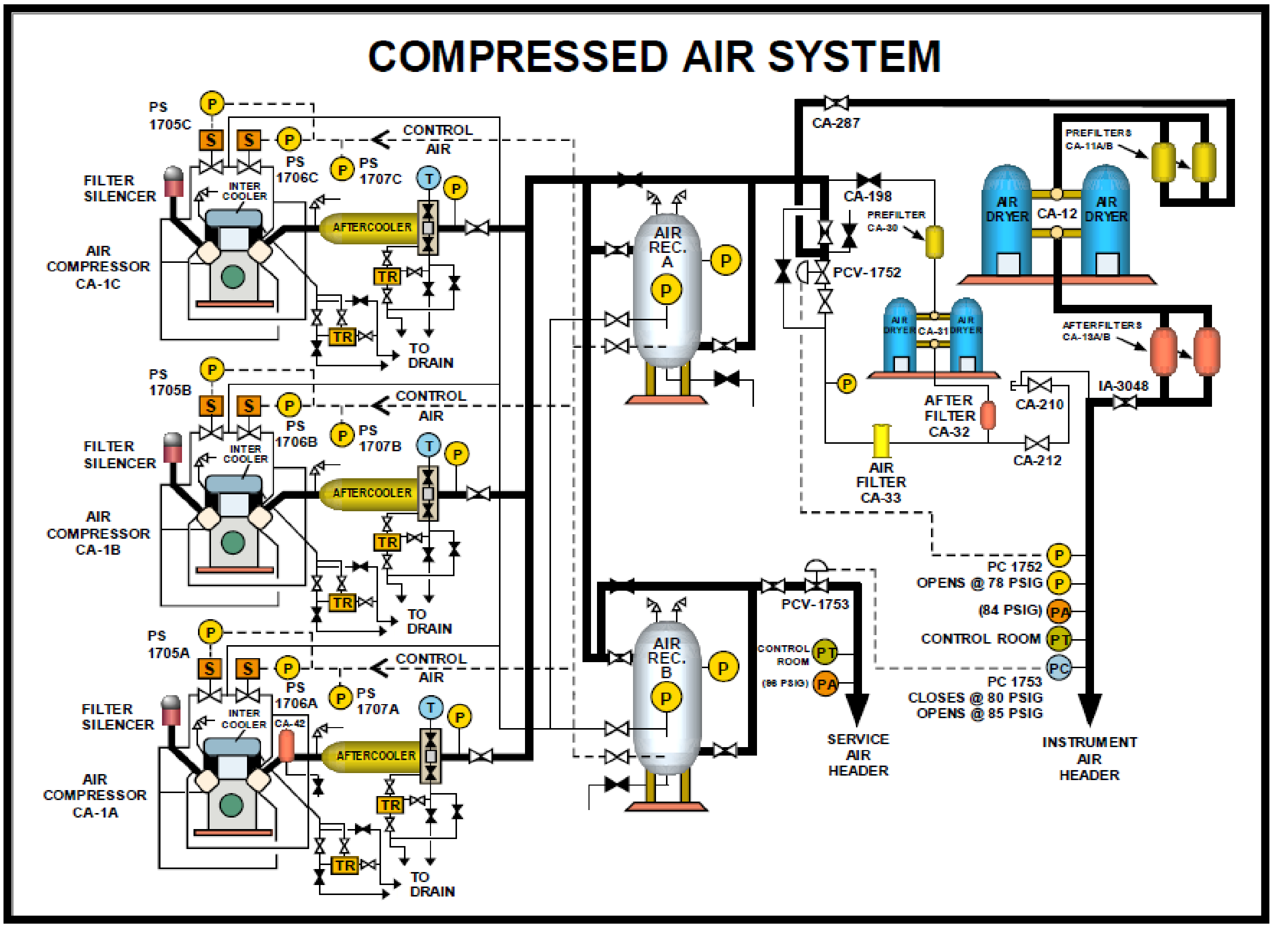
Alarms

CB-10/11, A21 in Control Room

“Plant Air Press Lo” is set at 96 psig (PA-1701)

“Instrument Air Press Lo” is set at 84 psig (PA-1751)

(Slide #33)


COMPRESSED AIR SYSTEM
PS 1705C
PS 1706C
PS 1707C
CONTROL AIR
FILTER SILENCER
INTER COOLER
AFTERCOOLER
AIR COMPRESSOR CA-1C
TO DRAIN
PS 1705B
PS 1706B
PS 1707B
AIR COMPRESSOR CA-1B
PS 1705A
PS 1706A
PS 1707A
CA-42
AIR COMPRESSOR CA-1A
AIR REC. A
AIR REC. B
CA-287
CA-198
PREFILTER CA-30
PCV-1752
PREFILTERS CA-11A/B
AIR DRYER
CA-12
AIR DRYER
CA-31
AFTERFILTERS CA-13A/B
AFTER FILTER CA-32
CA-210
CA-212
IA-3048
AIR FILTER CA-33
PCV-1753
CONTROL ROOM
(86 PSIG)
PC 1752 OPENS @ 78 PSIG
(84 PSIG)
CONTROL ROOM
PC 1753 CLOSES @ 80 PSIG OPENS @ 85 PSIG
SERVICE AIR HEADER
INSTRUMENT AIR HEADER

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	1	
	K/A #	103 A1.01	
Level of Difficulty: 3	Importance Rating	3.7	

Containment System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity.

Question: 28

Given the following conditions:

- Reactor Coolant System temperature is 532°F.
- VA-3A and VA-7C, Containment Vent Fans are in service.
- Containment air temperature is 122°F.
- Containment pressure is 0.9 psig.
- Raw Water/CCW heat exchangers AC-1B and AC-1C are in service.

Which of the following action(s) is procedurally directed to ensure Containment does NOT exceed design limits upon the occurrence of a Loss of Coolant Accident or a Steam Line Break inside Containment?

Reduce Containment…

A. …temperature by placing additional CCW Heat Exchangers in operation.

B. …temperature by placing additional Containment Vent Fans in operation.

C. …pressure by placing the Containment Purge Release System in operation.

D. …pressure by placing the Containment Pressure Relief System in operation.

Answer: B

K/A Match:

Applicant must have knowledge of pressure and temperature limits for Containment as well as the appropriate actions when the limits are exceeded.

Explanation:

A. Incorrect. Plausible because temperature is the concern and placing additional CCW Heat Exchangers in service would lower CCW temperature and indirectly lower Containment temperature. Incorrect because with only 2 of 4 Containment Vent Fans in service AOP-12, Step 15, refers to EOP/AOP Attachment CI-11, Containment Cooling System Operation, which places additional Containment Vent Fans in service.

B. **Correct**. Containment temperature exceeds the upper limit of 120°F. AOP-12, Loss of Containment Integrity, Step 15 is entered which refers to EOP/AOP Attachment CI-11, Containment Cooling System Operation. This places additional Containment Vent Fans in service.

C. Incorrect. Plausible because Containment pressure is higher than normal, however, the Containment Purge Release System is only permitted to be used when the plant is in Modes 4 or 5.

D. Incorrect. Plausible because Containment pressure is higher than normal and placing the Containment Pressure Relief System in operation would lower pressure but the concern is Containment temperature.

Technical Reference: OI-VA-1, Precautions 7, Rev. 85

(Attach if not previously provided including revision number) AOP-12, Entry Conditions H & I & Step 15, Rev. 8

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-8, Containment-Licensed Operator
EO 2.1 - **DESCRIBE** how Containment Integrity is monitored.
EO 2.3 - Briefly **DESCRIBE** actions necessary if Containment Integrity is violated as per AOP-12 and Technical Specification 2.6.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 9
55.43 ______

OI-VA-1, Precautions 7

7. Containment average temperature is to be maintained below 120°F during plant operations. If the ERF point "TAVCAN" is greater than 120°F, Tech Spec 2.01 should be entered until ERF point "TAVCAN" is less than 120°F.

AOP-12, Entry Conditions

ENTRY CONDITIONS

A loss of Containment Integrity has occurred which may be indicated by any of the following:

Non-automatic Containment Isolation Valves are open or blind flanges are not sealed as required for Containment Integrity.

The Equipment Hatch is not properly sealed.

Neither Personnel Air Lock Door is properly sealed.

Automatically operated Containment Isolation Valves are inoperable and not locked closed.

Containment building leakage rates have exceeded the allowable limits of Technical Specification 3.5, Containment Test.

The sealing mechanism associated with a Containment penetration (e.g., welds, bellows, or O-rings) are inoperable.

Noticeable air leakage from Containment.

Containment pressure is greater than 3 psig (PI-785, PI-786).

Containment temperature is greater than 120°F (TAVCAN, ERFCS).

AOP-12, Step 15

IF Containment temperature is greater than 120°F (TAVCAN, ERFCS),
THEN perform the following:

Enter Technical Specification 2.0.1, General Requirements.

Lower Containment temperature by maximizing Containment Cooling PER Attachment CI-11, Containment Cooling Operation.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	2	
	K/A #	002 K5.08	
Level of Difficulty: 3	Importance Rating	3.4	

Reactor Coolant System: Knowledge of the operational implications of the following concepts as they apply to the RCS: Why PZR level should be kept within the programmed band.

Question: 29

Which of the following is the reason Pressurizer level should be maintained within the programmed band in accordance with OI-RC-8, Reactor Coolant System Level Control Normal Operation, while at power?

A. Ensures Heaters remain covered during an outsurge and the PORVs don’t lift during an insurge.

B. Minimize insurge and outsurge cycles to protect the Pressurizer Heater sleeves.

C. Prevent a superheat condition in the steam space from repeated insurges and outsurges.

D. Allows sufficient volume for collection of non-condensable gases during continuous spray bypass flow.

Answer: A

K/A Match:
Identifies the operational implications associated with maintaining Pressurizer level within the programmed band.

Explanation:

A. **Correct**. Assures that the Heaters are not uncovered due to an outsurge following a 10% step decrease or 10% per minute ramp decrease in power. The steam volume is also large enough to accept a loss of load without the level reaching the PORVs or Safety Valves.

B. Incorrect. Plausible because this is described in SO-O-23, Standing Order for Systems and Equipment Usage Data (attached).

C. Incorrect. Plausible because superheat conditions can be created in the Pressurizer steam space from repeated insurges and outsurges but it’s not the reason for maintaining PZR level within program.

D. Incorrect. Plausible because non-condensable gases will collect in the steam space of the PZR but the programmed level is based on insurges and outsurges.

Technical Reference: LP 7-11-20,Slides #98, #242, & #382, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-11-20, Reactor Coolant System-Licensed Operator EO 1.6b - **LIST** the design parameters of the pressurizer and what the total volume is based on.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 3
55.43 ______

EO 1.6b **(Slide #98)**

Detailed Component Description

Pressurizer (RC-4)

Operating level (48-60%) is variable based on Tave (535-560ºF).

The total volume of the pressurizer is 900 ft^3.

It holds sufficient water volume necessary to prevent draining the pressurizer as the result of a reactor trip or an excess load incident.

Also, assures that the heaters are not uncovered due to an outsurge following a 10% step decrease or 10% per minute ramp decrease in power.

Sufficient water volume is maintained to accept level changes that result from load changes that result from load following transients without generating excessive waste and be compatible with CVCS flow capacity.

EO 1.6b **(Slide #99)**

EO 1.6b

Detailed Component Description

Pressurizer (RC-4)

The steam volume is large enough to yield acceptable pressure response during load change transients while minimizing the stored energy, in the form of hot water that could be released during a LOCA.

The steam volume is also large enough to accept a loss of load without the level reaching the PORVs or safety valves.

EO 4.3, 5.1b **(Slide #242)**

Detailed Component Description

Pressurizer (RC-4)

Pressurizer Temperature Instrumentation

Water space temperature (TE-108)

RTD supplies TI-108 indication on CB-3.

The RTD is located at the top of the heaters.

Range is 0-700ºF.

There are no alarm or control functions.

Steam temperature higher than water temperature may be caused by:

(a) Non-condensable gas

(b) Superheat (from repeated insurges and outsurges)

(c) Conduction from the hot metal mass

(Slide #382)

Attachment 7.21 - Pressurizer Insurge / Outsurge Significant Fatigue Cycles

1. REFERENCES

NED-DEN-05-0151 (AR# 35077 – 40)

2. LIMITS

Cooldown in progress:

- 450 cycles with ΔT of 50° - 250° F
- 50 cycles with ΔT of 251° - 300° F

Heatup in progress:

- 250 cycles with ΔT of 50° - 200° F
- 250 cycles with ΔT of 201° - 250° F

3. DATA

Cycle Number	____________________
Time / Date Cycle Initiated	__________/__________
Time / Date Cycle Completed	__________/__________
Reference Temperature (T144)	____________________
Peak Temperature (T144)	____________________
Cooldown or Heatup in progress (select one)	____________________

EO 1.6b **(Slide #382)**

Pressurizer Insurge/Outsurge Cycles

An insurge cycle is defined as T-144 lowering or raising at ≥50ºF at a rate of ≥100ºF/hr.

T-144 on the ERF will tell the operator the temperature of the liquid near the heater sleeves.

ARP-ERFCS provides guidance on the alarm and the recording requirements are found in SO-O-23.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 1	Group/Category #	2	
	K/A #	014 A2.04	
Level of Difficulty: 4	Importance Rating	3.4	

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

Question: 30

Given the following conditions:

- While performing OI-RR-1, Reactor Regulating System Normal Operation, Attachment 4, Axial Shape Index Control, Regulating Group 4 CEA #1 is at 120 inches withdrawn.
- All other Regulating Group 4 rods are positioned at 126 inches.

(1) Which of the following is the result of this alignment for Group 4, and
(2) What action is required per OI-RR-1, Reactor Regulating System Normal Operation?

A. (1) A Rod Position Deviation Alarm.
(2) Contact Reactor Engineering and restore CEA #1 to within 2 inches of Regulating Group 4.

B. (1) A Rod Position Deviation Alarm.
(2) Reduce Reactor power to less than 70%, then restore CEA #1 to within 2 inches of Regulating Group 4.

C. (1) A Rod Block Alarm.
(2) Reduce Reactor power to less than 70%, then restore CEA #1 to within 2 inches of Regulating Group 4.

D. (1) A Rod Block Alarm.
(2) Contact Reactor Engineering and restore CEA #1 to within 2 inches of Regulating Group 4.

Answer: A

K/A Match:

Applicant must know the cause of the yellow DEV alarm block and the Precautions contained in OI-RR-1, Reactor Regulating System Normal Operation.

Explanation:

A. **Correct**. When SCEAPIS detects a CEA deviation within the group greater than 4" a yellow DEV alarm block will appear. When SCEAPIS detects a CEA deviation within the group of 8" a magenta DEV alarm block will appear and a rod block alarm is received. Per the Precaution 14 in OI-RR-1, a CEA misaligned from others in its group by more than four inches shall not be realigned until the Reactor Engineer or designated alternate has been consulted.

B. Incorrect. Plausible because the Rod Position Deviation alarm is correct and per Precaution 12 in OI-RR-1, CEAs within a group should be kept within two inches of each other under normal circumstances. Additionally, Precaution 13 states that a deviation of greater than 4 inches should be realigned as soon as possible but RE must be contacted before realigning. If the CEA was greater than 12" but less than 18" misaligned, the correct action per AOP-02 is to reduce power.

C. Incorrect. Plausible if thought that immediate restoration was required but that is not consistent with the precaution in OI-RR-1. An 8 inch deviation on the group creates a “magenta” DEV alarm on SCEAPIS and a rod block. If the CEA was greater than 12" but less than 18" misaligned, the correct action per AOP-02 is to reduce power.

D. Incorrect. Plausible because RE must be contacted before realigning. An 8 inch deviation on the group creates a “magenta” DEV alarm on SCEAPIS and rod block.

Technical Reference:	OI-RR-1, Precautions 12, 13, & 14, Rev. 32
(Attach if not previously provided including revision number)	LP 7-12-26, Slides #98 & #100, Rev. 2
	AOP-02, Section III, NOTES 4, Rev. 10a

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-26, Control Rod Drive System-Licensed Operator
EO 1.7 - **DESCRIBE** the methods of control Rod position indication. Include the readouts and displays associated with each method.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	6
	55.43	

OI-RR-1, Precautions 12, 13, & 14

PRECAUTIONS (continued)

10. During all CEA motion all available indication including: Reactor Power (% Power and/or Counts per minute), Start-up Rate and RCS temperature shall be observed for response.

11. Continuous CEA motion shall be avoided whenever possible. CEA motion should be stopped at least every 33 inches (43 seconds of continuous CEA motion) to check position of CEAs in the group and Reactor response.

12. CEAs within a group should be kept within two inches of each other under normal circumstances.

13. A CEA misaligned from others in its group by four inches or less should be realigned as soon as possible.

14. A CEA misaligned from others in its group by more than four inches shall not be realigned until the Reactor Engineer or designated alternate has been consulted.

OI-RR-1, Precautions 12, 13, & 14

PRECAUTIONS (continued)

15. During all CEA motion all available indication including: Reactor Power (% Power and/or Counts per minute), Start-up Rate and RCS temperature shall be observed for response.

16. Continuous CEA motion shall be avoided whenever possible. CEA motion should be stopped at least every 33 inches (43 seconds of continuous CEA motion) to check position of CEAs in the group and Reactor response.

17. CEAs within a group should be kept within two inches of each other under normal circumstances.

18. A CEA misaligned from others in its group by four inches or less should be realigned as soon as possible.

19. A CEA misaligned from others in its group by more than four inches shall not be realigned until the Reactor Engineer or designated alternate has been consulted.

AOP-02, Section III, NOTES 4

B. **NOTES**

1. Technical Specification 2.10.2, Reactivity Control Systems and Core Physics Parameters Limits, addresses reactivity control system limits.

2. Excessive Linear Heat Rate may be indicated by Incore alarms on the ERF. Technical Specification 2.10.4, Power Distribution Limits, applies to linear heat rate limitations.

3. A full length Shutdown or Regulating CEA misaligned by more than 18 inches requires a power reduction to less than 70% ΔT Power within one hour.

4. A full length Shutdown or Regulating CEA misaligned by more than 12 inches but less than 18 inches requires that if the CEA can not be restored to within 12 inches of all other CEAs in its group within one hour, then the CEA must be declared inoperable.

EO *1.7 **(Slide #98)**

Major Component Description

Secondary CEA Position Indication System (SCEAPIS) – DCS

SCEAPIS Flat Panel Touch Monitors (DCS Screens)

Individual Groups (Regulating Group 4 shown)

If a rod position deviates 5” from other rods in the group, a yellow DEV alarm block appears in the lower left corner of the page.

EO *1.7 **(Slide #100)**

Major Component Description

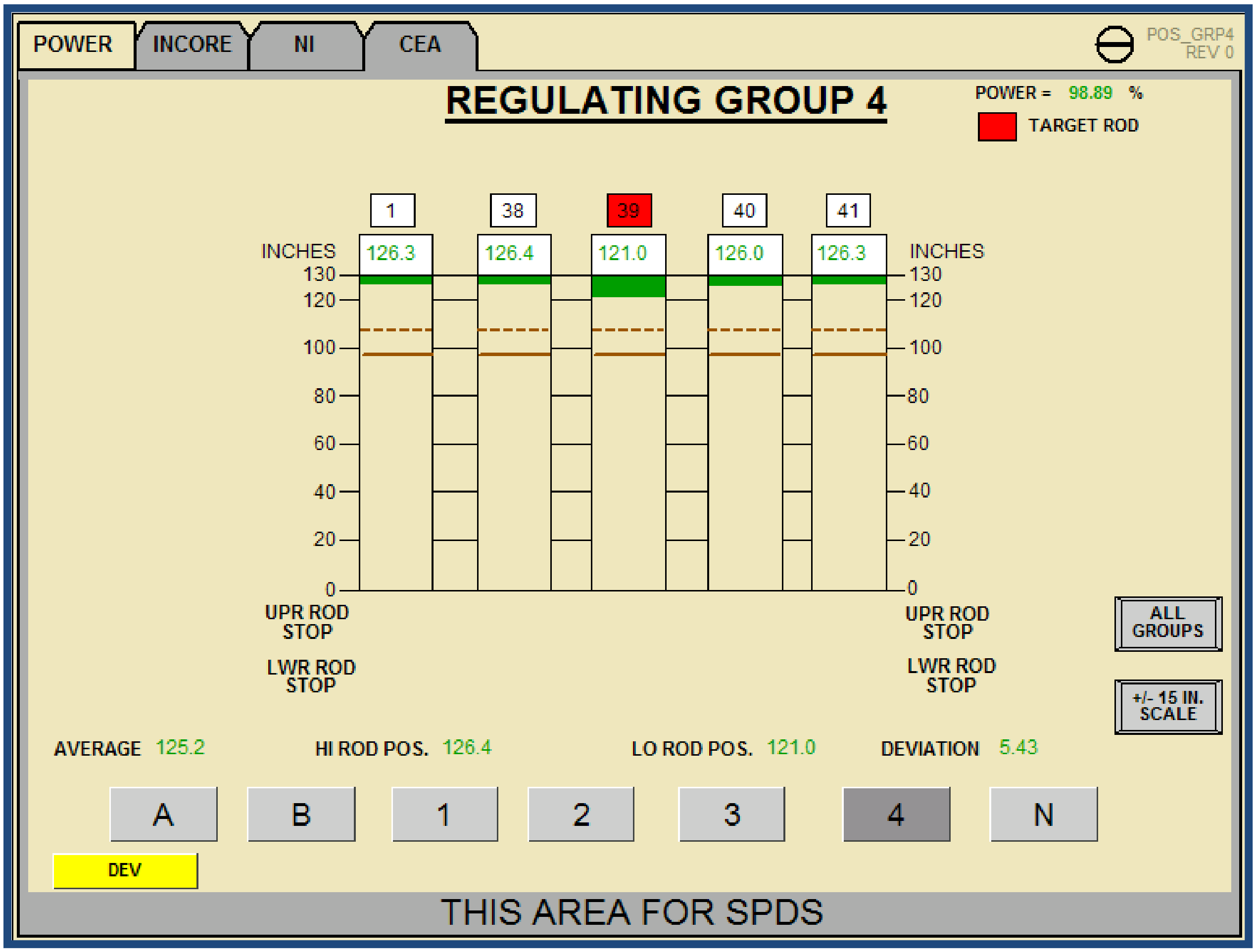
Secondary CEA Position Indication System (SCEAPIS) – DCS

SCEAPIS Flat Panel Touch Monitors (DCS Screens)

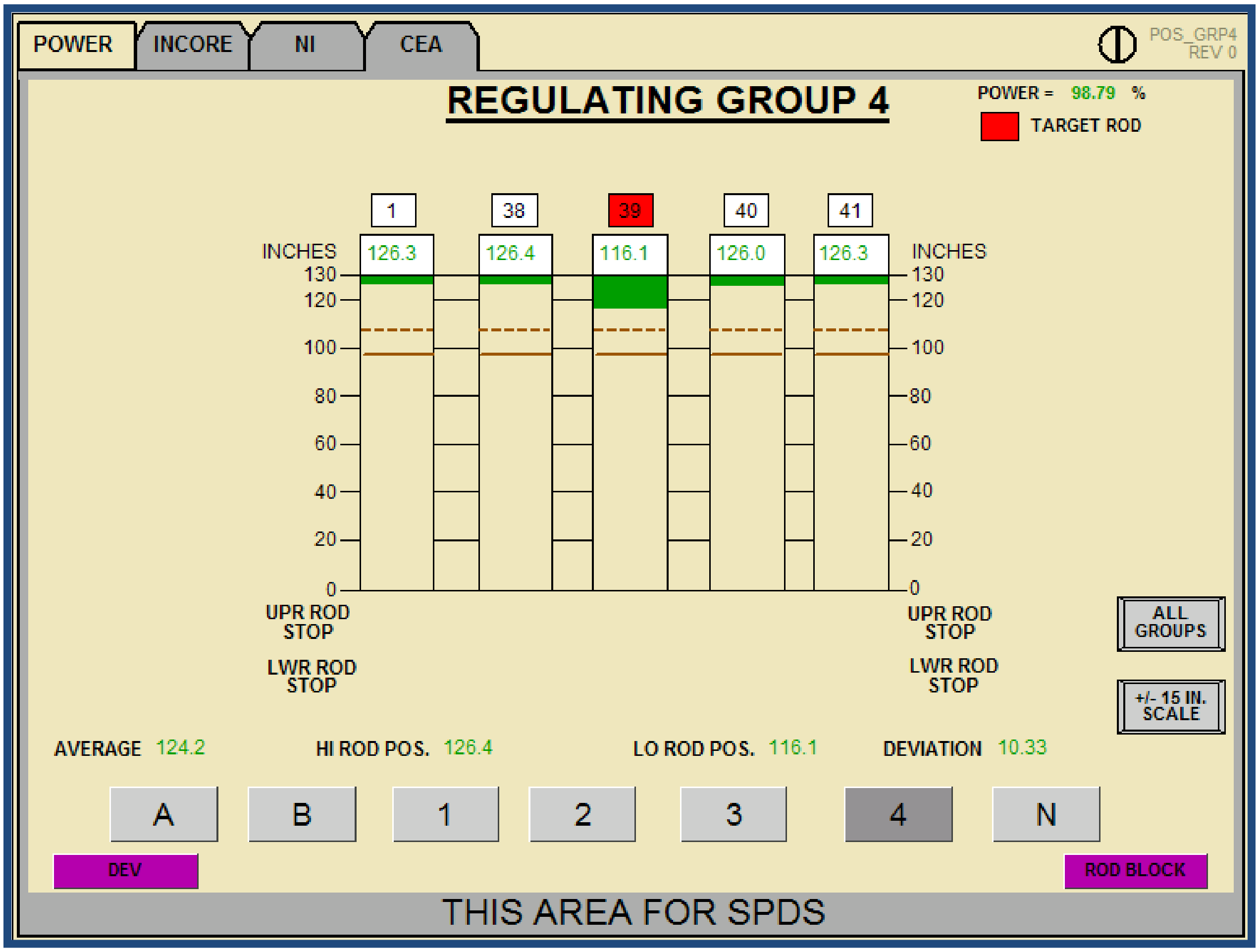
Individual Groups (Regulating Group 4 shown)

If a rod position deviates 8” from other rods in the group, a magenta DEV alarm block appears in the lower left corner of the page along with a magenta ROD BLOCK alarm block on the lower right hand corner.

(Slide #98)


POWER
INCORE
NI
CEA
POS_GRP4
REV 0
REGULATING GROUP 4
POWER = 98.89 %
TARGET ROD
1
38
39
40
41
INCHES
126.3
126.4
121.0
126.0
126.3
INCHES
130
120
100
80
60
40
20
0
UPR ROD STOP
LWR ROD STOP
UPR ROD STOP
LWR ROD STOP
ALL GROUPS
+/- 15 IN. SCALE
AVERAGE 125.2
HI ROD POS. 126.4
LO ROD POS. 121.0
DEVIATION 5.43
A
B
1
2
3
4
N
DEV
THIS AREA FOR SPDS

(Slide # 100)


POWER
INCORE
NI
CEA
POS_GRP4
REV 0
REGULATING GROUP 4
POWER = 98.79 %
TARGET ROD
1
38
39
40
41
INCHES
126.3
126.4
116.1
126.0
126.3
INCHES
130
120
100
80
60
40
20
0
UPR ROD STOP
LWR ROD STOP
UPR ROD STOP
LWR ROD STOP
ALL GROUPS
+/- 15 IN. SCALE
AVERAGE 124.2
HI ROD POS. 126.4
LO ROD POS. 116.1
DEVIATION 10.33
A
B
1
2
3
4
N
DEV
ROD BLOCK
THIS AREA FOR SPDS

ˆö

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/13/15	Tier #	2	
Change: 2	Group/Category #	2	
	K/A #	015 K6.01	
Level of Difficulty: 3	Importance Rating	2.9	

Nuclear Instrumentation System: Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Sensors, detectors, and circuits.

Question: 31

If calibrated at 100% power, when lowering power, the Power Range Nuclear Instruments (PRNI) decalibrate and indicate _____(1)_____ than actual power.

If calibrated at 10% power, when raising power, the PRNI decalibrate and indicate _____(2)_____ than actual power.

A. (1) lower
(2) lower

B. (1) lower
(2) higher

C. (1) higher
(2) lower

D. (1) higher
(2) higher

Answer: B

K/A Match:

Applicant must be aware of effects on Power Range Nuclear Instruments including the effect on plant indication when power level changes.

Explanation:

A. Incorrect. Plausible because the Power Range Nuclear Instruments read lower than actual when power is lowered due to the RCS temperature program. Incorrect because the opposite occurs as power is raised.

B. **Correct**. When lowering power the Power Range Nuclear Instruments read lower than actual due to the RCS temperature program. The opposite occurs when power is raised. This is caused by a change in core fluid density as power is raised and lowered.

C. Incorrect. Plausible as this is the opposite of what occurs when power is raised and lowered. Incorrect because the decalibration occurs as a function of core fluid density.

D. Incorrect. Plausible because the Power Range Nuclear Instruments read higher than actual when power is raised due to the RCS temperature program. Incorrect because the opposite occurs as power is lowered.

Technical Reference: OP-4, Attachment 1, Step 3 CAUTION, Rev. 51
(Attach if not previously provided including revision number) OP-4, Attachment 2, Step 5 CAUTION, Rev. 51

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-18, Nuclear Instrumentation System-Licensed Operator
EO 1.8 - **EXPLAIN** the indications available for monitoring the operation of the Power Range NI System.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	6
	55.43	

OP-4, Attachment 1, Step 3 CAUTION

CAUTION

While raising power, NIs decalibrate and indicate higher than actual. Use NIs conservatively during the approach to the desired power level.

3. Raise Reactor Power while performing the following steps:

   a. Raise Turbine Generator Load to maintain RCS T_{AVE} program per TDB Figure III.1.

OP-4, Attachment 2, Step 5 CAUTION

CAUTION

While lowering power, NIs decalibrate and indicate lower than actual power.

5. Lower Reactor Power while performing the following steps:

   a. Lower Turbine Generator Load to maintain RCS T_{AVE} program per TDB Figure III.1.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	2	
	K/A #	028 K5.03	
Level of Difficulty: 4	Importance Rating	2.9	

Hydrogen Recombiner and Purge Control System: Knowledge of the operational implications of the following concepts as they apply to the HRPS: Sources of hydrogen within containment.

Question: 32

Given the following:

- A Loss of Coolant Accident and Main Steam Line Break inside Containment have occurred.
- The core remains covered.

(1) Which of the following is the largest contributor to Hydrogen gas concentration in Containment, and

(2) What is the operational implication of these Containment conditions?

A. (1) Zinc-Water Reaction inside Containment.
(2) The Hydrogen Analyzer reads lower than actual.

B. (1) Zinc-Water Reaction inside Containment.
(2) The Hydrogen Analyzer reads higher than actual.

C. (1) Aluminum-Water Reaction inside Containment.
(2) The Hydrogen Analyzer reads lower than actual.

D. (1) Aluminum-Water Reaction inside Containment.
(2) The Hydrogen Analyzer reads higher than actual.

Answer: D

K/A Match:

Applicant must know the major source of hydrogen in Containment following a design basis LOCA and the operational implications of measuring hydrogen concentration following the accident.

Explanation:

A. Incorrect. Plausible because the Zinc-Water will contribute hydrogen to the Containment atmosphere but it is not the major source during a containment spray event. Incorrect because the Hydrogen Analyzer will be caused to read higher than actual due to the high containment humidity.

B. Incorrect. Plausible because the Zinc-Water will contribute hydrogen to the Containment atmosphere but it is not the major source during a containment spray event. The Hydrogen Analyzer response to high containment humidity is correct.

C. Incorrect. Plausible because Aluminum-Water corrosion inside Containment is the major contributor but high humidity causes the Hydrogen Analyzer to read higher than actual per OI-VA-1.

D. **Correct**. Aluminum-Water Corrosion inside Containment is the major contributor to hydrogen following a DBA. OI-VA-6 identifies the operational implication of using the Hydrogen Analyzer in a high humidity, i.e., post LOCA condition.

Technical Reference: OI-VA-6, Step 17 NOTE, Rev. 17
(Attach if not previously provided including revision number) LP 7-14-3, Slide #74, Rev. 1
LP 7-15-28, Page 73, Rev. 7

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:
Lesson Plan 7-14-3, Containment Hydrogen Purge System-Licensed Operator
EO 1.4 - **STATE** the function of each major component of the Containment Hydrogen Purge System.
EO 2.1 - **EXPLAIN** how the procedure is used to obtain accurate % H_2 levels under conditions of 100% relative humidity.

Lesson Plan 7-15-28, Mitigating Core Damage-Licensed Operator
EO 1.13 - **EXPLAIN** the generation of Hydrogen in an accident scenario.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 5
55.43 ____

NOTES

1. Stabilization of the % H_2 meter could take between 5 and 45 minutes.

2. The Dual-Range selector may be turned to the 0-20% scale for a more accurate reading if the % H_2 meter reads above 10%. The ERF and the Control Room Recorders, HR-81A and HR-81B, will read one half of actual % H_2 when placed in the 0-20% range.

3. The readings given by the Hydrogen Analyzers will be higher than the actual value and the true Hydrogen concentration must be calculated when the Containment atmosphere reaches 100% relative humidity such as after a Loss of Coolant Accident inside Containment. For all other cases, the indicated value is accepted as the true Hydrogen concentration.

4. The % Error obtained from Figure 1 for the corresponding Containment temperature should be subtracted from the value indicated by the control panel in order to obtain the true value as follows:

   <u>EXAMPLE:</u> Containment Temperature = 200°F
   Containment Relative Humidity = 100%
   Indicated Hydrogen Concentration = 20%

   Actual H_2 = Indicated H_2 - (Indicated H_2 x % ERROR)

   Where:

Indicated H_2	=	The indicated Hydrogen Concentration on AI-65A/65B
% Error	=	The % Error obtained from Figure 1 for a Containment Temperature of 200°F and a Relative Humidity of 100%
Actual H2	=	20% - (20% x 0.3) = 20% - 6%
Actual H2	=	14%

17. WHEN the selected analyzer % H_2 meter stabilizes,
    THEN the sample is complete:

    - VA-81A
    - VA-81B

EO *2.1 **(Slide #74)**

Procedures

Operate the Hydrogen Analyzer

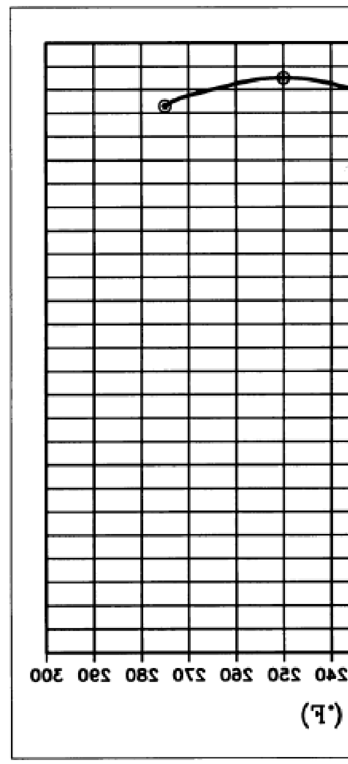
OI-VA-6, Containment Hydrogen Analyzer Operation, Figure 1, Containment Temperature Vs % Error in Hydrogen Reading, is used to in conjunction with the procedure to calculate the % error during 100% relative humidity conditions in containment such as after a LOCA.

INSTRUCTOR NOTE: Use OI-VA-6 and Figure 1 to calculate the actual %H_2 under 100% humidity conditions with an indicated H_2 of 4% and a containment temperature of 200ºF.

By using the % error graph for the corresponding containment temperature, multiply the % error by the reading on the %H_2 meter to obtain the error factor.

By subtracting this value from the reading on the meter, the actual %H_2 level will be obtained.

Figure 1 - Containment Temperature Vs % Error in Hydrogen Reading



LP 7-15-28, Page 73-75

4. Other Sources of Hydrogen

a. During containment spray operation, another means of hydrogen gas production exists. At Fort Calhoun, another reaction with aluminum which can liberate hydrogen is:

$$2Al + 3H_20 \backslash Al_20_3 + 3H_2$$

b. The major source of aluminum for this reaction at Fort Calhoun is the cooling coils and fins in the Containment Cooling and Filtering units. In fact, a March 1988 report by CE re-evaluated the containment post- accident pressure peak for Fort Calhoun based on the additional hydrogen produced by aluminum and another source we shall discuss shortly--zinc. This added production source of hydrogen changed the containment purging logistics to accommodate the increased production without an uncontrolled containment failure.

The rate of this production reactor varies strongly as a function of temperature and water vapor pH. The reaction rate in increased by increasing either temperature or pH. Calculations show that, if the entire inventory of aluminum in containment were to react with the containment spray, 149,571 SCF of free H_2 will be produced. That is about 14% of the net free volume of containment. The aluminum can produce 2,813 SCF of H_2 per day during accident conditions.

c. In addition, hydrogen at Fort Calhoun Station will be liberated by means of zinc-water reactions following a loss of reactor coolant within the containment. The typical reaction which takes place under these circumstances is:

$$Zn + H_2O \backslash ZnO + H_2$$

d. The zinc metal sources in the Fort Calhoun containment are:

- Zinc-based paint on the

Containment Liner, tanks and platforms

- Galvanized steel
- Platform and stair gratings
- Electrical conduit and cable trays
- Ventilation ducts and housings

e. The hydrogen generation rate does vary as a function of temperature for the zinc-water reaction. For example, considering some typical postulated temperatures in containment following a loss of coolant (LOCA) accident, the following reaction rates for the zinc-water reaction could be observed:

For zinc-based paint:

H_2 generation rate = 4.678 x

10^5 $e^{-(14,500/RT)}$ SCF/ft^2-hr

For galvanized steel:

H_2 generation rate = 1.3 x

10^5 $e^{-(14,500/RT)}$ SCF/ft^2-hr

Where:

R = the gas constant for Hydrogen = 1.986 cal/gm-mole °K, and

T = temperature in degrees Kelvin.

f. Calculations show that the zinc-base paint can produce 317.6 SCF of H_2 per day during accident conditions while the galvanized steel can contribute 130.7 SCF of H_2. That's a total contribution of 448.3 SCF of H_2 per day from the zinc reaction during accident conditions.

Bank Question:

Why must the readings obtained from the Containment Hydrogen Analyzers be corrected for high humidity?

A. High humidity causes damage to the Hydrogen Analyzers.

B. High humidity causes the Hydrogen Analyzers to read higher than actual.

C. High humidity causes the Hydrogen Analyzers to read lower than actual.

D. High humidity has no effect on Hydrogen Analyzer operation, but does reduce the life expectancy of the analyzers.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/15/15	Tier #	2	
Change: 0	Group/Category #	2	
	K/A #	033 A1.01	
Level of Difficulty: 3	Importance Rating	2.7	

Spent Fuel Pool System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Spent Fuel Pool Cooling System controls including: Spent fuel pool water level.

Question: 33

Given the following conditions:

- Refueling Operations are in progress.
- The Fuel Transfer Canal Gate Valve is OPEN.
- LI-2846, Spent Fuel Pool Level Indicator is in service.

Which of the following lists the indications for a Spent Fuel Pool (SFP) leak?

Low level is indicated by...

A. Spent Fuel Pool: lowering level, no alarm received.
Refueling Cavity: lowering level, Low Level Alarm received.
ONLY Spent Regen Tank level rises.

B. Spent Fuel Pool: lowering level, Low Level Alarm received.
Refueling Cavity: lowering level, no alarm received.
ONLY Spent Regen Tank level rises.

C. Spent Fuel Pool: lowering level, Low Level Alarm received.
Refueling Cavity: lowering level, no alarm received.
Containment Sump AND Spent Regen Tank levels rise.

D. Spent Fuel Pool: lowering level, Low Level Alarm received.
Refueling Cavity: lowering level, Low Level Alarm received.
Containment Sump AND Spent Regen Tank levels rise.

Answer: B

K/A Match:
Applicant must be aware of alarms and indications affecting Spent Fuel Pool level especially when cross connected with the Containment.

Explanation:

A. Incorrect. Plausible because the Spent Regen Tank level increase is correct, however, both the Refueling and Spent Fuel Pools will have low level alarms.

B. **Correct**. These are the correct alarm and indications for a leak with the conditions listed. SPENT FUEL POOL LEVEL LO will alarm at 39.1 ft. RCS REFUELING LEVEL LO will not alarm until level is at < 14 inches above the bottom of the Hot Leg (level would need to lower an additional 21 feet).

C. Incorrect. Plausible because SPENT FUEL POOL LEVEL LO will alarm at 39.1 ft. incorrect because only the Spent Regen Tank level will rise and the Refueling Cavity will lower.

D. Incorrect. Plausible because the Spent Fuel Pool level will alarm. Incorrect because the Containment Sump level will not rise.

Technical Reference: ARP-CB-1/2/3/A1, Window D-3U, Rev. 37
(Attach if not previously provided including revision number) ARP-CB-1/2/3/A4, Window D-3, Rev. 35

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-36, Loss of Spent Fuel Cooling-Licensed Operator
EO 1.2 - **DESCRIBE** how the plant response to a loss of spent fuel pool cooling.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Panel: **CB-1/2/3**	Annunciator: **A1**	Window: **D-3U**

SPENT FUEL POOL WATER LEVEL HIGH OR LOW — Page 1 of 2

SAFETY RELATED

SPENT FUEL POOL LEVEL HI OR LO

Tech Spec References: 2.8.3(2)

Setpoint (42.3 ft)

Initiating Device LC-2846 Setpoint (39.1 ft) Power AI-41B

OPERATOR ACTIONS

1. Determine Spent Fuel Pool level by visual inspection.

2. IF SFP cooling is in service per OI-SFP-7, Temporary Spent Fuel Pool Cooling via Chillers, visually inspect the Primary System for evidence of leakage.

   2.1 IF leakage exists, THEN reference OI-SFP-7 Emergency Operating Guidelines.

3. IF SFP level is below the upper suction strainer, THEN align the SFP Cooling System for recirculation using the lower suction per OI-SFP-1.

(continue)

PROBABLE CAUSES

Low level:
- Evaporation
- Leakage or break in the Spent Fuel Pool Cooling System
- Spent Fuel Pool liner leak

High level:
- SFP heatup

Panel: **CB-1/2/3**	Annunciator: **A4**	Window: **D-3**

REACTOR VESSEL LOW WATER LEVEL

SAFETY RELATED

RCS REFUELING LEVEL LO

Tech Spec References: 2.8.1(4)

Initiating Device LIS-119 Power AI-42A
Initiating Device LA-197 Setpoint<14 inches above Bottom of Hot Leg Power PQ-3/AI-42B

OPERATOR ACTIONS

1. Verify RCS Level on LIS-119 and/or LI-197.

2. IF RCS level is low, THEN monitor Shutdown Cooling Pump amps for indication of loss of suction.

3. IF level decrease is not controlled or SDC pumps indicate loss of suction, THEN GO TO AOP-19.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	2	
Change: 2	Group/Category #	2	
	K/A #	035 K6.02	
Level of Difficulty: 3	Importance Rating	3.1	

Steam Generator System: Knowledge of the effect of a loss or malfunction on the following will have on the SGS: Secondary PORV.

Question: 34

Given the following conditions:

- While at 100% power MS-291, Air Assisted Main Steam Safety Valve developed a leak and was gagged closed.
- An ensuing Loss of Load causes Steam Generator pressures as indicated on PI-902 and PI-905 to spike to 1027 psia.

Assuming all valves are set to their design setpoints, the total number of Main Steam Safety Valves that should have lifted as a result of this transient is…

A. …3.

B. …5.

C. …7.

D. …9.

Answer: B

K/A Match:
Applicant must know relationship between the lift pressure of the Air Assisted Main Steam Safety Valves and lift setpoints of the Secondary Safety Valves.

Explanation:

A. Incorrect. Plausible because the applicant should know that the Secondary Safety Valves operate in pairs. With one valve at the lowest setpoint gagged, an odd number of valves will lift.

B. **Correct**. 5 Secondary Safety Valves will lift if pressure spikes to 1027 psia with MS-291 out of service. 1 at 1000 psia; 2 at 1015 psia; and 2 at 1025 psia = 5 Safeties lifted.

C. Incorrect. Plausible if thought that setpoints to Secondary Safety Valves were set to lower values.

D. Incorrect. Plausible if thought that 1015 psia was the peak pressure for all Secondary Safety Valves to lift.

Technical Reference: LP 7-11-17,Slide #55, Rev. 2

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-17, Main Steam System-Licensed Operator
EO 1.4 - **EXPLAIN** the indications, automatic actions, operating logic, alarm setpoints, interlocks and permissives associated with the Main Steam Instrumentation System.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 14
55.43 ____

EO 1.6 **(Slide #55)**

Major Component Description

Secondary Safety Valves

Lift setpoints are staggered to permit only two valves to lift simultaneously thus minimizing unnecessary loss of steam while still providing adequate protection.

(a) 985 psig, MS-291 and MS-292

(b) 1000 psig, MS-275 and MS-279

(c) 1010 psig, MS-276 and MS-280

(d) 1025 psig, MS-277 and MS-281

(e) 1035 psig, MS-278 and MS-282

<u>Bank Question</u>:

While at 100% power MS-291 develops a leak and must be gagged [preventing it from opening]. An ensuing loss of load causes secondary pressure as indicated on PR-1046 and PR-1048 to spike to 1043 psia. How many secondary safety valves should have lifted as a result of this transient. [All valves set to design setpoint.]

A. 6

<u>B. 7</u>

C. 8

D. 9

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	2	
	K/A #	041 K1.06	
Level of Difficulty: 2	Importance Rating	2.6	

Steam Dump/Turbine Bypass Control System: Knowledge of the physical connections and/or cause-effect relationships between the SDS and the following systems: Condenser.

Question: 35

The setpoint at which an interlock closes the _____(1)_____ is Condenser Vacuum at _____(2)_____.

A. (1) Main Steam Isolation Valves
(2) 19 inches Hg

B. (1) Main Steam Isolation Valves
(2) 21.85 inches Hg

C. (1) Steam Dump and Bypass Valves
(2) 19 inches Hg

D. (1) Steam Dump and Bypass Valves
(2) 21.85 inches Hg

Answer: C

K/A Match:
Applicant must know the effect of Condenser pressure on the Steam Dump and Bypass Valves.

Explanation:

A. Incorrect. Plausible because the Steam Dump and Bypass Valves are disabled when Condenser Vacuum is 19" Hg. The Main Steam Isolation Valves (MSIVs) do not close on low vacuum.
B. Incorrect. Plausible because the Turbine will trip at 21.85 inches Hg Condenser Vacuum. The MSIVs do not close on low vacuum.
C. **Correct**. Steam Dump and Bypass Valves will close when Condenser Vacuum degrades to 19 inches Hg.
D. Incorrect. Plausible because the Turbine will trip at 21.85 inches Hg Condenser Vacuum and the Steam Dump and Bypass Valves are impacted. Incorrect because 21.85 inches Hg is the Turbine Trip setpoint.

Technical Reference: LP 07-11-17, Slide #106, Rev. 2
(Attach if not previously provided including revision number) LP 07-11-5, Slide #35, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-11-17, Main Steam System-Licensed Operator EO 1.2 - **EXPLAIN** the controls and indications associated with a Main Steam System equipment manipulated from the Control Room.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

EO 1.2 **(Slide #106)**

Major Component Description

Steam Dump and Turbine Bypass Valves (TCV-909-1/2/3/4 and PCV-910)

Trip Response Mode

Condenser vacuum must be >19” Hg in order to allow opening the steam dump and bypass valves (protects the condenser from overpressurizing).

DCS will alarm and show a LOW VACUUM window when condenser vacuum is <19” Hg.

EO 1.2 **(Slide #35)**

Major Component Description

Condenser Evacuation Pumps (FW-8A/B/C)

<u>NOTE</u>: Use a current revision of the ARPs to review operator actions.

If condenser vacuum decreases to 21.35 inches of mercury, as sensed by PT-5048-1/2/3(5049-1/2/3), DCS annunciates an EXHAUST HOOD VACUUM LO alarm in the Control Room on panel CB-10 (A9) and a turbine trip is initiated.

At 19 inches of mercury a Steam Dump and Bypass System inhibit initiates preventing steam from being dumped to the condenser with insufficient vacuum.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	2	
	K/A #	045 A4.02	
Level of Difficulty: 2	Importance Rating	2.7	

Main Turbine Generator System: Ability to manually operate and/or monitor in the control room: T/G controls, including breakers.

Question: 36

Which of the following is a Prerequisite to operate the Main Disconnect Switch (DS-T1) electrically, either locally or from the Main Control Board?

A. The Kirk Key interlock must be satisfied.

B. All Turbine Control Valves must be closed.

C. A Bus Duct Cooling Fan must be operating.

D. Only one of the Main Output Breakers, 3451-4 or 3451-5, must be open.

Answer: A

K/A Match:
Applicant must know the prerequisites to operating the Main Disconnect Switch which is manipulated during normal operations as well as in an emergency (back feeding power from the 345 KV System to the 4160 V Buses during a Station Blackout or Loss of Offsite Power).

Explanation:

A. **Correct**. Operation of the key is necessary to prevent motor operation and withdraws a locking bolt that releases the motor-manual handle.
B. Incorrect. Plausible because Turbine Stop Valves must be closed not the Turbine Control Valves.
C. Incorrect. Plausible if thought that cooling flow should be applied to the Isophase Duct but the fans must be off.
D. Incorrect. Plausible because both Main Output Breakers 3451-4 and 3451-5 must be open.

Technical Reference: LP 7-13-6, Slides #91 & #95, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-6, Main Generator-Licensed Operator
EO 1.4 - **EXPLAIN** the interlocks associated with DS-T1.

Question Source:	Bank #	X	
	Modified Bank #		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam	None	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

EO *1.4 **(Slide #91)**

Major Component Description

Main Disconnect Switch (DS-T1)

Operating Mechanism

The following interlocks must be met for electrical operation (remote or local) of DS-T1:

(1) The key interlock switch by inserting the key (obtained from the Turbine Building Operator key-ring or from CB-20) in the local key switch.

(2) Breakers 3451-4 and 3451-5 are open.

(3) All four (4) turbine stop valves are shut.

(4) The 22KV Bus is de-energized (as seen by an undervoltage relay).

(5) The generator field breaker is open.

(6) 4160V breakers 1A11, 1A13, 1A22 and 1A24 are open.

(7) Isolated Phase Bus Duct Cooling Unit(s) are off.

(8) The operating motor overload contact is closed.

(9) The 69 permissive is satisfied (operator lined up for electrical operation).

EO 1.3 **(Slide #95)**

Major Component Description

Main Disconnect Switch (DS-T1)

Operating Mechanism

Shifting from electrical to manual operation requires use of a Kirk Key.

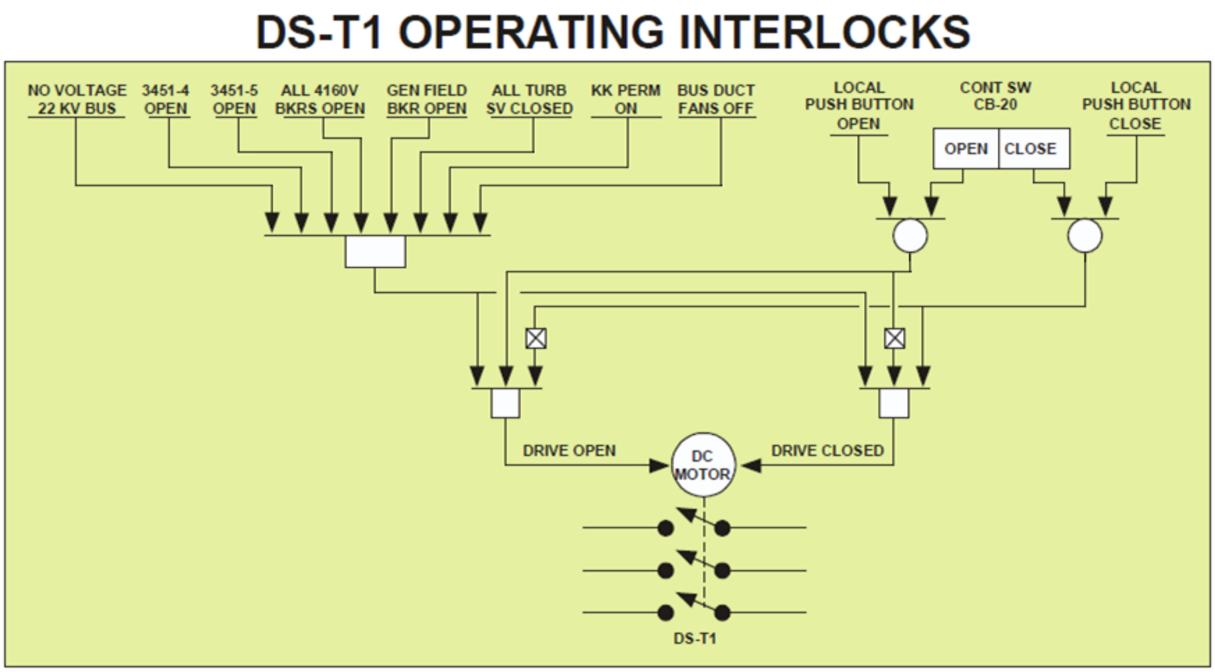
One key is on the Turbine Building Operator key-ring and an extra key is kept around the DS-T1 control switch on CB-20.

Operation of the key:

(a) Opens contacts to prevent motor operation.

(b) Withdraws a locking bolt that releases the motor-manual handle.

(Slide #91)


DS-T1 OPERATING INTERLOCKS
NO VOLTAGE 22 KV BUS
3451-4 OPEN
3451-5 OPEN
ALL 4160V BKRS OPEN
GEN FIELD BKR OPEN
ALL TURB SV CLOSED
KK PERM ON
BUS DUCT FANS OFF
LOCAL PUSH BUTTON OPEN
CONT SW CB-20
OPEN
CLOSE
LOCAL PUSH BUTTON CLOSE
DRIVE OPEN
DC MOTOR
DRIVE CLOSED
DS-T1

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/15/15	Tier #	2	
Change: 0	Group/Category #	2	
	K/A #	071 K3.04	
Level of Difficulty: 2	Importance Rating	2.7	

<u>Waste Gas Disposal System</u>: Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following: Ventilation system.

Question: 37

Given the following conditions:

- Plant is at 100% power.
- RM-051, Containment Radiation Monitor, is out of service for surveillance testing
- All other Radiation Monitors are OPERABLE.
- The in-service Waste Gas Decay Tank ruptures.

Which of the following Radiation Monitors provides the initiating signal that results in automatically aligning Control Room Ventilation to the Filtered Mode?

A. RM-043, Laboratory and Radioactive Waste Processing Building Exhaust Stack Monitor.

B. RM-073, Containment High Range Area Radiation Monitor.

C. RM-062, Auxiliary Building Vent Stack Normal Range Radiation Monitor.

D. RM-063, Accident Range Stack Radiation Monitor.

Answer: C

K/A Match:

Applicant must understand the effects of a failure of a Waste Gas Decay Tank on radiation levels in the Auxiliary Building, including how this failure will be detected by installed Radiation Monitors, and how those ventilation Radiation Monitors will affect the Ventilation System.

Explanation:

A. Incorrect. Plausible because RM-043 monitors Laboratory and Radioactive Waste Processing Building Exhaust. Depending on the alignment of the Auxiliary Building Ventilation System this process monitor could go into alarm. Incorrect because RM-043 does not cause a Containment Radiation High Signal (CRHS) which initiates a VIAS which realigns the Control Room Ventilation System.

B. Incorrect. Plausible if thought that the post-accident high range rad monitor performed this action .

C. **Correct**. RM-062 will initiate a CRHS which in turn causes a VIAS which places Control Room Ventilation System in the “Filtered Air” mode.

D. Incorrect. Plausible because RM-063 would sense the same high radiation as RM-062. Incorrect because there is no CRHS initiating signal from RM-063.

Technical Reference: LP 7-12-14, Slides #193, #199, #203, #222, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Control System-LO
EO 1.2 - **EXPLAIN** how each prime initiation signal is developed.
EO 1.5 - **EXPLAIN** the functions performed by each Engineered Safeguards Control signal.

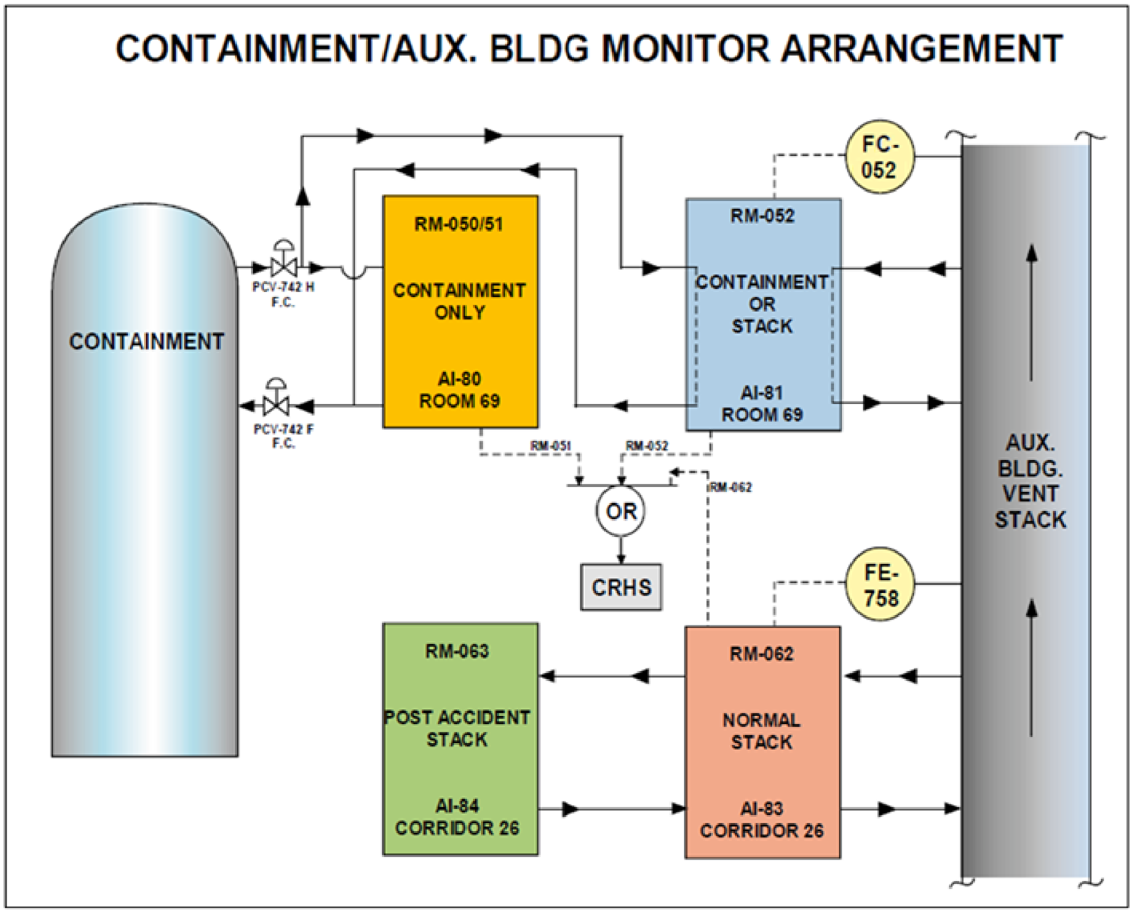
Question Source: Bank # X
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 11
55.43 ______

(Slide #193)


CONTAINMENT/AUX. BLDG MONITOR ARRANGEMENT
CONTAINMENT
PCV-742 H
F.C.
PCV-742 F
F.C.
RM-050/51
CONTAINMENT ONLY
AI-80
ROOM 69
RM-052
CONTAINMENT OR STACK
AI-81
ROOM 69
FC-052
RM-051
RM-052
RM-062
OR
CRHS
FE-758
RM-063
POST ACCIDENT STACK
AI-84
CORRIDOR 26
RM-062
NORMAL STACK
AI-83
CORRIDOR 26
AUX. BLDG. VENT STACK

EO *1.2 **(Slide #193)**

Containment Radiation High Signal (CRHS)

Three radiation monitors are used for CRHS.

(1) RM-051 – Containment gas monitor

(2) RM-052 – Containment/Ventilation Stack swing monitor

(3) RM-062 – Ventilation Stack monitor

RM-052 will normally be aligned to the ventilation stack but can be lined up to sample containment if RM-051 becomes inoperable or is undergoing maintenance or surveillance testing.

EO *1.5, 2.6 **(Slide #199)**

Containment Radiation High Signal (CRHS)

Prime CRHS Lockout Relays (86A/CRHS and 86B/CRHS)

When the CRHS L-O relays trip, operate to:

(1) Provide a signal to trip the VIAS L-O relays.

(2) Provide computer input.

(3) Provide annunciation.

EO *1.5 **(Slide #203)**

Ventilation Isolation Actuation Signal (VIAS)

VIAS isolates containment purge, air sample, and pressure relief to prevent release of significant quantities of gaseous radioactivity from containment in the event of a reactor coolant leak.

VIAS also performs the following:

(1) Secures waste gas release

(2) Puts the Control Room air ventilation system in the filtered mode

(a) Starts CR A/C units (VA-46A and VA-46B)

(b) Locks out 3rd stage compressor on air conditioners

(c) Closes CCW isolation valves to the A/C units.

(3) Cuts in ventilation for SI pump rooms and the spent regenerant tank room.

EO *1.4 **(Slide #222)**

RM-043

The Laboratory and Radioactive Waste Processing Building Exhaust Stack Monitor RM-043 is off line noble gas sample skid designed to monitor the combined ventilation exhaust from the Chemistry Laboratory and Radioactive Waste Processing Buildings.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	2	
Change: 1	Group/Category #	2	
	K/A #	075 K4.01	
Level of Difficulty: 2	Importance Rating	2.5	

Circulating Water System: Knowledge of circulating water system design feature(s) and interlock(s) which provide for the following: Heat sink.

Question: 38

A loss of the Screen Wash System occurs. As a result, the Raw Water Pumps lose their…

A. …sparging water source.

B. …primary seal water source.

C. …backup seal water source.

D. …bearing cooling water source.

Answer: A

K/A Match:

Applicant must be aware of design features that maintain the Raw Water System (heat sink). The Screen Wash System is a design feature of the Circulating Water System.

Explanation:

A. **Correct**. Loss of the Circulating Water System Screen Wash System will cause a loss of the sparging water source for the Raw Water Pumps which is the heat sink at Fort Calhoun Station. Sparging water is introduced at the suction of the Raw Water Pump 20 minutes prior to starting to eliminate sand accumulation that may be ingested into the pump suction bells when the pump is started (see referenced picture). Sparging flow is provided by the Screen Wash System.

B. Incorrect. Plausible because the Raw Water Pumps do require seal water. Incorrect because the primary seal water source is from Service/Potable Water Systems. Incorrect as there is no interface with Screen Wash System.

C. Incorrect. Plausible because there is a backup supply of seal water but it is provided by the discharge of the Raw Water Pump. Incorrect as there is no interface with Screen Wash System.

D. Incorrect. Plausible because the shaft sleeve bearings receive lubricating water from the Service/Potable Water Systems, and plausible because Screen Wash System flow is filtered and pressurized, so could be used as bearing cooling. Incorrect as there is no interface with Screen Wash System.

Technical Reference:	OI-RW-1, Precautions 4, Rev. 108
(Attach if not previously provided including revision number)	OI-RW-1, Attachment 1, Prerequisites 3, 4, and 6, Rev. 108
	LP 7-11-19, Slide #17, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-11-19, Raw Water System-Licensed Operator EO 1.2d - **DESCRIBE** the functional relationship between the Raw Water System and the: Circulating Water System. EO 1.2e - **DESCRIBE** the functional relationship between the Raw Water System and the: Screen Wash System.

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

Question History: Last NRC Exam 2007 NRC Exam, Question #65

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

10 CFR Part 55 Content:	55.41	7
	55.43	

PRECAUTIONS

1. Limiting Missouri River parameters for operation at, or above, an RCS temperature of 210°F are:
   - Minimum Level - 976 feet 9 inches
   - Maximum Level - 1,009 feet
   - Maximum Temperature - 87°F
2. If for any reason HCV-2893 or HCV-2894 are closed or the East Raw Water Header is isolated, ensure that at least one of the EFWST backup water supplies listed in AOP-30 are available.
3. Raw Water Pump minimum suction elevation is 973 feet 9 inches.
4. The Raw Water Sparging System for each RW pump should be in service for twenty minutes prior to starting these pumps, but is not required to maintain pump operability. The Raw Water Sparging System in normally in service continuously.

Attachment 1 - Raw Water System Startup

PREREQUISITES

1. Procedure Revision Verification:

   Revision No._____ Date:

2. Checklist OI-RW-1-CL-A has been completed.

3. IF the Raw Water Sparging System is available,
   THEN ensure the Raw Water Sparging System for the pump to be started is in service per OI-CW-2 for 20 minutes.

   - AC-10A, Raw Water Pump
   - AC-10B, Raw Water Pump
   - AC-10C, Raw Water Pump
   - AC-10D, Raw Water Pump

4. Screens and screenwash systems are in operation per OI-CW-2.

5. At least one Screen Inlet Sluice Gates to the oncoming Raw Water Pump is open, OR the Circ Water Pumps Interconnecting Sluice Gate is open to a cell with an open Screen Inlet Sluice Gate.

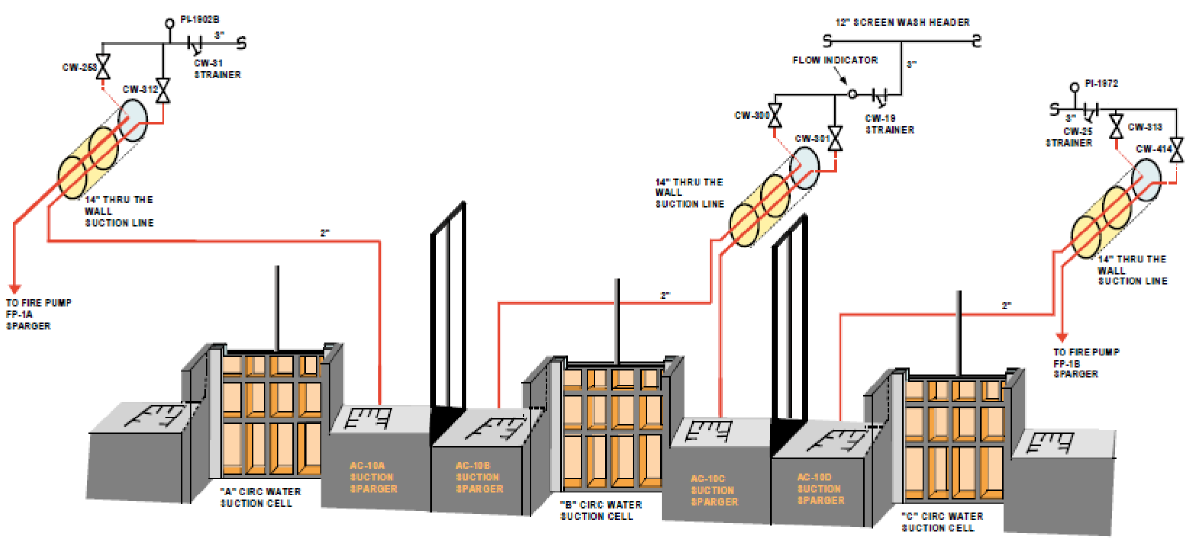
6. Potable Water is available to provide seal water to the Raw Water Pumps.

TO 2.0 **(Slide #17)**
General System Description
A sparger system has been installed on the inlet ledges of the pumps to eliminate sand accumulation that may be ingested into the pump suction bells when the pump is started.
Sparging flow is supplied by the Screen Wash System.
Lineup and control of this system is described in OI-CW-2.
<u>SUBJECT OE</u>: (Copy in OE Section)
LER-94-03, Inoperability of Raw Water Pumps Due to Excessive Sand Accumulation
IR 920196, AC-10B Overcurrent Trip Due to Being Sanded In

EO 1.2i **(Slide #25)**
System Interfaces
Potable and Service Water System
The RW Pumps receive seal water from the Service Water System (primary source) and the Potable Water System (secondary source).
The pumps can also supply their own seal water off the pump discharge if both outside sources are not available.

(Slide #17)

(Slide #17)



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/15/15	Tier #	1	
Change: 2	Group/Category #	1	
	K/A #	008 AK3.03	
Level of Difficulty: 4	Importance Rating	4.1	

Pressurizer Vapor Space Accident: Knowledge of the reasons for the following responses as they apply to the Pressurizer Vapor Space Accident: Actions contained in EOP for PZR vapor space accident/LOCA.

Question: 39

Following a leak in the Pressurizer steam space, the following conditions exist:

- Three Charging Pumps are running.
- Safety Injection Refueling Water Tank level is 170 inches and slowly lowering.
- Reactor Coolant System (RCS) pressure is 1750 psia and slowly lowering.
- RCS T_{COLD} is 520°F and stable.
- Four Reactor Coolant Pumps (RCPs) are running.
- Pressurizer level is 100%.

Which of the following actions should be taken per EOP-03, Loss of Coolant Accident?

A. Begin steaming of both Steam Generators to initiate a controlled cooldown at rate of 75°F/hr.

B. Stop all Reactor Coolant Pumps and initiate a Natural Circulation cooldown.

C. Conduct a rapid cooldown to place Shutdown Cooling in service within one hour.

D. Initiate a controlled cooldown and block the Pressurizer Pressure Low Signal prior to reaching RCS pressure of 1600 psia.

Answer: A

K/A Match:
Requires applicant knowledge of RCS and Pressurizer response to a Vapor Space Accident including the EOP required actions.

Explanation:

A. **Correct**. As outlined in TDB-EOP-03, Step 22, a controlled cooldown within Technical Specification limits is initiated. An aggressive cooldown is desired but must be within Technical Specification limits.

B. Incorrect. Plausible because this action would reduce flow out of the break. Incorrect because it is desirable to maintain forced circulation.

C. Incorrect. Plausible because the ultimate goal will be to establish Shutdown Cooling to repair the leak. Incorrect because this would violate cooldown limits.

D. Incorrect. Plausible because a cooldown needs to be initiated. Incorrect because blocking PPLS with a RCS leak is not allowed.

Technical Reference: (Attach if not previously provided including revision number)	EOP-03, Step 22, Rev. 38 TDB-EOP-03, Steps 22 & 28, Rev. 37a

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-15-23, LOCA-Licensed Operator
EO 2.4 - **EXPLAIN** the operator actions required to mitigate a Loss of Coolant Accident.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

EOP-03, Step 22

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CAUTIONS

1) When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr.
When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.

2) No more than three RCPs shall be in operation when RCS temperature is less than 500°F.

****

22. Commence a Steam Generator cooldown by manually operating the Steam Dump and Bypass Valves PER Attachment HR-12, Secondary Heat Removal Operation.

Time: ____________

22.1 **IF** local operation of MS-291, MS-292 is required,
THEN commence a Steam Generator cooldown PER Attachment HR-13, Local MS-291, MS-292 Operation.

c. Stabilize RCS temperature PER Attachment HR-2, Secondary Heat Removal Operation.

TDB-EOP-03, Step 22

EPG Step: 15

Deviation:
1) The EOP provides a plant specific caution to limit RCS cooldown rates.
2) EOP Contingency Actions discuss the use of Air Assisted Main Steam Safety Valves as an alternative steam generator controlled steaming path. EPG only recognizes the turbine bypass system (steam dump and bypass) and atmospheric dump valve systems as the means of controlled steam generator steaming.
3) The EOP provides a caution to remind operators of RCP operational limits.

If the LOCA has not been isolated, then the following actions are directed toward reestablishing RCS inventory control while maintaining RCS heat removal. The goal of this section is to establish shutdown cooling, if possible, as the means of core heat removal. A rapid plant cooldown via the steam generators is beneficial for all LOCAs, particularly small breaks. For small breaks, the steam generators are the major heat sink for RCS heat removal. An aggressive cooldown (while holding the cooldown rate within Technical Specification Limit) improves RCS heat removal by enhancing natural circulation and reflux boiling. Furthermore, an aggressive cooldown hastens the depressurization of the RCS. This results in higher safety injection flows which aids in regaining RCS inventory control.

For the largest breaks, the RCS depressurizes to an equilibrium pressure with the containment. In this condition, the RCS fluid is at a lower temperature than that of the steam generators. The steam generators, therefore, act as a heat source, superheating any steam in the RCS which may be flowing through the S/Gs to the break. By cooling down the steam generators, heat input to the RCS is reduced.

TDB-EOP-03, Step 28

EPG Step: 39

Deviation:
1) The plant-specific means of initiating Low Temperature Overpressurization Protection is described.
2) Contingency actions are provided if temperature instrumentation is inoperable.
3) The EOP provides steps to ensure the PORV Block Valves are open.

The EPG states to enable LTOP when enabling criteria are met. Details of initiation are not provided in the EPG. This justifies deviation 1.

The PPLS Block signal enables the LTOP circuitry when RCS temperature or pressure exceed limits of TDB-III.7.a. If either T-113 or T-123 is inoperable, PORVs may lift when LTOP is initiated[C2]. PPLS can still be blocked in this condition to allow reset of safeguards. The only implicit qualifier on setting the PPLS block signal is that, in a LOCA event, primary pressure has already fallen below the PPLS setpoint of 1600 psia. This justifies deviation 2.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/15/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	009 EA2.08	
Level of Difficulty: 2	Importance Rating	2.9	

<u>Small Break Loss of Coolant Accident</u>: Ability to determine or interpret the following as they apply to a small break LOCA: Letdown isolation valve position indication.

Question: 40

Given the following conditions:

- Plant is at 100% power.
- LCV-101-1, Letdown Throttle Valve, was being used for automatic control of Letdown.
- LCV-101-2, Letdown Throttle Valve, was closed.
- Normal Charging and Letdown were in service with CH-1A, Charging Pump, running.

The following conditions currently exist following a manual Reactor Trip:

- Pressurizer level is 25% and lowering.
- Pressurizer pressure is 1900 psia and lowering.
- Reactor Coolant System T_{COLD} is 540°F.
- All three Charging Pumps are running.

Assuming NO operator action, what is the expected status of Letdown Throttle Valves LCV-101-1 and LCV-101-2?

A. Both LCV-101-1 and LCV-101-2 are closed due to low Pressurizer level.

B. Both LCV -101-1 and LCV-101-2 are open further due to increased Charging flow.

C. LCV-101-1 is open further due to the increased Charging flow. LCV-101-2 will remain closed.

D. LCV-101-1 is throttled to the minimum flow position due to low Pressurizer level. LCV-101-2 will remain closed.

Answer: D

K/A Match:

Applicant must identify condition of the Letdown Throttle Valves based on RCS conditions during a Small Break LOCA.

Explanation:

A. Incorrect. Plausible because LCV-101-1 will be affected by the low Pressurizer level. Incorrect because the only way for LCV-101-2 to be impacted is when HC-101-2, Letdown Heat Exchanger Flow Control Selector Switch is in the "BOTH" position.

B. Incorrect. Plausible because all 3 Charging Pumps are running. Incorrect because even though there is an increase in Charging flow, it is the deviation from setpoint that is driving the position of the Letdown Throttle Valves

C. Incorrect. Plausible because LCV-101-2 will remain closed. Incorrect because even though there is an increase in Charging flow, it is the deviation from setpoint that is driving the position of the Letdown Throttle Valves.

D. **Correct**. LCV-101-1 is throttled to the minimum flow (26 GPM) position due to the Pressurizer level deviation from setpoint.

Technical Reference: LP 4-22-4, Slide #10, Rev. 0
(Attach if not previously provided including revision number) LP 7-11-2, Slides #59 & #60, Rev. 2

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-15-23, Loss of Coolant Accident-Licensed Operator
EO 2.2 - **STATE** the automatic actions that would be taken by Fort Calhoun systems to mitigate a LOCA.

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

Question History: Last NRC Exam None

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

EO *4.1, 5.1, *5.2 **(Slide #59)**

Major Component Description

Letdown Flow Control Valves (LCV-101-1 & LCV-101-2)

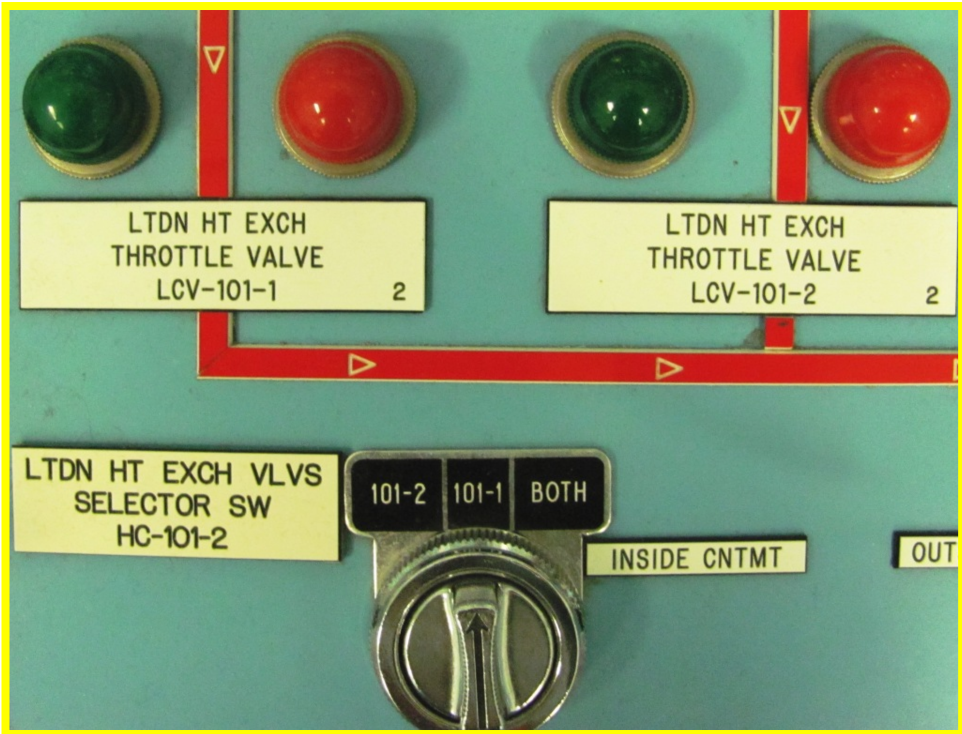
One handswitch on CB-1/2/3 allows selecting either or both valves for operation.

Normally, one valve is in use. "BOTH" position is selected in reduced RCS pressure situations (startup or shutdown conditions).

Valve position is indicated above the control switch on CB-1/2/3.

(LC) "LETDOWN CONTROL OFF NORMAL" alarm on annunciator A2 indicates that BOTH valves are in service.

(Slide #59)



EO *1.2 **(Slide #60)**

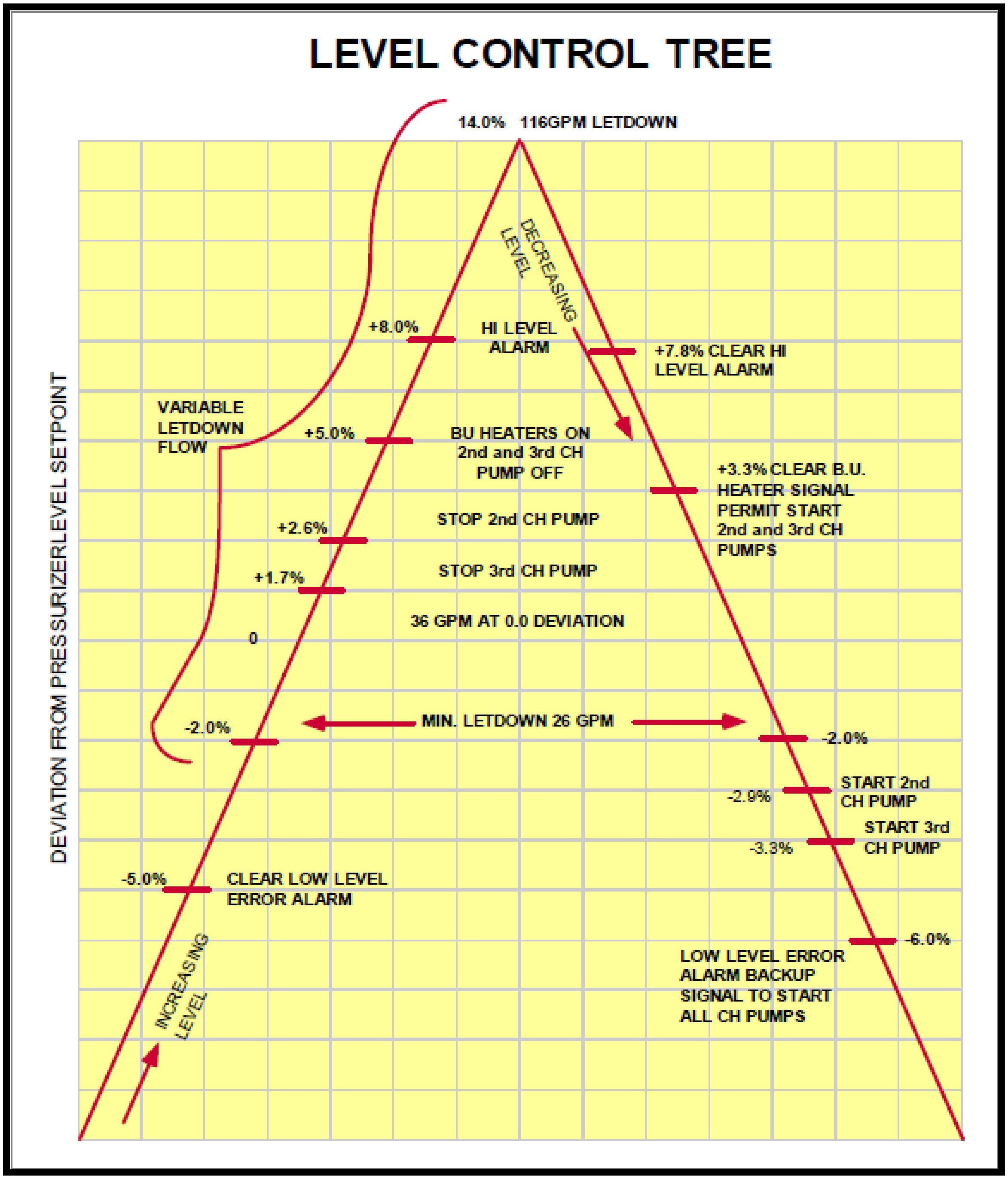
Major Component Description

Letdown Flow Control Valves (LCV-101-1 & LCV-101-2)

The valve position varies in response to the pressurizer level control system – when pressurizer level deviates from its setpoint value, the control system increases or decreases letdown flow to restore proper level.

Normally letdown flow is a minimum of 26 gpm (-2% level deviation) and a maximum of 116 gpm (+14% level deviation) by a flow limiter in the pressurizer level control circuitry.

Slide #10


LEVEL CONTROL TREE
14.0% 116GPM LETDOWN
DEVIATION FROM PRESSURIZERLEVEL SETPOINT
DECREASING LEVEL
+8.0%
HI LEVEL ALARM
+7.8% CLEAR HI LEVEL ALARM
VARIABLE LETDOWN FLOW
+5.0%
BU HEATERS ON 2nd and 3rd CH PUMP OFF
+3.3% CLEAR B.U. HEATER SIGNAL PERMIT START 2nd and 3rd CH PUMPS
+2.6%
STOP 2nd CH PUMP
+1.7%
STOP 3rd CH PUMP
36 GPM AT 0.0 DEVIATION
0
-2.0%
MIN. LETDOWN 26 GPM
-2.0%
-2.9%
START 2nd CH PUMP
-3.3%
START 3rd CH PUMP
-5.0%
CLEAR LOW LEVEL ERROR ALARM
-6.0%
LOW LEVEL ERROR ALARM BACKUP SIGNAL TO START ALL CH PUMPS
INCREASING LEVEL

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	011 EA1.13	
Level of Difficulty: 2	Importance Rating	4.1	

Large Break Loss of Coolant Accident: Ability to operate and monitor the following as they apply to a Large Break LOCA: Safety injection components.

Question: 41

Given the following conditions:

- A Large Break Loss of Coolant Accident has occurred.
- 161 KV has been lost to the site.
- Both Emergency Diesel Generators started.
- All other Safeguards equipment and systems operated properly.
- An STLS (Safety Tank Low Level) actuation has occurred.

Which of the following valves should be closed?

A. LPSI Loop Injection Valves, HCV-327, 329, 331, 333.

B. SIT Outlet Valves, HCV-2914, 2934, 2954, 2974.

C. Safety Injection Pumps SIRWT Recirculation Valves, HCV-385, 386.

D. Safety Injection/Containment Spray Pumps CCW Outlet Valve, HCV-474.

Answer: C

K/A Match:

Applicant must recognize and monitor Safety Injection valve repositioning during a Large Break Loss of Coolant Accident following a Recirculation Actuation Signal.

Explanation:

A. Incorrect. Plausible because the LPSI Pumps trip on a Recirculation Actuation Signal (RAS) but the LPSI Loop Injection Valves are opened by an SIAS and do not receive a RAS signal.

B. Incorrect. Plausible if thought that a Safety Injection Actuation Signal was sent to these valves. These valves do not receive an accident signal.

C. **Correct**. Given the initial conditions and STLS has occurred, a Recirculation Actuation Signal has occurred. RAS is initiated at a SIRWT level of 16 inches.

D. Incorrect. This valve is opened by a Containment Isolation Actuation Signal and does not receive a Recirculation Actuation Signal.

Technical Reference: LP 7-12-14, Slides #144 & #333, Rev. 1

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Control-Licensed Operator
EO 1.5 - **EXPLAIN** the functions performed by each Engineered Safeguards Control signal.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

(Slide #144)

RAS FUNCTIONS

- Trip and lock out LPSI pumps.
- Shift HPSI and containment spray pump suction to the containment sump.
  - LCV-383-1 and LCV-383-2 close.
  - HCV-383-3 and HCV-383-4 open.
- Isolate safety injection pump minimum recirculation to the SIRWT.
  - HCV-385 and HCV-386 close.
- Establishes full component cooling water flow to the shutdown cooling heat exchangers when RAS signal enable switch is in enable.
  - HCV-480, HCV-481, HCV-484 and HCV-485 open.

EO *1.5 **(Slide #333)**

Emergency Operation

If SIRWT level lowers to 16”, STLS initiates RAS to align the ECCS for long term cooling.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	015/017 AK3.02	
Level of Difficulty: 3	Importance Rating	3.0	

Reactor Coolant Pump Malfunctions: Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): CCW lineup and flow paths to RCP oil coolers.

Question: 42

Given the following conditions:

- EOP-03, Loss of Coolant Accident, is in progress.
- Reactor Coolant Pumps were tripped due to symptoms of a Reactor Coolant System (RCS) to Component Cooling Water (CCW) leak.
- HCV-438C, Inside Containment Isolation Valve, failed to close.

Which of the following is the reason for hand jacking HCV-438D, Outside Containment Isolation Valve CLOSED once RCS pressure is less than 380 psia?

A. Meet closure requirements for Containment Integrity.

B. Prevent overpressurizing the CCW Surge Tank.

C. Prevent a CCW to RCS leak when seal pressure is less than CCW pressure.

D. Prevent a Loss of Coolant Accident outside Containment.

Answer: D

K/A Match:
Applicant must understand modifications made to CCW supply to RCPs to avoid a LOCA outside Containment.

Explanation:

A. Incorrect. Plausible if thought that Containment Integrity was a concern. Incorrect because HCV-438D is equipped with a nitrogen backup bottle to assure Containment isolation.

B. Incorrect. Plausible because the leak is into the CCW System and the CCW Surge Tank is pressurized with nitrogen. Incorrect because the CCW Surge Tank contains a relief valve that vents to the Waste Gas Vent Header.

C. Incorrect. Plausible because in the initial conditions, a LOCA is occurring into the CCW system. When seal pressure lowers to less than CCW pressure, it is plausible that the leak could “go the other way” and become a CCW-to RCS leak. While this could happen, it isn’t the reason that the valve is Handjacked closed. Incorrect because the reason HCV-438D is hand jacked closed is to prevent a LOCA outside Containment.

D. **Correct**. Unlike HCV-438A/B/D, HCV-438C is a “flow to close” valve. Engineering evaluation determined that a LOCA outside Containment could occur due to the design of these valves. Consequently, HCV-438C was reversed to minimize the potential of a LOCA outside Containment and this is the reason that HCV-438D is hand jacked closed.

Technical Reference: LP 7-11-6, Slides #45 & #136, Rev. 1
(Attach if not previously provided including revision number)
TDB-EOP-03, Contingency Action Step 13.b, Rev. 37a

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-6, Component Cooling Water System-Licensed Operator
EO 1.2 - **EXPLAIN** the operation of controls associated with the CCW System valves operated from the Control Room.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 10
55.43 ____

EO *1.2 **(Slide #136)**

Components Cooled by CCW

Reactor Coolant Pump seal coolers and lube oil coolers (RC-3A/B/C/D)

Containment isolation valves (HCV-438A/B/C/D) are air-operated, fail open valves.

Nitrogen bottle back-up is supplied to HCV-438B and HCV-438D to assure containment isolation.

The potential existed for an RCP seal cooler heat exchanger tube rupture to produce an unisolable leak outside containment.

Leak isolation required closing the HCV-438A/B/C/D valves. (The original valves were flow-to-open globe valves.)

HCV-438A/B/D are flow-to-open, where HCV-438C is flow-to-close to aid in isolating a RCP seal cooler heat exchanger leak.

In the unlikely event that HCV-438C does not close, then per AOP-22, EOP-03 and EOP-20, HCV-438D must be closed manually. Caution: RCS pressure must be less than 380 psia to prevent possible operator injury.

EO 4.3 **(Slide #45)**

Major Component Description

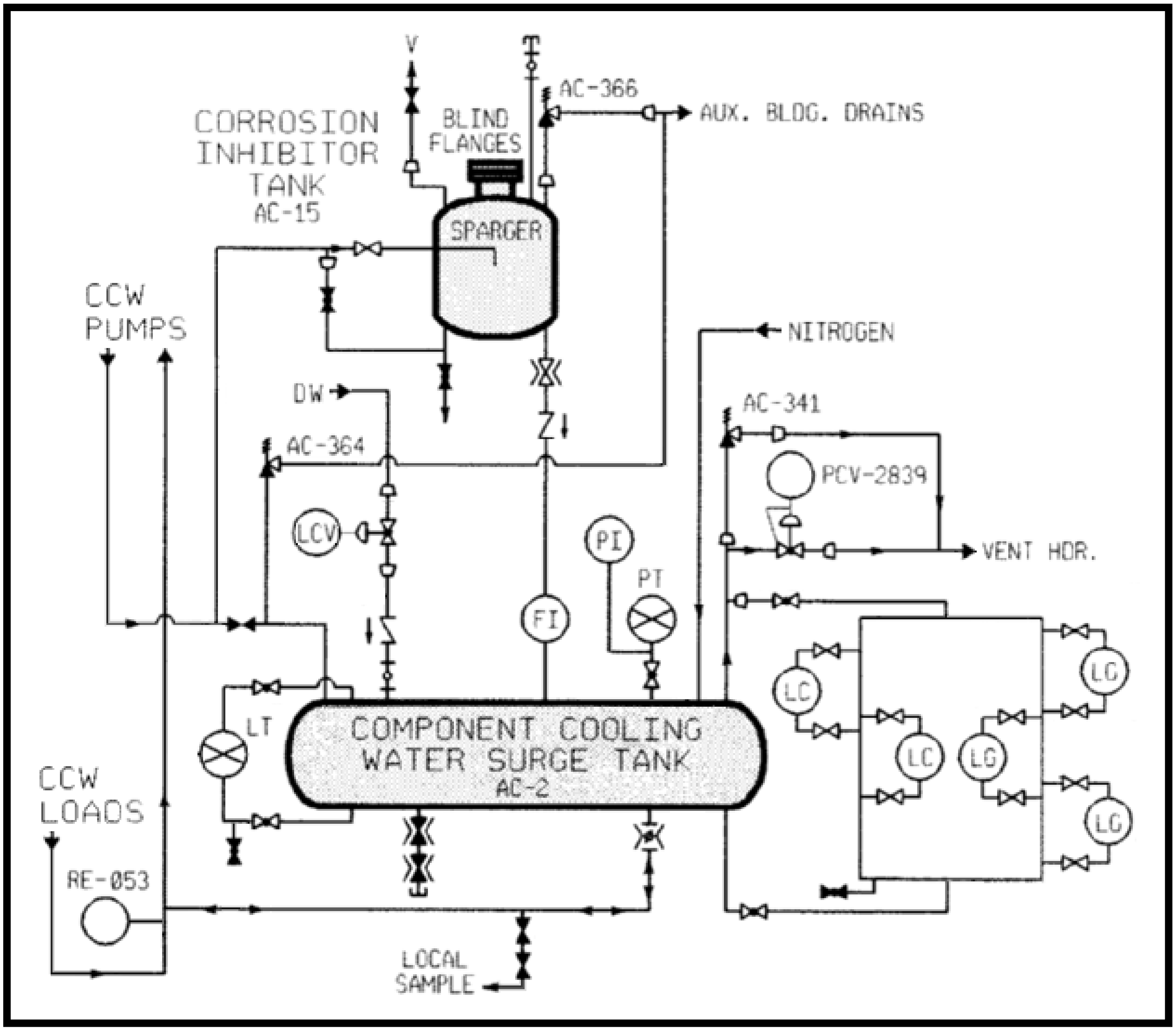
Component Cooling Water Surge Tank (AC-2)

The CCW Surge Tank is protected from overpressure:

PCV-2839 opens to bleed-off nitrogen gas to the Waste Disposal System vent header when pressure reaches approximately 46 psig.

A relief valve (AC-341) opens at 48 psig, directing nitrogen gas to the Waste Disposal System through the same vent path as PCV-2839.

(Slide #45)



1) **IF** an RCS-to-CCW leak is evident, **THEN** minimize RCS leakage by performing the following:

A. Trip all RCPs.

B. Close all of the RCP Coolers CCW Valves, HCV-438A/B/C/D.

****

CAUTION

The flange upstream of HCV-438D, RCP Cooler CCW Valve, may fail if it is pressurized to greater than 380 psia.

****

b.1 **IF** HCV-438C, RCP Cooler CCW Valve, fails to close,
THEN perform the following:

1) Lower RCS pressure to less than 380 psia PER Attachment PC-11, Pressure Control.
2) Direct RP to Survey Room 13 for entry.
3) Loosen Stem nut for HCV-438D, "RCP RC-3A-D LUBE OIL & SEAL CLRS CCW OUTLET OUTBOARD ISOL VLV". (Room 13)
4) Unlock and handjack HCV-438D, "RCP RC-3A-D LUBE OIL & SEAL CLRS CCW OUTLET OUTBOARD ISOL VLV", closed. (Room 13)
5) **IF** HCV-438D, RCP Cooler CCW Valve, fails to close,
THEN lower RCS pressure to less than 180 psia.
6) Close HCV-438D, RCP Cooler CCW Valve, from CB-4.

C. Close TCV-202, Letdown Isolation Valve.

D. Close HCV-204, Letdown Isolation Valve.

One path for a radiological release from the primary is from RCP to CCW cooling coils to the CCW system outside of containment. Hence, if an indication of a primary to CCW break exists, closing the supply and return valves for RCP cooling isolates this leak path. Primary coolant can also leak into the CCW system across the tube surface of the non-regenerative heat exchanger in the letdown system. Closing TCV-202 and HCV-204 prevents further loss of RCS inventory upstream of the letdown isolation valve via this path. The remaining inventory between the NR HX outlet and the Volume Control Tank inlet or the CVCS ion exchangers may continue to leak into the CCW until pressure is equalized across the NR HX tube sheet. This justifies deviation 2.

The CEDMs pose another potential source of RCS to CCW leakage. CCW is the cooling medium for the CEDM seal coolers. It should be noted that the CCW to the CEDMs is isolated when the CCW to RCP Isolation Valves, HCV-438A/B/C/D, are closed.

Closing HCV-438A/B/C/D will secure cooling of bearing and seal surfaces of all Reactor Coolant Pumps. In order to minimize the damage to these bearings and seal surfaces, any operating RCP is secured before component cooling water is isolated. This justifies deviation 3.

The EOP provides a note to the operator that a rising count rate on CCW radiation monitor RM-053, or rising CCW surge tank level or pressure may be indicative of a RCS to CCW leak. This information is more appropriately contained in a note. This justifies deviation 4.

If RCS to CCW leakage is evident, attempts are made to isolate the leak. This is done by isolating the potential leak paths across RCP cooling coils and the Non-Regenerative Heat Exchanger's heat transfer tubing. Before isolating the supply and return of CCW from the RCPs, all operating Reactor Coolant Pumps are stopped. This is done to minimize the damage done to RCP bearings and seal cartridge surfaces. The EOP gives instructions to isolate components that could possibly be the source of an RCS to CCW leak should one be indicated. The instruction is given as an action statement, not a verification. The EOP gives guidance for plant specific isolation and pressure requirements for isolation of RCP coolers. This justifies deviation 5.

The EOP contains a caution and contingency if HCV-438C fails to close. If HCV-438C (the inside containment isolation valve) fails to close, HCV-438D will not close due to excessive pressure. The caution and contingency will protect the operator from personal injury while manually closing HCV-438D. If HCV-438D can not be manually closed an attempt is made to remotely close when pressure is reduced low enough to allow it. This justifies deviation 6.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	022 AK1.03	
Level of Difficulty: 3	Importance Rating	3.0	

<u>Loss of Reactor Coolant Makeup</u>: Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level.

Question: 43

Given the following conditions from an event that occurred one minute ago:

- Reactor Coolant System (RCS) pressure is 2050 psia and lowering.
- Pressurizer level is 56% and lowering.
- Letdown flow is 26 gpm.
- Charging flow is 120 gpm.
- Reactor power is 99.5% and steady.
- RCS T_{COLD} is 542°F and steady.
- RCS T_{HOT} is 594°F and steady.
- Steam Generator Blowdown Radiation Monitors indicate 96 cpm & 220 cpm, both are steady.
- Containment Sump level is 18" and steady.
- Volume Control Tank level is 47% and lowering.

Which of the following occurred?

A. Steam Generator Tube Leak.

B. RCS leak inside Containment.

C. Charging header leak in Room 13.

D. Steam Generator Safety Valve failed open.

Answer: C

K/A Match:

Applicant must be able to evaluate conditions and understand the operational implication of a delta between Charging flow and Pressurizer level.

Explanation:

A. Incorrect. Plausible because a Steam Generator Tube Leak Would Cause Pressurizer level to lower. Incorrect because SG Blowdown Radiation Monitors are not reading the same but are both reading constant.

B. Incorrect. Plausible because and RCS leak inside Containment would cause Pressurizer level to lower. If the leak were inside Containment, sump level should be rising.

C. **Correct**. The mismatch between Charging and Letdown flow is indicative of either a Charging header leak, a leak from the RCS or an excessive heat removal event. If the leak was into the SGs, RM-054A/B would be increasing. If the leak was into Containment, Containment Sump level would be rising. If a SG safety valve was stuck open, RCS temperatures would be lowering.

D. Incorrect. Plausible because a SG Safety Valve opening would cause RCS to cool and Pressurizer level to lower. Incorrect because RCS temperatures would also be lowering for this event.

Technical Reference: LP 7-17-33, Pages 5 & 6, Rev. 5

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-33, CVCS Leak-Licensed Operator
EO 1.2 - **DESCRIBE** how the plant responds to a CVCS leak in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

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

Question History: Last NRC Exam 2007 NRC Exam, Question #5

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

10 CFR Part 55 Content: 55.41 5
55.43 ____

A. Plant Response

1. AOP-33 deals with only a leak in the CVCS that can be contained by isolating Charging and Letdown. Otherwise, it is considered an RCS Leak and actions will be taken per AOP-22.

2. If PZR level decrease is abnormal then the leak is not contained and the contingency directs operators to AOP-22.

3. With a CVCS leak in the Auxiliary Building, AOP-09 will be implemented. If the leak is to the Radwaste System, implementation of AOP-09 may not be necessary.

4. The VCT level will increase at a rate of 1% every 6 minutes due to RCP bleedoff with Charging and Letdown isolated.

5. If VCT level is lowering, then the leak is either in the VCT or downstream of the VCT.

6. If the Reactor is critical and Charging flow cannot be restored prior to reaching an actual PZR level of 32% the Reactor will be tripped within 6 hours.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/17/15	Tier #	1	
Change: 0	Group/Category #	1	
	K/A #	025 AK2.03	
Level of Difficulty: 3	Importance Rating	2.7	

Loss of Residual Heat Removal System: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following: Service water or closed cooling water pumps.

Question: 44

Given the following conditions:

- Shutdown Cooling (SDC) System is in service using AC-4A, SDC Heat Exchanger.
- Reactor Coolant System (RCS) is at 290°F and 235 psia.
- Plugging of both Raw Water Strainers has resulted in a loss of Raw Water.

Which of the following is the preferred method for restoring core cooling?

A. Place AC-4B, SDC Heat Exchanger, in service using the CCW System.

B. Establish Fire Water backup cooling to SDC Heat Exchanger, AC-4A.

C. Start one HPSI Pump to provide injection into the RCS.

D. Establish Fire Water backup cooling to two CCW Heat Exchangers.

Answer: D

K/A Match:
Applicant must have knowledge of system alignments available to cool the RCS when the heat sink to the Shutdown Cooling Heat Exchangers is lost.

Explanation:

A. Incorrect. Plausible because AC-4B is available but it has also lost cooling water.
B. Incorrect. Plausible because Fire Water does provide backup cooling to some components but there is no procedural guidance for aligning it to an SDC Heat Exchanger.
C. Incorrect. Plausible because one HPSI Pump is still available for injection because RCS T_{COLD} is not less than 270°F (the point at which all 3 HPSI Pumps must be disabled). In order to perform this action Pressurizer manway would have to be removed (or any RCS venting of at least 0.94 in.^{2}). Incorrect because the RCS is intact.
D. **Correct**. Per AOP-18, Loss of Raw Water, Attachment B, Fire Protection System Backup, aligns cooling water to 2 CCW Heat Exchangers.

Technical Reference: AOP-18, Attachment B, Rev. 8b
(Attach if not previously provided including revision number) Technical Specification LCO 2.3(3), LTOP, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-11, Loss of Component Cooling Water-Licensed Operator EO 1.2 - **DESCRIBE** how the plant responds to a Loss of Component Cooling Water in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

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

Question History: Last NRC Exam 2007 NRC Exam, Question #6

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

10 CFR Part 55 Content:	55.41	10
	55.43	

A loss of the Raw Water System will result in the overheating or loss of components cooled by CCW such as Reactor Coolant Pumps, CEDMs, and air conditioning units. At least one RW pump must be in operation during normal plant operation. During shutdown cooling and the period following a SIAS, the operation of two pumps maybe required.

AOP-18 is designed specifically to mitigate a heavy load drop in the Intake Structure, which damages the power cables to all four Raw Water Pumps. The cooldown to less than 300°F is performed to comply with Technical Specification 2.0.1 to be less than 300°F within 6 hours of entering mode 3. Attachment B is implemented in order to maintain the plant in a stable condition while repairs are made to regain Raw Water. The Fire Protection backup to the Raw Water CCW Heat Exchangers does not provide enough cooling capacity to cool the plant down on Shutdown Cooling. Technical Specification 2.0.1 allows 30 hours to perform a plant cooldown from 300°F to cold shutdown conditions.

Attachment B

Fire Protection System Backup

INSTRUCTIONS | CONTINGENCY ACTIONS

1. Inform Security and the RP Technician that the door between Room 18 and Room 19 will be open.

<u>NOTES</u>

1. Hoses and couplings (180° coupling for AC-1A, straight hose for AC-1B and a 90° coupling for AC-1C or AC-1D) required for connecting RW/CCW HX drains to the Fire Protection System are located in the AI-100, AOP-06 Supply Cabinet (Corridor 4).

2. SO-G-103, <u>Fire Protection Operability and Surveillance Requirements</u>, contains requirements for fire protection system.

2. <u>Determine</u> **TWO** RW Heat Exchangers to be used for backup cooling:

   - AC-1A
   - AC-1B
   - AC-1C
   - AC-1D

3. <u>Connect</u> a 2 ½ inch fire hose from FP-418, "FIRE HOSE CABINET FP-7C 2 1/2" AUX HOSE CONNECTION VALVE" (Room 19), to **ONE** of the following RW/CCW HX Inlet Drain Valves:

   - RW-213, "CCW HEAT EXCHANGER AC-1A DRAIN VALVE" (Corridor 4)

   - RW-197, "CCW HEAT EXCHANGER AC-1B DRAIN VALVE" (Corridor 4)

   - RW-214, "CCW HEAT EXCHANGER AC-1C DRAIN VALVE" (Room 18)

   - RW-215, "CCW HEAT EXCHANGER AC-1D DRAIN VALVE" (Room 18)

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.3 Emergency Core Cooling System (Continued)

(3) Protection Against Low Temperature Overpressurization

The following limiting conditions shall be applied during scheduled heatups and cooldowns. Disabling of the HPSI pumps need not be required if the RCS is vented through at least a 0.94 square inch or larger vent.

Whenever the reactor coolant system cold leg temperature is below 350°F, at least one (1) HPSI pump shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 320°F, at least two (2) HPSI pumps shall be disabled.

Whenever the reactor coolant system cold leg temperature is below 270°F, all three (3) HPSI pumps shall be disabled.

In the event that no charging pumps are operable when the reactor coolant system cold leg temperature is below 270°F, a single HPSI pump may be made operable and utilized for boric acid injection to the core, with flow rate restricted to no greater than 120 gpm.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 2	Group/Category #	1	
	K/A #	026 AA1.05	
Level of Difficulty: 3	Importance Rating	3.1	

Loss of Component Cooling Water: Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm.

Question: 45

Given the following conditions:

- Plant is at 100% power.
- CB-1/2/3/A1, Window C-3L – COMPONENT COOLING WATER SURGE TANK HIGH OR LOW LEVEL has alarmed.
- LCV-2801, Component Cooling Water (CCW) Surge Tank Makeup Valve, was opened and is maintaining Surge Tank level stable at 38 inches.
- CCW Pump discharge pressure and Surge Tank pressure are normal.

If conditions remain unchanged, which of the following actions should be taken per AOP-11, Loss of Component Cooling Water?

A. Trip the Reactor and stop all CCW Pumps.

B. Start DW-40B, Demineralized Water Transfer Pump.

C. Close HCV-474, Safety Injection/Containment Spray Pump Coolers Inlet Valve.

D. Start DW-41B, Primary Water Booster Pump.

Answer: C

K/A Match:

Applicant must examine initial conditions and make a determination to address lowering Surge Tank level.

Explanation:

A. Incorrect. Plausible because the first step in AOP-11 asks if the CCW/RW System operation is normal, if not, the Reactor is tripped and CCW Pumps are stopped, however, that is not part of the conditions listed here.

B. Incorrect. Plausible because starting DW-40B is addressed in AOP-11 as a Contingency Action but only if LCV-2801 failed to open. Incorrect because LCV-2801 has opened and is maintaining level; this implies that DW-40A/B, Deaerated Water Transfer Pumps are already running.

C. **Correct**. Per the Annunciator Response Procedure, should Surge Tank level continue to lower or just be maintained, then AOP-11 should be referenced. In an attempt to isolate the leak, AOP-11 goes through a series of valves that are the most likely places for the lowering Surge Tank level.

D. Incorrect. Plausible because starting DW-41B is addressed in AOP-11 but only if a 480 V Load Shed has occurred. Incorrect because LCV-2801 has opened and is maintaining level.

Technical Reference: ARP-CB-1/2/3/A1, Window C-3L, Rev. 38
(Attach if not previously provided including revision number) AOP-11, Steps 1, 3, 4, 8, 11, & 12, Rev. 16

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-11, Loss of Component Cooling Water-Licensed Operator
EO 1.3 - **DESCRIBE** the major recovery actions of this AOP.

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

Question History: Last NRC Exam 2005 NRC Exam

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

Panel: **CB-1/2/3**	Annunciator: **A1**	Window: **C-3L**

COMPONENT COOLING WATER SURGE TANK
HIGH OR LOW LEVEL

Page 2 of 2

SAFETY RELATED

OPERATOR ACTIONS (continued)

5. IF AC-1179 and LCV-2801 are open AND Level does not rise, THEN check the demineralized water supply to the CCW Surge Tank (Room 69).

6. IF Level is lowering, THEN implement AOP-11.

7. IF CCW surge tank is high AND LCV-2801 is closed, THEN check for Primary to CCW leak by checking RM-053.

8. IF an RCS-to-CCW leak is indicated, THEN GO TO AOP-22.

1) ATTEMPT TO isolate the CCW leak by performing the following:

A. Place HCV-474, "SI/CS PUMP CLRS AC INLET VALVE" in "CLOSE".

1). (continued)

B. Place **ANY** or all of the following switches in "CLOSE":

- "CONT RM AIR COND VA-46A CCW VALVES HCV-2898A/B"
- "CONT RM AIR COND VA-46B CCW VALVES HCV-2899A/B"
- "SPRAY PUMP SI-3A BEARING COOLER CCW VALVES HCV-2813A/B"
- "LPSI PUMP SI-1A BEARING COOLER CCW VALVES HCV-2808A/B"
- "HPSI PUMP SI-2A BEARING COOLER CCW VALVES HCV-2810A/B"

1. Verify normal CCW/RW System operation by performing the following: a. Ensure at least one CCW Pump is operating.	1.1 **IF** CCW/RW System operation is **NOT** normal, **THEN** GO TO Step 11.

11. **IF ANY** of the following conditions exist:

- CCW System pressure is less than 60 psig
- A known unisolable leak exceeds makeup capability

THEN <u>shutdown</u> the CCW System by placing all CCW Pump Control Switches, AC-3A/B/C, in "PULL-TO-LOCK".

12. **IF** the Reactor is **NOT** tripped,

AND ANY of the following conditions exist:

- CCW flow is lost for five minutes
- Motor radial or thrust bearing temperatures are greater than or equal to 203°F for RC-3A/3C/3D (ERF page 342)
- Motor radial or thrust bearing temperatures are greater than or equal to 230°F for RC-3B (ERF page 342)
- Lower seal temperature is greater than or equal to 200°F (ERF page 342)

THEN initiate a Reactor shutdown by performing the following:

a. Trip the Reactor.

12.1 **IF** the Reactor is tripped,

AND ANY of the following conditions exist:

- CCW flow is lost for five minutes
- Motor radial or thrust bearing temperatures are greater than or equal to 203°F for RC-3A/3C/3D (ERF page 342)
- Motor radial or thrust bearing temperatures are greater than or equal to 230°F for RC-3B (ERF page 342)
- Lower seal temperature is greater than or equal to 200°F (ERF page 342)

THEN terminate forced RCS flow by performing the following:

a. Stop all RCPs.

13. **IF** CCW can **NOT** be restored,

THEN align RW to desired components as follows:

a. **IF** SIAS/CIAS has occurred,

THEN ensure **ALL** of the following switches are in "OVRRD":

- HC-2809/11/14/15, "SI PUMP AC VALVES SIAS OVERRIDE SWITCH"
- HC-2808/10/12/13, "SI PUMP AC VALVES SIAS OVERRIDE SWITCH"
- HC-400/403, "CNTMT CLR AC VLVS CIAS OVERRIDE SWITCH"

b. <u>Ensure</u> **BOTH** RW/CCW Backup Header Isolation Valves are open:

- HCV-2893
- HCV-2894

(continue)

3. **IF** the CCW Surge Tank level is less than 42 inches, **THEN** fill the CCW Surge Tank by performing the following:

   a. Open LCV-2801, CCW Surge Tank Makeup Valve, as necessary to refill the CCW Surge Tank.

   b. **IF** desired, **THEN** place LCV-2801 in "CLOSE".

NOTE

Hose and fittings are located South of AC-2, CCW Surge Tank.

3.1 **IF** LCV-2801 does not open, **THEN** fill the CCW Surge Tank by performing the following:

   a. Ensure either DW-40A/B is running.

   b. Connect hose between DW-130, "PRIMARY WATER STORAGE TANK DW-45 DEMIN WATER INLET DRAIN VALVE" and AC-351, "COMP COOLING WTR SURGE TANK AC-2 SURGE LINE SAMPLE VALVE" (Room 69).

4. **IF** a 480 V load shed has occurred, **THEN** ATTEMPT TO replenish CCW Surge Tank level by performing the following:

   a. Close **BOTH** of the following (Room 69):

      - DW-118, "PRIMARY WTR VALVE PCV-1553 OUTLET VALVE"
      - DW-117, "PRIMARY WTR VALVE PCV-1553 INLET VALVE"

b. Open **ALL** of the following valves (Room 69):

- DW-119, "PRIMARY WATER STORAGE TANK DW-45 PRESSURE REGULATING VALVE PCV-1553 BYPASS VALVE"

- DW-127, "DEMIN WATER HEADER TO DEAERATED WATER HEADER CROSSTIE ISOLATION VALVE"

- DW-128, "DEMIN WATER HEADER TO DEAERATED WATER HEADER CROSSTIE ISOLATION VALVE"

c. Start DW-41B, Primary Water Booster Pump.

d. Maintain AC-2, CCW Surge Tank level between 42 and 44 inches by performing the following:

1) Open LCV-2801, CCW Surge Tank Makeup Valve, as necessary to refill the CCW Surge Tank.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/15/15	Tier #	1	
Change: 0	Group/Category #	1	
	K/A #	027 G 2.4.2	
Level of Difficulty: 2	Importance Rating	4.5	

Pressurizer Pressure Control System Malfunction: Emergency Procedures/Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Question: 46

Which of the following describes the events that occur if PIC-103X, Pressurizer Pressure Indicator, fails low? Assume Channel X of Pressurizer Pressure and Level Control are the controlling channels, and NO operator action is taken.

A. Proportional Heaters deenergize, Backup Heaters energize, and Pressurizer pressure rises until a High Pressure trip occurs.

B. Proportional and Backup Heaters energize and Pressurizer pressure rises until a High Pressure trip occurs.

C. Proportional and Backup Heaters are blocked until reset, and Pressurizer pressure slowly lowers until a TM/LP trip occurs.

D. Proportional Heaters energize, Backup Heaters deenergize, and Pressurizer pressure stabilizes at a lower value.

Answer: B

K/A Match:
Applicant should be able to determine response of the RPS due to a Pressurizer Pressure Control System Malfunction with no operator action. This condition would result in EOP entry.

Explanation:

A. Incorrect. Plausible because the Backup Heaters will energize. Incorrect because the Proportional Heaters will also energize.
B. **Correct**. When the Pressurizer Pressure Transmitter fails low, the Proportional and Backup Heaters energize. This causes pressure to rise until a High Pressure Reactor trip is generated. The two Pressurizer Pressure Control Channels (PT-103X and PT-103Y) are separate from the four Pressurizer Pressure Reactor Protective System Channels (PT-102A/B/C/D).
C. Incorrect. Plausible if thought that the Pressurizer Heaters are locked out when the Pressurizer Pressure Transmitter fails low. This only occurs at 2350 psia when the transmitter fails high.
D. Incorrect. Plausible because even with the Proportional Heaters energized there is insufficient heat input to overcome the bypass flow to the Pressurizer Spray Valve. Incorrect because the Backup Heaters will also energize and the Reactor would trip on high pressure.

Technical Reference: LP 7-11-20, Slides #174 & #204, Rev. 1
(Attach if not previously provided including revision number) LP 7-20-12, Slide #135, Rev. 0

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-20, Reactor Coolant System-Licensed Operator
EO 4.0 - When given specific plant conditions, **EXPLAIN** operating principles to predict response of Reactor Coolant System (RCS) Instrumentation.
EO 4.4 - **EXPLAIN** the interlocks and control functions associated with RCS Instrumentation.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

EO *4.2b **(Slide #204)**

Detailed Component Description

Pressurizer (RC-4)

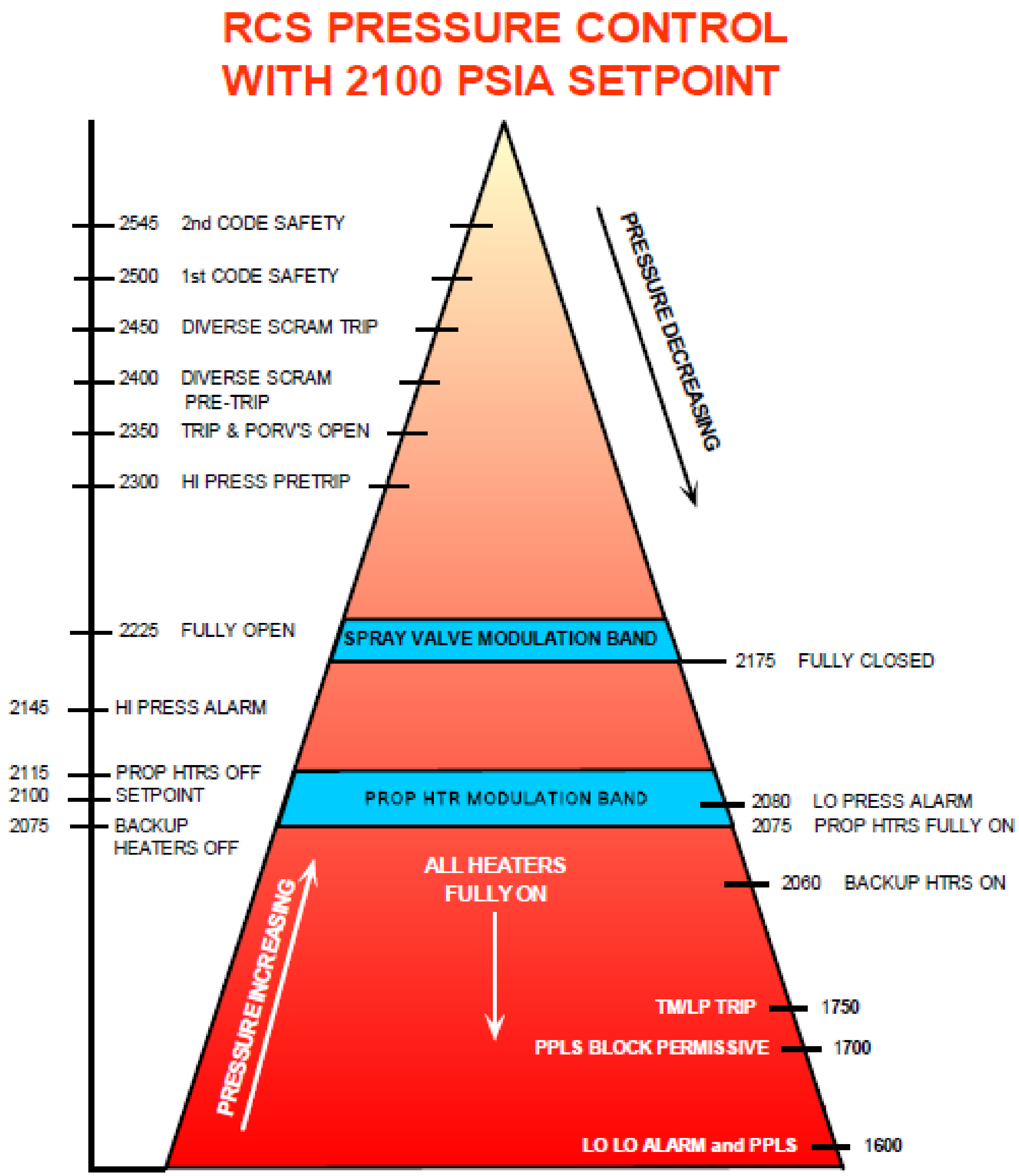
Pressurizer Pressure Instrumentation

Pressurizer Pressure Control Channels (PT-103X and PT-103Y)

Provides signals for automatic control of pressurizer pressure.

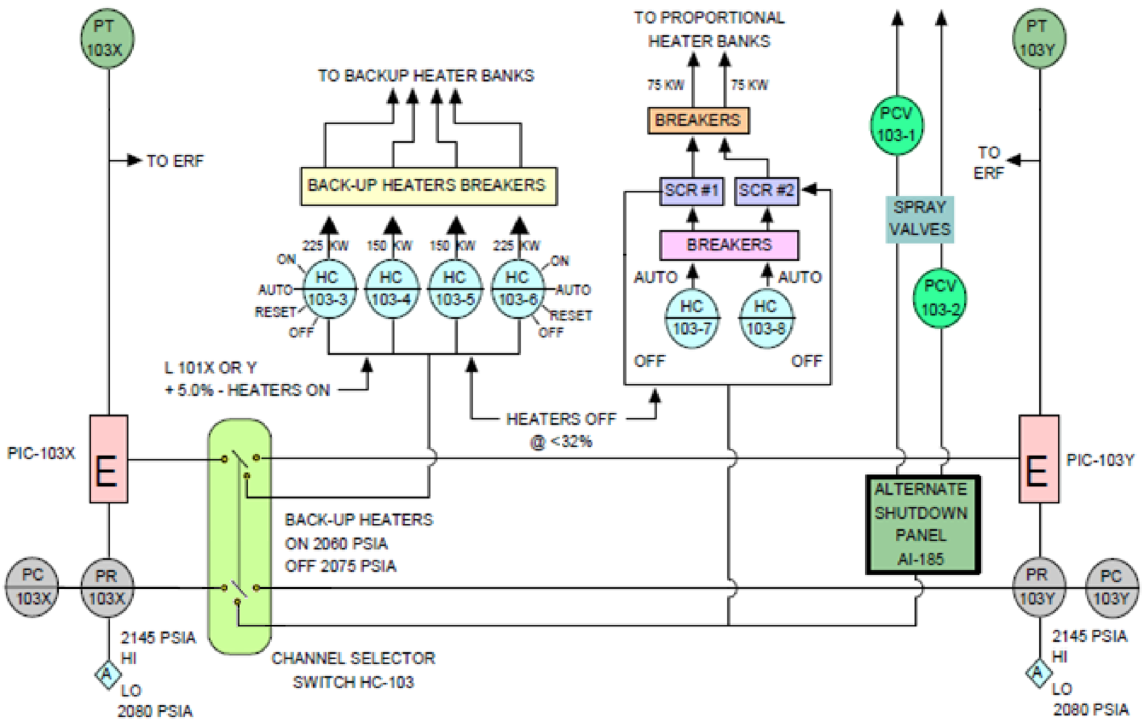
The ERF computer receives a pressure signal from each control channel.

(Slide #174)


RCS PRESSURE CONTROL
WITH 2100 PSIA SETPOINT
2545 2nd CODE SAFETY
2500 1st CODE SAFETY
2450 DIVERSE SCRAM TRIP
2400 DIVERSE SCRAM PRE-TRIP
2350 TRIP & PORV'S OPEN
2300 HI PRESS PRETRIP
PRESSURE DECREASING
2225 FULLY OPEN
SPRAY VALVE MODULATION BAND
2175 FULLY CLOSED
2145 HI PRESS ALARM
2115 PROP HTRS OFF
2100 SETPOINT
2075 BACKUP HEATERS OFF
PROP HTR MODULATION BAND
2080 LO PRESS ALARM
2075 PROP HTRS FULLY ON
ALL HEATERS FULLY ON
2060 BACKUP HTRS ON
PRESSURE INCREASING
TM/LP TRIP 1750
PPLS BLOCK PERMISSIVE 1700
LO LO ALARM and PPLS 1600

(Slide #204)

PRESSURIZER PRESSURE CONTROL BLOCK DIAGRAM



EO 1.3 **(Slide #135)**

Major Component Description

High Pressurizer Pressure Trip (TU-8)

The signal is provided by four pressurizer pressure transmitters (102 channels A, B, C and D).

NOTE: The RPS pressure transmitters do not input to the DSS. Separate transmitters (120 channels) are installed on the sensing lines.

The input signal is proportional to pressurizer pressure.

The instrument range is 1500 to 2500 psia.

This pressure signal used for three separate safety functions:

(1) Input to the Engineered Safeguards Control (ESC) System for generating the pressurizer pressure low signal (PPLS) at 1600 psia.

(2) Input for the TM/LP trip unit of the RPS.

(3) Input to the high pressurizer pressure trip unit of the RPS.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	029 G 2.1.30	
Level of Difficulty: 2	Importance Rating	4.4	

ATWS: Conduct of Operations: Ability to locate and operate components, including local controls

Question: 47

Given the following conditions:

- An Anticipated Transient without Scram (ATWS) occurred.
- When the Reactor was tripped, two Shutdown Group CEAs did NOT fully insert.
- HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve, will NOT open from the Control Room.

Which of the following is required to locally open HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve

Open HCV-268 breaker at _____(1)_____ located in _____(2)_____.

A. (1) MCC-3C2
(2) Corridor 4

B. (1) MCC-4A2
(2) Corridor 4

C. (1) MCC-4A2
(2) Corridor 26

D. (1) MCC-3C2
(2) Corridor 26

Answer: D

K/A Match:

Applicant must be able to direct local control of Emergency Boration Valve HCV-268.

Explanation:

A. Incorrect. Plausible because MCC-3C2 is the power supply to HCV-268. There are boric acid makeup valves (HCV-258) on the motor control center located in Corridor 4 (MCC-4A2).
B. Incorrect. Plausible because MCC-4A2 is the power supply to HCV-258, CH-11B Gravity Feed Valve, and the correct location of this MCC is listed.
C. Incorrect. Plausible because the breaker location is correct, however, HCV-268 is powered from MCC-3C2.
D. **Correct**. Corridor 26 is located in the Auxiliary Building at the 1007' elevation. HCV-268 is powered from MCC-3C2.

Technical Reference: (Attach if not previously provided including revision number)	AOP-03, Step 2.b, Rev. 6
	LP 4-43-1,Slide #268, Rev. 3

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 4-43-1, Chemical and Volume Control System-AON
EO 1.6c - **STATE** the location of the power supplies for each of the following components of the Chemical and Volume Control System: Boric acid motor operated valves.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	
55.41	6
55.43	

Open **ALL** of the following valves:

- HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
- HCV-265, CH-11A Gravity Feed Valve
- HCV-258, CH-11B Gravity Feed Valve

b.1 **(LOCAL) IF** HCV-268 did **NOT** open, **THEN** perform the following (Corridor 26):

1) Open MCC-3C2-C02, "EMERGENCY BORATION MOV HCV-268".

2) Manually open HCV-268, "CHARGING PUMPS CH-1A, B, C EMERGENCY SUCTION HEADER STOP VALVE".

EO *1.6c **(Slide #268)**

Major Component Description

Motor Operated Boric Acid Valves (HCV-258 , HCV-265 & HCV-268)

The motor operated boric acid valves are powered from MCC-3C2 (HCV-265 and HCV-268) and MCC-4A2 (HCV-258) in Corridor 26.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	038 EK1.03	
Level of Difficulty: 4	Importance Rating	3.9	

Steam Generator Tube Rupture: Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation.

Question: 48

Which of the following describes the desired method of Natural Circulation during a Steam Generator Tube Rupture?

A. Single phase Natural Circulation as it keeps the tube bundle region of the affected Steam Generator in a subcooled condition.

B. Dual phase Natural Circulation minimizes any slugs of water with reduced boron concentration in the affected loop due to backflow.

C. Single phase Natural Circulation limits backflow in the affected loop which enhances the cooldown.

D. Dual phase Natural Circulation because the latent heat of vaporization is more effective at avoiding stagnation during an asymmetric cooldown.

Answer: A

K/A Match:
Applicant requires knowledge of the operational implications of Natural Circulation during a Steam Generator Tube Rupture including methods of Natural Circulation cooling.

Explanation:

A. **Correct**. Single phase Natural Circulation (NC) implies that the RCS liquid maintains a subcooled state throughout the RCS, Loops, and SG tube bundles. Keeping the tube bundle in a subcooled condition will eliminate voiding that can disturb or stop NC flow.

B. Incorrect. Plausible because this condition will occur during Natural Circulation and is a concern should reactor coolant pumps be restarted due to a potential for a positive reactivity addition. Incorrect because dual phase NC flow does not improve this condition.

C. Incorrect. Plausible because single phase NC flow is desired and so is limiting backflow. Incorrect because backflow refers to the dilution of the RCS when SG water mixes with RCS water and reduces the boron concentration.

D. Incorrect. Plausible because avoiding stagnation during an asymmetric cooldown is a concern when on NC flow. Incorrect because dual phase NC would create vice avoid stagnation.

Technical Reference: TDB-EOP-04, Steps 18, 19, & 32 Bases, Rev. 28

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-15-33, Steam Generator Tube Rupture-Licensed Operator EO 3.5 - **STATE** from memory the four indications used to verify the development of Subcooled Natural Circulation

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 14
55.43 ______

TDB-EOP-04, Step 18 Bases

The EPG step was split into separate steps, one to commence depressurization and another to maintain RCS pressure within certain criteria. This was done to provide the operator with a specific instruction when it applied rather than early in the procedure. This justifies deviation 1.

The matching of RCS pressure and S/G pressure will minimize leakage between the RCS and the affected S/G. This is an important goal in the recovery process during a SGTR. This note stresses the fact that in order to meet this goal in the most efficient manner, RCPs should remain running. If the RCPs are not running, then it is important to maintain single phase natural circulation. Maintaining the RCP NPSH and a minimum RCS subcooling of 20°F will take priority over matching RCS pressure. This justifies deviation 2.

Boron dilution of the RCS would occur due to unborated secondary water flowing through the tube rupture into the RCS. However, under most circumstances, this dilution will not threaten the maintenance of adequate shutdown margin.

A key point in the strategy for the SGTR event involves maintaining or restoring forced circulation. However, maintaining adequate NPSH for RCP operation or at least 20°F subcooling may cause the operator to hold RCS pressure above secondary pressure. Maintaining RCP NPSH or providing adequate subcooling takes precedence over the procedural strategy of bringing primary pressure to the point where it will be approximately equal to secondary pressure.

During the forced circulation cooldown process, the isolated S/G may cool faster in the tube regions. The cooling of the isolated S/G steam space will significantly lag in the cooldown and cause the fluid in the lower regions to be subcooled. If the tube rupture is located in this subcooled region, as it is most likely to be, then the primary fluid can be at the same pressure as the secondary fluid and still be subcooled. However, the continued depressurization of the primary during the cooldown will now be limited by the ability to depressurize the isolated S/G.

During natural circulation cooldown conditions, the isolated S/G will not cool unless there is a transfer of mass in the isolated S/G. This complicates RCS pressure control during the cooldown. It is desirable to cool the RCS such that the tube bundle region of the affected S/G remains subcooled. Voiding in the tube bundle region can be expected and may result in the region becoming a

pressurizing source for the RCS. Maintaining the presence of subcooled liquid in the affected loop will be a complicated process under natural circulation conditions. Forced circulation conditions are much more desirable and if possible, should be maintained or restored. During natural circulation conditions, the cooldown and depressurization of the RCS will be limited to the operator's ability to control the conditions of the isolated S/G.

TDB-EOP-04, Step 19 Bases

CEN-152, Rev. 04 directs disabling RCPs in the affected loop if all RCPs have been stopped. This action minimizes the possibility of operator error in starting the wrong RCP first following a natural circulation cooldown. During a natural circulation cooldown, a slug of water with reduced boron concentration may collect in the affected loop, due to S/G backflow. If under these circumstances, the first RCP started is in the affected loop, a positive reactivity addition may occur. A list of specific RCPs in each loop is given to simplify the disabling of the correct loop. This meets the intent of the EPG step. This justifies deviation 1.

TDB-EOP-04, Step 32 Bases

An orderly cooldown is established to decrease the RCS temperature below the shutdown cooling temperature criteria. Forced circulation is preferred while conducting the cooldown to prevent secondary backflow from accumulating in the affected loop. If natural circulation is used, an asymmetric cooldown should be performed slowly (less than 30°F/hr) if possible. Proceeding slowly ensures that the two SGs remain thermodynamically coupled and are cooled together to the greatest extent possible. Cooldown of the upper section (secondary side) of the affected SG will lag the rest of the SG. If the coodown rate is too high, the SGs will uncouple and flow in the isolated loop will stagnate. If backflow occurs in this condition, unborated secondary water may accumulate in the affected loop.

The operator should continually monitor for stagnation of the affected loop. Once the cooldown has started and continued for at least 15 to 20 minutes, loop temperatures should be observed to make the determination. If flow is occurring in both loops, all loop temperatures should be stable or decreasing. If flow in the affected loop stagnates, temperatures will stop changing. Note that SI flow to the affected loop may cause cold leg temperatures to be colder through natural circulation. If stagnation occurs, the cooldown rate should be reduced or stopped until natural circulation flow is re-established as indicated by temperatures stable or decreasing.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	E05 EA1.1	
Level of Difficulty: 3	Importance Rating	3.9	

Steam Line Rupture-Excessive Heat Transfer: Ability to operate and / or monitor the following as they apply to the Excess Steam Demand: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Question: 49

Given the following condition:

- A Steam Line Break downstream of the Main Steam Isolation Valves occurred.

Which of the following conditions terminates the event without Operator action?

A. Containment pressure rises to 3.5 psig.

B. Both Steam Generator pressures lower to 485 psia.

C. Reactor Coolant System pressure lowers to 1455 psia.

D. Both Steam Generator narrow range levels lower to 27%.

Answer: B

K/A Match:
Applicant must be able to monitor Control Room instrumentation and determine the impact on plant conditions based on those instrument settings.

Explanation:

A. Incorrect. Plausible because a Containment Pressure High Signal (CPHS) or Steam Generator Low Signal (SGLS) will generate a SGIS. Incorrect because this does not occur until 5 psig.
B. **Correct**. A Steam Generator Low Signal (SGLS) generated from both Steam Generators less than 500 psia in turn caused a Steam Generator Isolation Signal (SGIS) which secures steaming and feeding of the affected Steam Generator.
C. Incorrect. Plausible because a Pressurizer Pressure Low Signal (PPLS) has actuated. Incorrect because the SGIS is generated from either a SGLS or CPHS.
D. Incorrect. Plausible if thought that some automatic action occurred associated with narrow range level of 27%. This value is used during a UHE to commence steaming the unaffected SG to prevent a Pressurized Thermal Shock condition.

Technical Reference: LP 7-18-15, Pages 10-11, Rev. 20
(Attach if not previously provided including revision number)
LP 7-12-14, Slide #17, #214, #223, #225, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Control System- LO
EO 1.5 - EXPLAIN the functions performed by each Engineered Safeguards Control signal.

Lesson Plan 7-18-15, Uncontrolled Heat Extraction-Licensed Operator
EO 1.6 - **STATE** from memory the Exit Conditions for EOP-05, UHE.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43

EO *1.2 **(Slide #214)**

Steam Generator Low (pressure) Signal (SGLS)

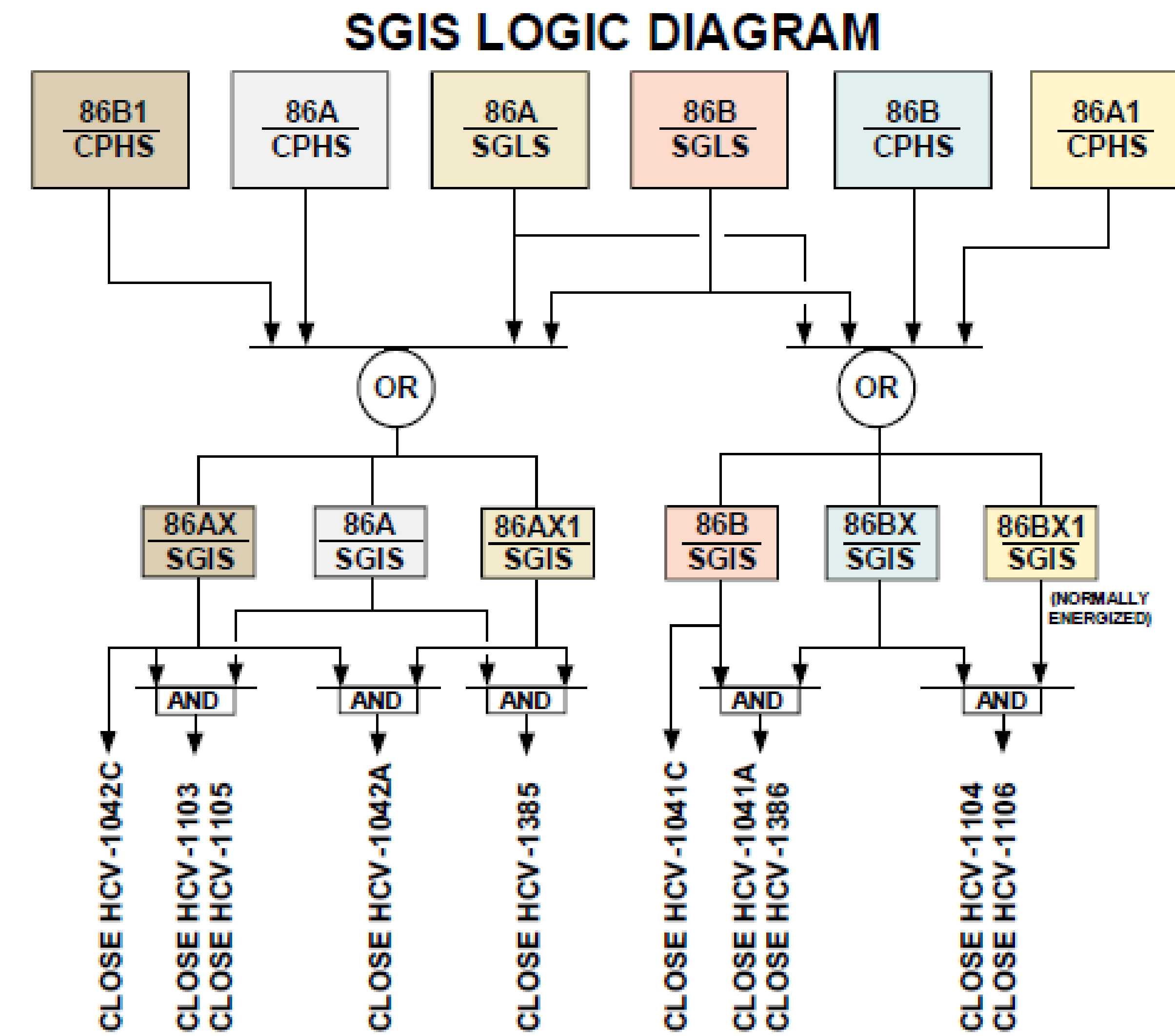
SGLS Matrix Relays

Each S/G pressure meter energizes one self-resetting matrix relay, located in CB-4, when pressure decreases to 500 psia.

Each matrix relay closes two contacts in the two-of-four coincidence logic matrix for SGLS.

This slide shows the relationship between the meters, matrix relay contacts, logic channels, and power sources.

(Slide #223)


SGIS LOGIC DIAGRAM
86B1 CPHS
86A CPHS
86A SGLS
86B SGLS
86B CPHS
86A1 CPHS
OR
OR
86AX SGIS
86A SGIS
86AX1 SGIS
86B SGIS
86BX SGIS
86BX1 SGIS
(NORMALLY ENERGIZED)
AND
AND
AND
AND
AND
CLOSE HCV-1042C
CLOSE HCV-1103
CLOSE HCV-1105
CLOSE HCV-1042A
CLOSE HCV-1385
CLOSE HCV-1041C
CLOSE HCV-1041A
CLOSE HCV-1386
CLOSE HCV-1104
CLOSE HCV-1106

EO *1.5 **(Slide #225)**

Steam Generator Isolation Signal (SGIS)

SGIS Relays (86A/SGIS, 86AX/SGIS, 86AX1/SGIS, 86B/SGIS, 86BX/SGIS, and 86BX1/SGIS)

Plant Modification MR-FC-92-044 revised the SGIS circuit so that a single relay failure would not cause the valves to reposition. With the exception of HCV-1041C and HCV-1042C (which are normally closed valves) it requires TWO self-resetting relays to DE-ENERGIZE to cause valves to change position.

SGIS relays affect the following valves:

(1) 86A/SGIS and 86AX/SGIS will close HCV-1103 and HCV-1105

(2) 86A/SGIS and 86AX1/SGIS will close HCV-1385

(3) 86AX/SGIS and 86AX1/SGIS will close HCV-1042A

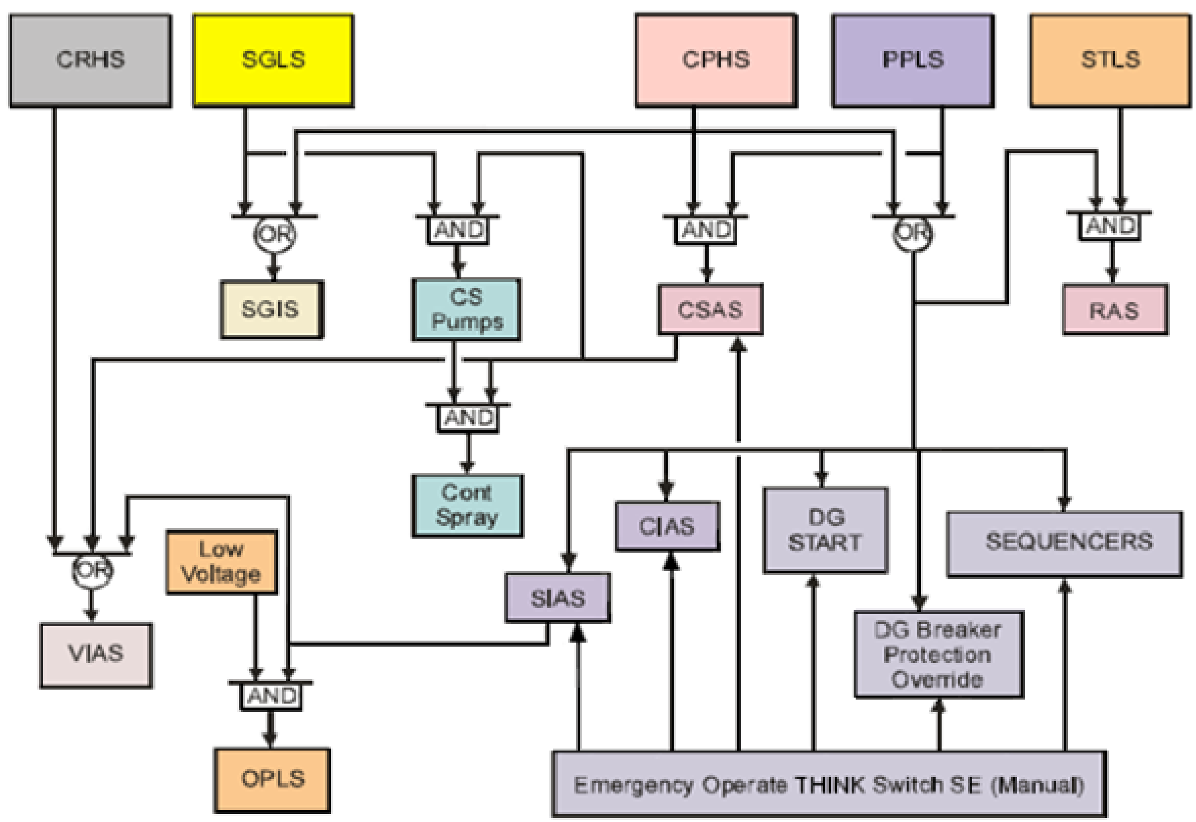
(4) 86B/SGIS and 86BX/SGIS will close HCV-1041A and HCV-1386

(5) 86BX/SGIS and 86BX1/SGIS will close HCV-1104 and HCV-1106

(6) 86AX/SGIS closes HCV-1042C

(7) 86B/SGIS closes HCV-1041C

(Slide #17)



B. Entry Conditions

1. SPTA have been performed

AND

2. Any of the following:

a. Lowering pressure in one or both SGs, possible SGLS

b. SGLS has initiated

c. Lowering Tavg

d. Rise in MFW flows

e. Possible CIAS caused by high containment pressure with rising in containment temperature, humidity and sump level

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/28/15	Tier #	1	
Change: 0	Group/Category #	1	
	K/A #	E06 EK3.3	
Level of Difficulty: 3	Importance Rating	3.7	

Loss of Main Feedwater: Knowledge of the reasons for the following responses as they apply to the Loss of Feedwater: Manipulation of controls required to obtain desired operating results during abnormal and emergency situations.

Question: 50

Given the following conditions:

- Plant tripped from 100% power due to a Loss of Offsite Power.
- Following the reactor trip, DC Bus 1 was lost.
- FW-6, FW-10 and FW-54, Auxiliary Feedwater Pumps, failed to start and entry into EOP-20, Functional Recovery, is in progress.
- While in EOP-20, PB-2/1A1-1A3 and ATD-D1, DIESEL D1 125 VDC MANUAL TRANSFER SWITCH pushbuttons have been pressed.

Which of the following is the reason for performing this action?

A. Allows FW-6, Auxiliary Feedwater Pump to be started by restoring control power.

B. Bypasses the 43/FW interlock to allow starting a Condensate Pump.

C. Allows FW-4A, Main Feedwater Pump to be restarted by restoring control power.

D. Bypasses the 43/FW interlock to allow starting FW-6, Auxiliary Feedwater Pump.

Answer: A

K/A Match:
Applicant must know the reason why Loss of All Feedwater was entered and the required switch positions to restore feedwater flow.

A. **Correct**. The Loss of Offsite Power and DC Bus 1 have rendered FW-6, AFW Pump without control power and unable to be started either remotely or locally. Depressing PB-2 and ATD-D1 restores control power and allows FW-6 to be restarted.
B. Incorrect. Plausible because before FW-6 can be started from CB-10, the 43/FW switch must be in OFF. Incorrect reason for depressing PB-2 and ATD-D1.
C. Incorrect. Plausible because this does restore control power to FW-4A. Incorrect because Offsite Power has not been restored and this pump would not be started.
D. Incorrect. Plausible because before FW-6 can be started from CB-10, the 43/FW switch must be in OFF. Incorrect reason for depressing PB-2 and ATD-D1.

Technical Reference: (Attach if not previously provided including revision number)
EOP-20, Continuing Actions for MVA-AC, Step 50, Rev. 28
OI-AFW-1, Attachment 2, Step 4, Rev. 83
LP 7-18-16, Pages 13 and 14, Rev. 15

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-16, Loss of All Feedwater-Licensed Operator
EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of a LOAF.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ____

50. Transfer DC control power to emergency by performing the following:

a. Press PB-2/1B3A-4A-MTS, "MANUAL TRANSFER PUSHBUTTON 1B3A-4A-MTS EMERG. SOURCE" push button (East Switchgear Room).
b. Press PB-2/1B3C-4C-MTS, "MANUAL TRANSFER PUSHBUTTON 1B3C-4C-MTS EMERG. SOURCE" push button (East Switchgear Room).
c. Press PB-2/1A1-1A3-MTS, "MANUAL TRANSFER PUSHBUTTON 1A1-1A3-MTS EMERGENCY SOURCE" push button ("1A1-1A3 AUX POWER COMPARTMENT").
d. Press ATD-D1, "DIESEL D1 125 VDC MANUAL TRANSFER SWITCH" "EMERGENCY" push button (D-1 Room, North Wall).

4. Step 11 and 11.1

a. ACTION - Initiate Auxiliary Feedwater using FW-6 or FW-10 per Attachment HR-17, FW-6/FW-10 Operation.

(1) If FW-6 and FW-10 are not available, then initiate AFW using FW-54 per Attachment HR-16, FW-54 Operation.

b. BASIS - If efforts to establish MFW have failed, then the operator is directed to establish AFW using FW-6 or FW-10. Contingency actions are provided to feed the S/Gs using FW-54 via the feedrings or AFW nozzles.

Attachment 2, Auxiliary Feedwater Switch Position Description

4. 43/FW, Cond & FW Pumps Transfer SW:

- Aligns nine pumps for auto operation in Auto
- Must be placed in Off for manually starting FW-6 from CB-10

i

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/22/15	Tier #	1	
Change: 0	Group/Category #	1	
	K/A #	055 EA2.05	
Level of Difficulty: 3	Importance Rating	3.4	

Station Blackout: Ability to determine or interpret the following as they apply to a Station Blackout: When battery is approaching fully discharged.

Question: 51

Given the following conditions:

- A Station Blackout occurred at 0600.
- EOP-07, Station Blackout, was entered and MVA-24, Minimizing DC Loads, has been performed.
- Estimates are that Onsite Power will be restored at 2000.
- No estimate is available for return of Offsite Power.

Which of the following describes the LATEST time that required control room instrumentation can be relied upon?

A. 1400 hours due to loss of 120 VAC.

B. 0800 hours due to loss of 125 VDC.

C. 1000 hours due to prolonged loss of buses 1A3 and 1A4.

D. 1400 hours due to loss of Instrument Bus IB-3.

Answer: A

K/A Match:

Applicant must know the credited time for instrument control power and station batteries following minimizing DC Loads within the required time frame.

Explanation:

A. **Correct**. MVA-24, Minimizing of DC Loads within 15 minutes allows the batteries to continue powering control and instrumentation devices necessary for Reactor shutdown for up to 8 hours. Failure to perform this action could result in a loss of 120 VAC prior to 1400.

B. Incorrect. Plausible because part of Minimizing DC Loads includes waiting for the Turbine to stop rolling so that DC Lube Oil Pumps can be secured. This is done before 2 hours have elapsed since the loss of Battery Chargers. Incorrect because the batteries are rated for an 8 hour discharge. Even if Lube Oil Pumps were not secured, 125 VDC would still be available at 0800.

C. Incorrect. Plausible because a prolonged loss of buses 1A3 and 1A4 for 4 hours causes station personnel to declare a General Emergency per EAL SG1, based on the station being designated as a 4-hour coping plant. Incorrect because even though the loss of power presents risk, for core cooling, DC power is credited for 8 hours for instrumentation and controls.

D. Incorrect. Plausible because IB-3 provides power to the Plant Data Network (PDN), Distributed Control System (DCS), and Data Acquisition System (DAS) which has both 120 VAC and 125 VDC components. Incorrect because these systems are not required for reactor shutdown, and IB-3 batteries are rated for only 2 hours.

Technical Reference: EOP/AOP Attachment MVA-24, Step 1 NOTE, Steps 3 & 4, Rev. 1
(Attach if not previously provided including revision number) LP 7-13-4, Slide #26 & #70, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-17, EOP-07, Station Blackout-Licensed Operator
EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of a SBO.

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

Question History: Last NRC Exam 2001 NRC Exam

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

MVA-24, Step 1

NOTE

Performing the following will allow up to 8 hours operation of the control and instrumentation devices required for Reactor shutdown without Battery Charger operation.

1. Reduce DC loads by performing the following within 15 minutes of the loss of Battery Chargers:

A. Place BOTH of the following DC Bus 2 breakers in "OFF" (West Switchgear Room):

- EE-8G-CB12, "400 CYCLE INVERTER EE-21"
- EE-8G-CB8, "EMERGENCY LIGHTING PNL ELP-2 TRANSFER SWITCH"

MVA-24, Steps 3 & 4

NOTE

To ensure adequate battery capacity, the DC Oil Pump should be stopped as soon as the turbine stops rolling, which will occur in approximately one hour.

3. WHEN the turbine has stopped rolling, THEN stop LO-4, DC Oil Pump.

4. BEFORE two hours has elapsed since the loss of Battery Chargers, THEN reduce DC loads by performing the following:

B. Ensure BOTH of the following breakers are closed (AI-42A):

- I-BUS-I1-1, "INSTRUMENT BUS 1 MAIN BREAKER"
- "CIRCUIT #1 AI-53 NORM FEED"

EO 1.1 **(Slide #26)**

Major Component Description

Batteries

Two batteries, each is rated for 2060 amp-hours at the eight hour discharge rate.

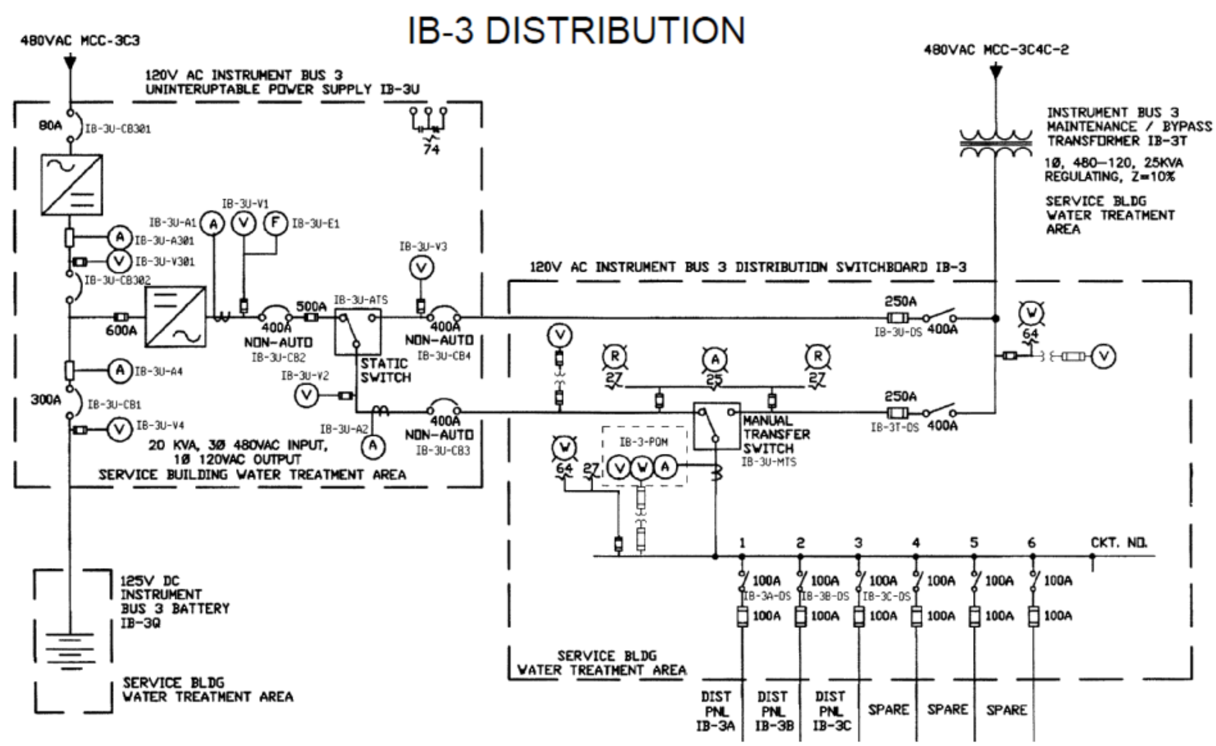
58 cells – two cells removed (originally 60 cells) to reduce operating voltage.

The batteries are adequate to supply all necessary loads up to eight hours without charger operation.

Operation for eight hours without a charger is accomplished by a scheduled deletion of DC bus loads.

Batteries normally provide current to the DC bus only when starting heavy loads, or during automatic transfers of Buses 1A3/1A4.

(Slide #70)



Bank Question:

The following plant conditions exist:

- ALL offsite and onsite power was lost at 0600 hours
- At 0800 hours the new operating crew realized the failure of the previous crew to minimize DC bus loads as per steps 15.1b and 22 of EOP-07, "Station Blackout."
- ALL other steps had been performed
- Estimates are that onsite power will be restored between 1600 and 2000 hours.
- No estimate is available for return of offsite power.

Which ONE of the following will be the consequence of a failure of either crew to perform step 15.1b (minimize DC bus loads) of EOP-07, "Station Blackout?"

A. 120 VAC could be lost prior to 1400 hours.

B. Loss of DC control power to all 4.16 KV switchgear could be lost prior to 0900 hours.

C. 125 VDC could be lost prior to 0900 hours.

D. No adverse affects should be seen prior to onsite power being available.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	056 AA2.02	
Level of Difficulty: 3	Importance Rating	3.5	

Loss of Offsite Power: Ability to determine and interpret the following as they apply to the Loss of Offsite Power: ESF load sequencer status lights.

Question: 52

Given the following conditions:

Following a loss of offsite power, 4.16 KV Bus 1A3 Sequencer Status lights on Panel S1-1 are as follows:

- Red light is OUT.
- Green light is ON.

Which of the following are the conditions and MINIMUM bus voltage that will generate this Sequencer Status light indication?

A. (1) A degraded voltage condition exists.
(2) Bus 1A3 voltage is 4000 Volts.

B. (1) The degraded voltage condition has just cleared.
(2) Bus 1A3 voltage is 4000 Volts.

C. (1) The degraded voltage condition has just cleared.
(2) Bus 1A3 voltage is 4050 Volts.

D. (1) A degraded voltage condition exists.
(2) Bus 1A3 voltage is 4050 Volts.

Answer: A

K/A Match:

Applicant must know the condition associated with the OPLS Sequencer Status lights including the condition that will cause them to actuate/change.

Explanation:

A. **Correct**. Red lights will go OUT and green lights will turn ON to indicate degraded voltage. This is less than the degraded voltage setpoint for Bus 1A3.

B. Incorrect. Plausible because this is degraded voltage less than setpoint. Incorrect because the light condition is reversed.

C. Incorrect. Plausible if thought this is the degraded voltage setpoint. Incorrect because the light condition is reversed.

D. Incorrect. Plausible because the light condition to indicate a degraded voltage is correct. Incorrect because this is not a degraded voltage condition.

Technical Reference: (Attach if not previously provided including revision number)
ARP-AI-30A/A33-2, Window F-3, Rev. 25
ARP-AI-30B/A34-2, Window F-2, Rev. 26
LP 7-12-14, Slide #22, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-14, Engineered Safeguards Control-Licensed Operator
EO 1.5 - **EXPLAIN** the functions performed by each Engineered Safeguards Control signal.
EO 2.1 - Given the control boards or simulator, **EXPLAIN** the Control Room indications associated with ESC.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Panel: **AI-30A**	Annunciator: **A33-2**	Window: **F-3**

OFFSITE POWER LOW SIGNAL A OR C UNDERVOLTAGE

SAFETY RELATED

OPLS-A OR C SENSORS ACTUATION

Tech Spec References: 2.15

Initiating Device 27-74 / 1A3		Power AI-40A
Initiating Device 27-74 / T1A1		Power AI-40C
Initiating Device 27-74 / T1A3	Setpoint <4012 Volts	Power AI-40C

OPERATOR ACTIONS

1. Check 4160V Bus 1A3 voltage (ERF page 310).

2. IF Bus 1A3 is less than 4012 Volts, THEN notify System Operations.

   2.1 Consider the transfer of Bus 1A3 feed to its alternate source per OI-EE-1.

3. IF Bus 1A3 is greater than 4012 Volts, THEN initiate notification to the Work Week Manager.

Panel: **AI-30B**	Annunciator: **A34-2**	Window: **F-2**

OFFSITE POWER LOW SIGNAL B OR D UNDERVOLTAGE

SAFETY RELATED

OPLS-B OR D SENSORS ACTUATION

Tech Spec References: 2.15

Initiating Device 27-74/1A4		Power AI-40B
Initiating Device 27-74/T1A2		Power AI-40D
Initiating Device 27-74/T1A4	Setpoint < 4014 Volts	Power AI-40D

OPERATOR ACTIONS

4. Check 4160V Bus 1A4 voltage (ERF page 310).

5. IF Bus 1A4 is less than 4014 Volts, THEN notify System Operations.

   5.1 Consider the transfer of Bus 1A4 feed to its alternate source per OI-EE-1.

6. IF Bus 1A4 is greater than 4014 Volts, THEN initiate notification to the Work Week Manager.

EO *1.5 **(Slide #252)**

Offsite Power Low Signal (OPLS)

OPLS Sensor Channels

Also, after the relay times out, status lights on the sequencer sections of AI-30A(B) will change.

RED lights will go OUT and GREEN lights will turn ON to indicate degraded voltage.

Sensor circuit A controls the status lights "4.16 KV BUS 1A3" on sequencer panel S1-1.

Sensor circuit B controls the status lights "4.16 KV BUS 1A4" on sequencer panel S2-1.

(Slide #252)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 2	Group/Category #	1	
	K/A #	057 G 2.2.44	
Level of Difficulty: 3	Importance Rating	4.2	

Loss of Vital AC Instrument Bus: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question: 53

Given the following conditions:

- A Plant heatup is in progress when a Loss of Instrument Bus AI-40C occurs.
- AOP-16, Loss of Instrument Bus, was entered.
- Reactor Coolant System (RCS) temperature is 325°F.
- RCS pressure is 850 psia.

Which of the following is a concern at this time?

A. Reactor Coolant Pump RC-3A & RC-3C Seal & Motor Cooling is lost.

B. Inability to control Component Cooling Water flow to Containment Cooling Coils.

C. Low Temperature Overpressure Protection actuates if another channel trips.

D. Letdown isolates if Pressurizer Level Controller Channel Y is in service.

Answer: C

K/A Match:

Applicant must be able to interpret control room indications following a loss of an instrument bus. They must then use this information to determine how this lineup affects their ability to control or respond to plant component status changes.

Explanation:

A. Incorrect. Plausible because the 4 valves (HCV-442/444/446/448) associated with seal and motor cooling to each RCP lose power. Incorrect because the valves fail open.

B. Incorrect. Plausible because Loss of a Vital Instrument Bus will render Containment Cooling Coil Outlet Isolation Valve Controllers HCV-400C/401C/402C/403C inoperable. Incorrect because this occurs with a Loss of Instrument Bus AI-40A.

C. **Correct**. With one channel already tripped, Letdown Temperature Overpressure Protection could actuate if another channel trips. This is the reason the crew would consider closing the PORV Block Valves at Step 13.

D. Incorrect. Plausible because this would occur if Instrument Bus AI-40B were lost. Incorrect because this failure does not occur with AI-40C.

Technical Reference: (Attach if not previously provided including revision number)	AOP-16, Section IV, Steps 6 & 13 NOTE, Rev. 20
	AOP-16, Section III, Step 8 NOTE, Rev. 20
	AOP-16, Section II, Step 5 NOTE, Rev. 20

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-16, Loss of Instrument Bus-Licensed Operator
EO 1.2 - **DESCRIBE** how the plant responds to a loss of instrument bus power in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	7
	55.43	

AOP-16, Section IV, Step 6 NOTE

NOTE

Upon loss of Instrument Bus C, **ALL** of the following instrumentation or equipment associated with the **Vital Auxiliaries Safety Function** is affected as follows:

- "RC-3A PUMP SEAL CLR AC OUTL HCV-442" fails open (AI-45)
- "RC-3A MOTOR OIL CLR AC OUTL HCV-446" fails open (AI-45)
- "RC-3C PUMP SEAL CLR AC OUTL HCV-444" fails open (AI-45)
- "RC-3C MOTOR OIL CLR AC OUTL HCV-448" fails open (AI-45)
- "CC HT EXCH AC-1C RW OUTLET TEMP TIC-2887" is inoperable
- "CCW SURGE TK AC-2 LEVEL LIC-2801" is inoperable
- ERF computer point L2801, "CCW Surge Tank AC-2 Level" is inoperable and alarms
- "CCW SURGE TK AC-2 PRESS PIC-2802" is inoperable

6. Ensure CCW system operation by satisfying **BOTH** of the following conditions:
   - At least one CCW Pump, AC-3A/B/C, is running
   - CCW pressure is greater than or equal to 60 psig

6.1 **IF** the CCW System is **NOT** operational, **THEN** initiate a Reactor shutdown by performing the following:

a) Trip the Reactor.

b) IMPLEMENT EOP-00, Standard Post Trip Actions.

AOP-16, Section IV, Step 13 NOTE

NOTES

1. Only one additional channel trip is needed to actuate the PORVs, even if the channel in trip is bypassed.

2. When RCS Heatup or Cooldown is in progress, the PORVs are the primary means of Low Temperature Overpressure Protection.

3. Closing the PORV block valves requires entry into Tech Spec 2.1.6.

13. Consider closing **BOTH** of the PORV Block Valves:
   - HCV-150
   - HCV-151

AOP-16, Section III, Step 13 NOTE

NOTE

Upon loss of Instrument Bus B, **ALL** of the following instrumentation or equipment associated with the **RCS Inventory Control Safety Function** is affected as follows:

- Letdown may be isolated, depending on the running Charging Pump(s)
- CH-1B, Charging Pump, is inoperable
- PZR Level Controller Channel Y is inoperable

8. Place HC-101, "PRESSURIZER LEVEL CHAN SELECTOR SWITCH", in "CHAN X".

AOP-16, Section II, Step 5 NOTE

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Vital Auxiliaries Safety Function** are inoperable:

- "WEST RW SUPPLY HEADER FLOW FIC-2891" indicator
- "CC HT EXCH AC-1A RW OUTLET TEMP TIC-2885"
- "CNTMT CLG COIL VA-1A OUTLT ISOL VLV CNTRLR HCV-400C"
- "CNTMT CLG COIL VA-1B OUTLT ISOL VLV CNTRLR HCV-401C"
- "CNTMT CLG COIL VA-8A OUTLT ISOL VLV CNTRLR HCV-402C"
- "CNTMT CLG COIL VA-8B OUTLT ISOL VLV CNTRLR HCV-403C"

5. Ensure CCW System operation by satisfying **BOTH** of the following conditions:

   - At least one CCW Pump, AC-3A/B/C, is running
   - CCW pressure is greater than or equal to 60 psig

5.1 **IF** the CCW System is **NOT** operational, **THEN** initiate a Reactor Shutdown by performing the following:

a) Trip the Reactor.

b) IMPLEMENT EOP-00, Standard Post Trip Actions.

c) Stop all RCPs.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	058 AK1.01	
Level of Difficulty: 4	Importance Rating	2.8	

<u>Loss of DC Power</u>: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation.

Question: 54

Given the following conditions:

- Annunciator CB20/A15, Window A-5 – BATTERY CHARGER #1 TROUBLE, is in alarm.
- Annunciator CB20/A15, Window C-3 – DC BUS #1 LOW VOLTAGE, is in alarm.
- Five (5) minutes ago, DC Bus #1 voltage rose to 155 VDC and remained at that voltage for one (1) minute.
- DC Bus #1 voltage is now 103 VDC and lowering.
- There are <u>NO</u> other Control Room annunciators in alarm at this time.

Which of the following is the cause of these conditions?

DC Bus #1 Battery...

A. Breaker CB-1 opened.

B. Charger #1 shutdown on low voltage.

C. Charger #1 AC input to the Charger failed.

D. Charger #1 shutdown on high voltage.

Answer: D

<u>K/A Match:</u>
Applicant must be familiar with Battery Charger indications and operation.
Explanation:

A. Incorrect. Plausible because DC voltage is low. Incorrect because if the Battery Breaker opened DC voltage would be 0.
B. Incorrect. Plausible because DC voltage is low. Incorrect because DC Bus 1 voltage rose to 155 VDC before it lowered to 103 VDC.
C. Incorrect. Plausible because DC voltage is low.
D. **<u>Correct</u>**. Malfunction of the Battery Charger caused it to shutdown on high voltage.

Technical Reference: ARP-CB-20/A15, Windows A-5 & C-3, Rev. 42
(Attach if not previously provided including revision number) LP 7-13-4, Slides #23, #33, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-4, 125 VDC and 120 VAC Distribution-Licensed Operator
EO 1.2 - **EXPLAIN** the operation of each major component during all modes of operation.
EO 1.4 - **EXPLAIN** the Control Room indications for the systems and LIST the normal values for these indications.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 7
55.43 ______

Panel: **CB-20**	Annunciator: **A15**	Window: **A-5**

BATTERY CHARGER NO. 1 TROUBLE

SAFETY RELATED

BATTERY CHARGER #1 TROUBLE

Tech Spec References: 2.7

Initiating Device DSH	Setpoint >140 VDC	
Initiating Device DSL	Setpoint <125 VDC	Power EE-8C
Initiating Device CFA	Setpoint 0 VDC	Power MCC 3B1

OPERATOR ACTIONS

1. Dispatch operator to check the following:
   - EE-8C-CB1, Batt Charger Number 1 AC Input Breaker, closed (East Switchgear Room)
   - EE-8C, 125V DC Battery Charger Number 1, DC Voltmeter (East Switchgear Room)
   - MCC-3B1-C2L, EE-8C Battery Charger Number 1, closed (East Electrical Penetration Room)
2. IF MCC-3B1-C2L is tripped, THEN attempt to close.
3. IF EE-8C has no output, THEN place EE-8E, 125 V DC Battery Charger Number 3, in service for EE-8F, 125V DC Number 1 Main Distribution Panel, per OI-EE-3.
4. IF DC output voltage is less than 125 VDC, THEN initiate notification to the Work Week Manager of battery charger malfunction.
5. Refer to Technical Specification 2.7.

PROBABLE CAUSES

- DC output voltage is high or low
- AC input to the Charger has failed
- Battery Charger is overloaded

Panel: **CB-20**	Annunciator: **A15**	Window: **C-3**

DC BUS 1 LOW VOLTAGE

Page 1 of 2

DC BUS #1
LOW VOLTAGE

SAFETY RELATED

Tech Spec References: 2.7

Initiating Device 27/DC-BUS 1   Setpoint <125 VDC   Power DC Bus 1

OPERATOR ACTIONS

1. Check V/DC-BUS-1, AI-41A DC Bus 1 voltage indication. (AI-41A)

2. IF V/DC-BUS-1 is less than 105V DC, THEN IMPLEMENT AOP-16.

3. Dispatch operator to the following:

   3.1 IF EE-8C, 125V DC Battery Charger Number 1 in service, THEN check the following:

   - EE-8C-CB2, Batt Charger Number 1 DC Output Breaker (East Switchgear Room)
   - EE-8C DC Voltmeter (East Switchgear Room)
   - EE-8C DC Ampmeter (East Switchgear Room)
   - EE-8F-CB3, Batt Charger 1, EE-8C, closed (East Switchgear Room)

   3.2 IF EE-8E, 125V DC Battery Charger Number 3 in service, THEN check the following:

   - EE-8E-CB2, Batt Charger Number 3 DC Output Breaker (West Switchgear Room)
   - EE-8E DC Voltmeter (West Switchgear Room)
   - EE-8EC DC Ampmeter (West Switchgear Room)
   - EE-8F-CB2, Batt Charger 3, EE-8E, closed (East Switchgear Room)

(continued)

PROBABLE CAUSES

- Battery Charger output voltage low
- Breaker open between the Battery Charger and DC Bus
- High load on the Battery Charger

EO 1.4 **(Slide #23)**

Major Component Description

Battery Chargers

NOTE: Use a current revision the ARPs to review operator actions.

Annunciators at A15 on CB-20 alarm to indicate "Battery Charger Trouble".

(a) Probable causes:

(1) DC output voltage is high or low (>140V or low – no setpoint).

(2) AC input to the charger has failed (low voltage)

(3) Battery charger is overloaded

(LC) NOTE: Each charger has an alarm permissive (normal/inhibit) switch to prevent this alarm when the charger is secured.

EO 1.4 **(Slide #33)**

Major Component Description

DC Buses (panels EE-8F/8G)

Alarms on CB-20

DC BUS #1 (#2) LOW VOLTAGE

(a) Alarm on annunciator A15 (A19).

(b) Setpoint – <125 VDC

(c) At <105V, go to AOP-16, "

(d) Probable causes:

(1) Battery charger output voltage low

(2) Breaker open between charger and bus

(3) High load on the battery charger

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 10/08/15	Tier #	1	
Change: 2	Group/Category #	1	
	K/A #	062 AK3.02	
Level of Difficulty: 4	Importance Rating	3.6	

Loss of Nuclear Service Water: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

Question: 55

Given the following conditions:

- Plant is at 100% power.
- AC-10A and AC-10D, Raw Water Pumps, are RUNNING.
- AC-10B and AC-10C, Raw Water Pumps, are in STANDBY.
- Buses 1A3 and 1A4 remained energized.

How will the Raw Water System respond following an Engineered Safeguards Features (ESF) actuation?

With NO operator action, AC-10A and AC-10D…

A. …continue to run. AC-10B and AC-10C remain in Standby.

B. …continue to run. AC-10B and AC-10C start.

C. …are load shed. All four Raw Water Pumps are started by ESF Sequencers.

D. …are load shed and restarted by ESF Sequencers. AC-10B and AC-10C remain in Standby.

Answer: B

K/A Match:

Applicant must understand how the Raw Water Pumps respond to a ESF Sequencer actuation and why they are designed that way.

Explanation:

A. Incorrect. Plausible because AC-10A and AC-10D continue to run. Incorrect because AC-10B and AC-10C will receive a ESF Sequencer start signal.

B. **Correct**. Per the attached Logic Diagram, 3 conditions will start a Raw Water Pump. 1.) Placing the control switch in START, 2.) ESC Sequencer START, or 3.) TRIP of a running RW Pump (simplified explanation; refer to Logic Diagram). The Load Shed signal does not affect a running pump because of the logic associated with the ESF Sequencer and the installed "anti-pumping" relay (prevents a breaker with a "locked in" START signal and a TRIP signal from opening and closing repeatedly). This is the reason that AC-10A and AC-10D continue to run while AC-10B and AC-10C start on the ESF Sequencer.

C. Incorrect. Plausible because all 4 Raw Water Pumps will end up running. Incorrect because AC-10A and AC-10D are not load shed.

D. Incorrect. Plausible if thought that a running pump would be load shed when the ESF Sequencer fires.

Technical Reference: LP 7-11-19, Slides #31 & #33, Rev. 1

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-19, Raw Water System-Licensed Operator
EO 1.5 - **EXPLAIN** the automatic start features associated with the Raw Water Pumps.

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

Question History: Last NRC Exam 2002 NRC Exam

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

10 CFR Part 55 Content:	55.41	7
	55.43	

EO *1.5 **(Slide #31)**

Major Component Description

Raw Water Pumps (AC-10A/B/C/D)

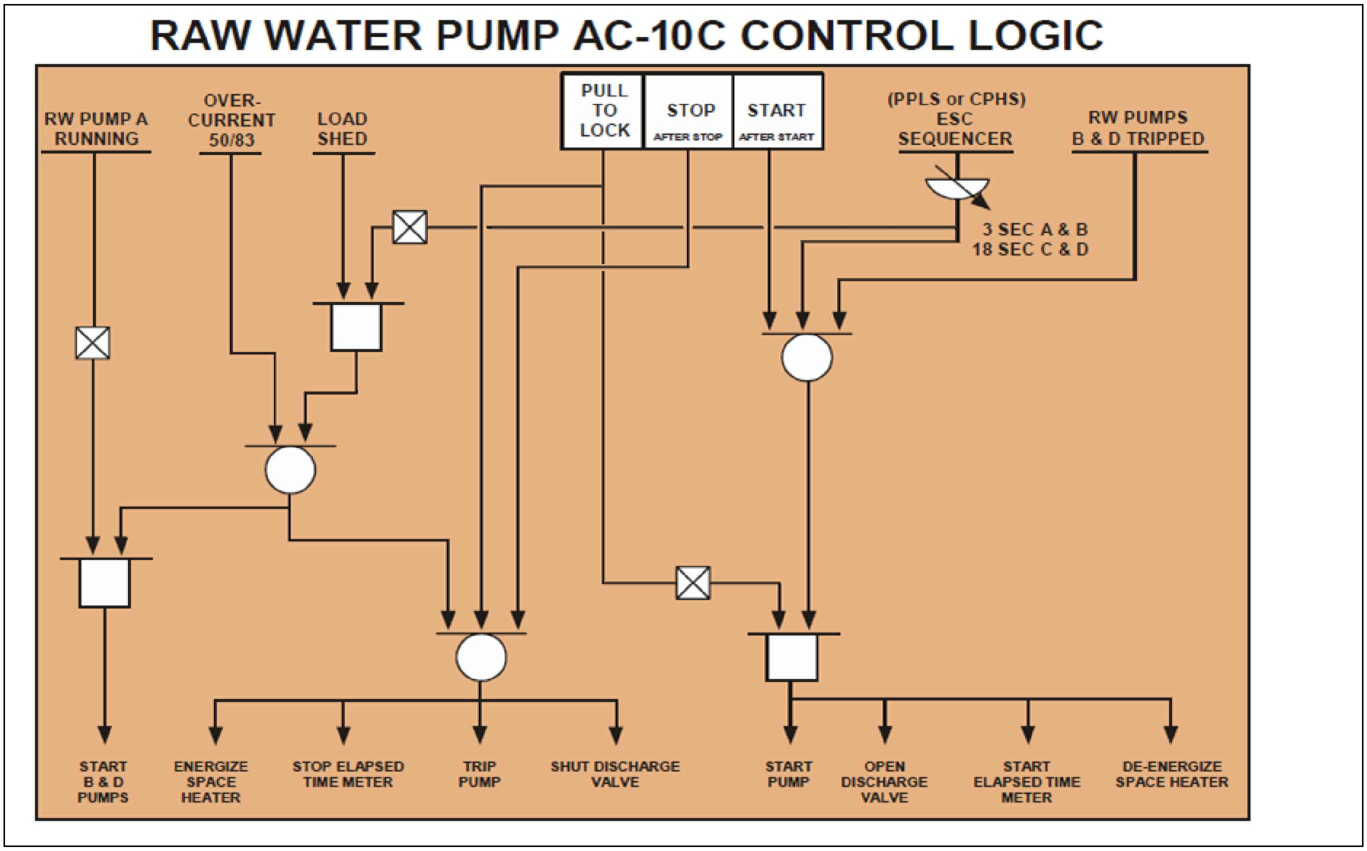
Auto Start Features - The RW pumps will auto start on the following signals:

(LC) A safeguards actuation signal (either PPLS or CPHS) causes sequencers to start all four pumps.

Pumps A and B will start after a 3 second time delay.

Pumps C and D will start after a 18 second time delay.

(Slide #31)



EO *1.5 **(Slide #33)**

Major Component Description

Raw Water Pumps (AC-10A/B/C/D)

Auto Start Features

ANTI-PUMP DEVICE

The RW pumps have an “anti-pump” relay installed in the breaker to ensure that a breaker with a “locked-in” start signal and a “trip” signal will not attempt to close and then reopen the breaker repeatedly.

This feature is to prevent the breaker from destroying itself by repeated cyclings.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	1	
Change: 1	Group/Category #	1	
	K/A #	065 AA2.08	
Level of Difficulty: 3	Importance Rating	2.9	

Loss of Instrument Air: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment.

Question: 56

What effect will a total Loss of Instrument Air header pressure have on Safety Injection Refueling Water Tank (SIRWT) level indication?

Assume that there is NO actual change in SIRWT level and that the loss of pressure is of short enough duration that the credited local Air Accumulators maintain pressure.

A low level will…

A. …NOT be indicated.
Low level alarms will NOT be received.
Safety Tank Low Signal actuation will occur.

B. …be indicated.
Low level alarms will be received.
Safety Tank Low Signal actuation will NOT occur.

C. …be indicated.
Low level alarms will NOT be received.
Safety Tank Low Signal actuation will NOT occur.

D. …NOT be indicated.
Low level alarms will be received.
Safety Tank Low Signal actuation will occur.

Answer: B

K/A Match:

Applicant must know the 2 forms of level indication on the SIRWT including which have a backup accumulator when a Loss of Instrument Air occurs.

Explanation:

A. Incorrect. Plausible if thought that normal SIRWT level and alarm instruments have Accumulators. Incorrect because only LC-383A/B/C/D, used for Safety Tank Low Signal (STLS) actuation, have Accumulators that are tested and credited during Loss of Instrument Air.

B. **Correct**. LT-381 and LT-382 provide level indication and alarm and do not have credited Accumulators. LC-383A/B/C/D provide the STLS signal and they do have Accumulators.

C. Incorrect. Plausible if thought that level indication and STLS actuation come from the same channels that are provided with Accumulators. Incorrect because the level indication channels do not use Accumulators that are tested and credited during Loss of Instrument Air.

D. Incorrect. Plausible if thought that the STLS and normal level indication instruments do not have Accumulators, but the alarm indicators do have accumulators.

Technical Reference: LP 7-12-14, Slide #134 & #137, Rev. 1

(Attach if not previously provided including revision number) LP 7-11-22, Slide #45, Rev. 3

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:

Lesson Plan 7-12-14, Engineered Safeguards Control-Licensed Operator
EO 1.2 - **EXPLAIN** how each prime initiation signal is developed.
EO 2.4 - **EXPLAIN** how ESC signals respond to loss of power or sensor failures.

Lesson Plan 7-11-22, Engineered Safeguards Control-Licensed Operator
EO 1.3 - **EXPLAIN** the indications located in the Control Room associated with ECCS.

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

Question History: Last NRC Exam 2005 NRC Exam, Question #54

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

10 CFR Part 55 Content: 55.41 7
55.43

EO *1.2 **(Slide #134)**

Safety Injection Tank (SIRWT) Low Signal (STLS)

SIRWT Level Sensors (A/LC-383-1, A/LC-383-2, B/LC-383-1, B/LC-383-2, C/LC-383-1, C/LC-383-2, D/LC-383-1, and D/LC-383-2)

Two sets of four SIRWT level sensors provide input for channel A and channel B 2/4 coincidence logic matrices for STLS.

Four pneumatic bubblers are installed in the SIRWT.

EO 2.4 **(Slide #137)**

Safety Injection Tank (SIRWT) Low Signal (STLS)

The level sensors would provide low-low level signal (contacts close) upon loss of instrument air supply.

Air accumulators for each pneumatic bubbler will provide backup air supply for at least 12 hours if instrument air is lost.

EO *1.3 **(Slide #45)**

Major Component Description

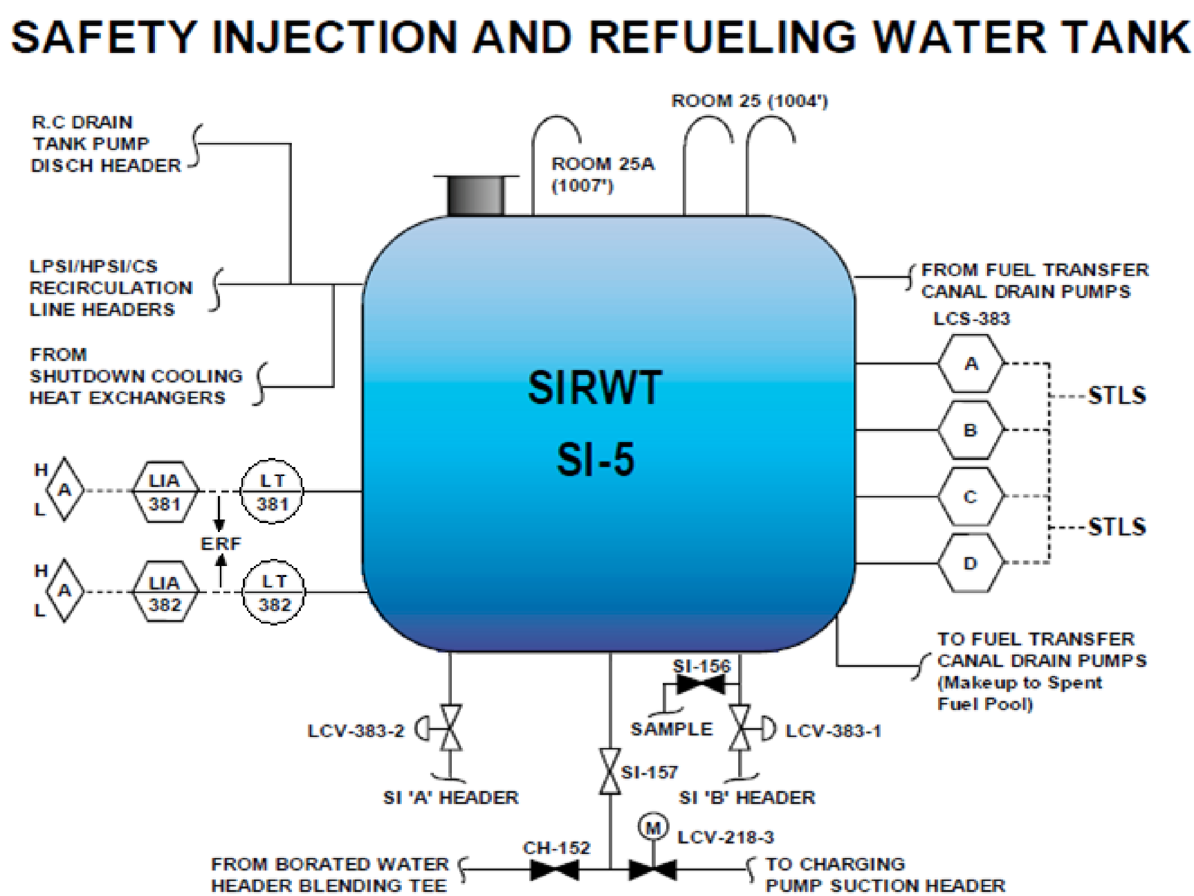
Safety Injection and Refueling Water Tank (SIRWT) (SI-5)

Level Instrumentation

LT-381 provides tank level indication on Panel AI-30A and a signal to the ERF computer. (Bubbler style)

LT-382 provides tank level indication on Panel AI-30B and a signal to the ERF computer. (Bubbler style)

(Slide #45)



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	005 AK2.02	
Level of Difficulty: 3	Importance Rating	2.5	

Inoperable/Stuck Control Rod: Knowledge of the interrelations between the Inoperable/Stuck Control Rod and the following: Breakers, relays, disconnects, and control room switches.

Question: 57

Given the following conditions:

- Regulating Group 4 is being moved for Axial Shape Index control.
- While inserting Group 4, all Control Element Assembly (CEA) movement stopped.
- ARP-CB-1/2/3/A6, Window E-5 – HCV-151 TROUBLE, alarmed concurrently when the CEAs stopped moving.
- AOP-02, Control Rod Malfunctions, Section I, Inoperable, Stuck or Untrippable CEA, has been entered.

Which of the following has occurred?

Power to the CEA _____(1)_____ has been lost.

Place EE-22-HS, Rod Drive Cabinet Power Select Switch to the alternate power supply in the _____(2)_____ .

A. (1) Magnetic Clutches
   (2) Upper Electrical Penetration Room

B. (1) Motors and Rectifiers
   (2) Control Room Rod Drop Test Switch Cabinet

C. (1) Magnetic Clutches
   (2) Control Room Rod Drop Test Switch Cabinet

D. (1) Motors and Rectifiers
   (2) Upper Electrical Penetration Room

Answer: D

K/A Match:
Applicant must have knowledge of power sources to the CEAs including applicable switch positions and locations.

Explanation:

A. Incorrect. Plausible because the Rod Drive Cabinet Power Select Switch will be aligned in the Upper Electrical Penetration Room. Incorrect because power to the magnetic clutches comes from Instrument Buses
AI-40A/B/C/D.

B. Incorrect. Plausible because power to the CEA motors and rectifiers has been lost. Incorrect because the Rod Drive Cabinet Power Select Switch is in the Upper Electrical Penetration Room.

C. Incorrect. Plausible if thought that the magnetic clutches had lost power. Incorrect because the Rod Drive Cabinet Power Select Switch will be aligned in the Upper Electrical Penetration Room.

D. **Correct**. The Rod Drive Cabinet Power Select Switch aligns power to either MCC-3A1 or MCC-4A1. The HCV-151 TROUBLE alarm alerts the operator to a potential loss of power since the alarm came in and the CEAs stopped moving at the same time. With knowledge of the power supply to HCV-151 and that 2 separate power sources are available to the CEA motors and rectifiers it can be deduced that power needs to be realigned to the Rod Drive Control Cabinets at the Upper Electrical Penetration Room.

Technical Reference: TDB-AOP-32, Section XIII, Step 2, Rev. 20
(Attach if not previously provided including revision number)
ARP-CB-1/2/3/A6, Window E-5, Rev. 48
LP-7-12-26, Slide #42, Rev. 2

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-26, Control Rod Drive System-Licensed Operator
EO 1.2a - **DESCRIBE** the interface/ interaction between the CRDS and the following systems/ components: Electrical Distribution System.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 6
55.43 ____

Panel: **CB-1/2/3**	Annunciator: **A6**	Window: **E-5**

LOSS OF POWER TO HCV-151, PZR PORV PCV-102-1 BLOCK VALVE

SAFETY RELATED

HCV-151 TROUBLE

Tech Spec References: 2.1.6

Initiating Device 74/HCV-151 Setpoint UNDERVOLTAGE Power MCC-4A1

OPERATOR ACTIONS

1. Check the indicating lights on HCV-151.

2. IF the valve indicating lights are off, THEN dispatch an Operator to check the valve breaker closed at MCC-4A1-C05. (Room 57)

    2.1 IF breaker is tripped, THEN initiate notification to the Work Week Manager of valve motor fault.

    2.2 IF breaker is closed, THEN initiate notification to the Work Week Manager to check the Control Power Fuses.

    2.3 Refer to Technical Specification 2.1.6.

3. IF the breaker is closed, THEN reset breaker overloads.

Section XIII - Loss of MCC-4A1

INSTRUCTIONS CONTINGENCY ACTIONS

NOTE

Loss of MCC-4A1 may result in the loss of CEA drive motors.

4. IF MCC-3B1 is energized,
AND control rod motion is desired,
THEN ensure EE-22-HS, "ROD DRIVE CABINET EE-22 POWER SELECT SWITCH" is in "3B1-G4L". (Upper Electrical Penetration Room - inside EE-22, "ROD DRIVE CONTROL SYSTEM RELAY CABINET "A"" door with CEDM ground indication)

CEA drive motor power can be manually selected from either MCC-4A1 or MCC-3B1. The CEAs would remain trippable because the magnetic clutches are powered from instrument buses AI-40A/B/C/D.

EO *1.6c **(Slide #42)**

Major Component Description

CEDM Components

Drive Motor

Power Supply

MCC-3B1 and MCC-4A1 supply 480/120 VAC transformers.

The transformer outputs feed through a selector switch in the rod drive cabinets (upper electrical penetration room).

Single phase, 120 VAC powers the motor and rectifier.

The output of the rectifier (90 VDC) is supplied to the brake.

(Slide #42)


MCC-3B1-G4L
TRANSF. #1
OFF
MCC-4A1-A02
TRANSF. #2
3B-1
TRANSFER SWITCH
EE/22 MAIN CB
EE-22-HS
ROD DRIVE CABINET EE-22
POWER SELECT SWITCH

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/22/15	Tier #	1	
Change: 0	Group/Category #	2	
	K/A #	024 AK1.04	
Level of Difficulty: 2	Importance Rating	2.8	

Emergency Boration: Knowledge of the operational implications of the following concepts as they apply to Emergency Boration: Low temperature limits for boron concentration.

Question: 58

What is the reason for keeping the temperature of the Boric Acid Storage Tanks above the low temperature limit?

A. Ensure all of the boric acid inside the tanks remains in solution.

B. Prevent a reactivity excursion due to cold water being added to the Volume Control Tank.

C. Prevent excessive thermal stress at the pipe connections to the tanks.

D. Prevent an excessive deviation between the actual and indicated tank levels.

Answer: A

K/A Match:

Requires applicant knowledge of the solubility of concentrated boric acid.

Explanation:

A. **Correct**. As temperature increases, the solubility of boric acid in solution also increases. Failure to keep boric acid storage tanks above a minimum temperature (based on concentration) can result in solidification of the boric acid.

B. Incorrect. Plausible because this would result in cold water being added to the VCT and a cold water addition to the Reactor would cause a positive reactivity addition. This is incorrect because the water in the VCT is heated through the regenerative heat exchanger and mixed in the Reactor Coolant System loops before it makes it to the reactor core.

C. Incorrect. The reason is to maintain boron in solution. Plausible because other components in the charging and volume control system do have thermal cycle limits based on differential thermal stress (charging nozzles, auxiliary spray valves, regenerative heat exchanger). Incorrect because the tank and piping inject boric acid to either the suction of the charging pumps or the VCT, none of which are maintained at an elevated temperature which would create a thermal stress point in the system.

D. Incorrect. The reason is to maintain boron in solution. Plausible because a difference in temperature in a tank with an external reference line will cause a difference in indicated vs. actual level. Incorrect because the difference in densities between the low limit based on boron concentration and the ambient temperature in the room is negligible, since the minimum temperature even at the highest concentration is 130 degrees and the liquid remains subcooled.

Technical Reference: LP 7-11-2, Slide #199, Rev. 2

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-2, Chemical and Volume Control System-Licensed Operator
EO 1.5 - Given a current copy of the Technical Specifications, **EXPLAIN** the Technical Specifications and bases applicable to the Chemical and Volume Control System.

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

Question History: Last NRC Exam 2005 NRC Exam, Question #20

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

10 CFR Part 55 Content: 55.41 6
55.43 ____

EO 1.1 **(Slide #199)**

Major Component Description

Concentrated Boric Acid Storage Tanks (CH-11A/B)

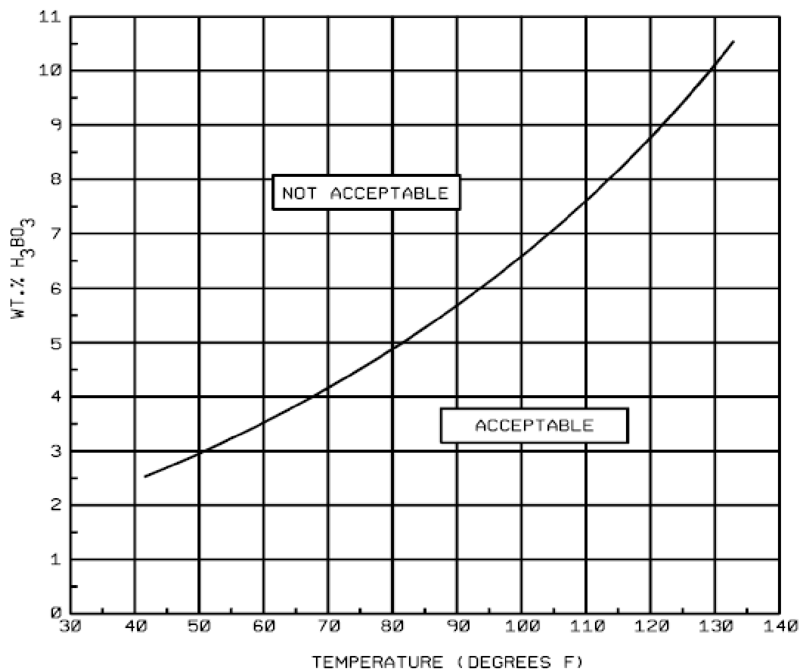
Function: Provide normal and emergency source of concentrated boric acid.

Two tanks designed to hold 5800 gallons each of 2.5% to 4.5% boric acid.

One tank holds enough boric acid to bring the reactor to cold shutdown without control rods.

(LC) Each tank has two channels of heat tracing to keep the contents above the solubility temperature.

(Slide #199)



BORIC ACID SOLUBILITY IN WATER	OMAHA PUBLIC POWER DISTRICT FORT CALHOUN STATION-UNIT 1	FIGURE 2-12

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/22/15	Tier #	1	
Change: 0	Group/Category #	2	
	K/A #	036 AK2.01	
Level of Difficulty: 3	Importance Rating	2.9	

Fuel Handling Accident: Knowledge of the interrelations between the Fuel Handling Incidents and the following: Fuel handling equipment.

Question: 59

Given the following conditions:

- A Fuel Assembly is being inserted into the core.
- After it has been inserted half way, the CABLE SLACK limit switch is tripped.
- The Hoist Load Indicator shows low Hoist weight.
- AOP-08, Fuel Handling Incident, was entered.

What actions are required per OP-12, Fueling Operations?

A. Withdraw the partially inserted Fuel Assembly from the core, rotate it 90°. When rotation is complete, reinsert the Fuel Assembly.

B. Extend the fuel spreader to unbind the Fuel Assembly. Retract the fuel spreader as soon as the Fuel Assembly unbinds.

C. Extend the fuel spreader to unbind the Fuel Assembly. Retract the fuel spreader when the Fuel Assembly is fully inserted.

D. Withdraw the partially inserted Fuel Assembly from the core. Extend the fuel spreader, and then reinsert the Fuel Assembly into the core.

Answer: D

K/A Match:
Applicant must be familiar with precautions and operation of Fuel Handling equipment. As licensed operators, the candidate is required to know precautions and limitations for this operating procedure.

Explanation:

A. Incorrect. Plausible because the correct action is to initially withdraw the partially inserted fuel assembly. Incorrect because Reactor Engineering would need to perform calculations prior to inserting the fuel assembly in a different rotation than planned.

B. Incorrect. Plausible because extending the Fuel Spreader could provide additional room to complete insertion. Incorrect because the Fuel Spreader can only be used when a fuel assembly is fully inserted or fully withdrawn.

C. Incorrect. The Fuel Spreader can only be used when a fuel assembly is fully inserted or fully withdrawn.

D. **Correct**. Per Precaution 2 in OP-12, Fuel Handling, Attachment 8, Special Fuel Handling Techniques, the Fuel Spreader shall not be operated while a Fuel Assembly is partially inserted in the Core. The Spreader may only be operated when the Fuel Assembly is fully inserted or fully withdrawn. Failure to follow this precaution could cause fuel damage.

Technical Reference: AOP-08, Entry Conditions, Rev. 10a
(Attach if not previously provided including revision number) OP-12, Precautions 1-3, Rev. 70
LP 7-11-13, Slide #62 & #101, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-13, Fuel Handling-Licensed Operator
EO 2.1 - **DISCUSS** the prerequisites and precautions associated with fuel handling equipment and the refueling machine.

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

Question History: Last NRC Exam 2005 NRC Exam

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

10 CFR Part 55 Content: 55.41 2
55.43 ___

AOP-08, Section 2

2.0 ENTRY CONDITIONS

A fuel assembly has been damaged which may be indicated by any of the following:

A. Area radiation monitors increase.

B. "RM-050 CNTMT PARTICULATE HIGH RADIATION" alarm (AI-33C; A33C).

C. "RM-051 CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

D. "RM-052 STACK/CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

E. "RM-062 AUX BLDG VENT STACK HIGH RADIATION" alarm (AI-33C; A33C).

F. Containment air particulate high radiation indication upscale.

G. Ventilation Isolation Actuation Signal (VIAS).

H. While handling fuel, the Hoist Load Indicator shows low Hoist weight.

I. Possible damage to Fuel Assembly is observed.

OP-12, Attachment 8, Precautions

1. The hoist Overload and Underload Limit shall not be exceeded without verbal approval from the Reactor Engineer. This is the load force minimum required to cause the Spacer grids to fail per the fuel vendors.

2. The Spreader shall not be operated while a Fuel Assembly is partially inserted in the Core. The Spreader may only be operated when the Fuel Assembly is fully inserted or fully withdrawn. Failure to follow this precaution could cause fuel damage.

3. To prevent excessive interacting forces, the Hoist Overload and Underload Setpoints have been set to prevent Hoist Movement when an excessive weight deviation occurs.

EO *1.2 **(Slide #62)**

Major Component Description

Fuel Spreader

The Fuel Spreader is used to align adjacent bundles to the bundle being moved to prevent bundle to bundle interference.

EO 1.3 **(Slide #101)**

Instruments and Controls

Computer Touch Screen

Hoist Screen

(17) GRAPPLE OPEN (yellow); hoist grapple is open.

(18) CABLE SLACK (yellow); grapple is full down and slack cable limit switch is actuated. Also, hoist load is below 30 lbs.

(19) ENCODER UP LIMIT (blue); hoist encoder reaches up limit.

(20) LOAD BYPASS ACTIVE (red); Hoist Load Bypass has been activated by pushbutton or automatically.

(21) HOIST BOX LATCHED (green); hoist box is on the down stop over the core or on up stop over the upender.

(22) FUEL SPREADER RETRACTED (green); fuel spreader is fully retracted.

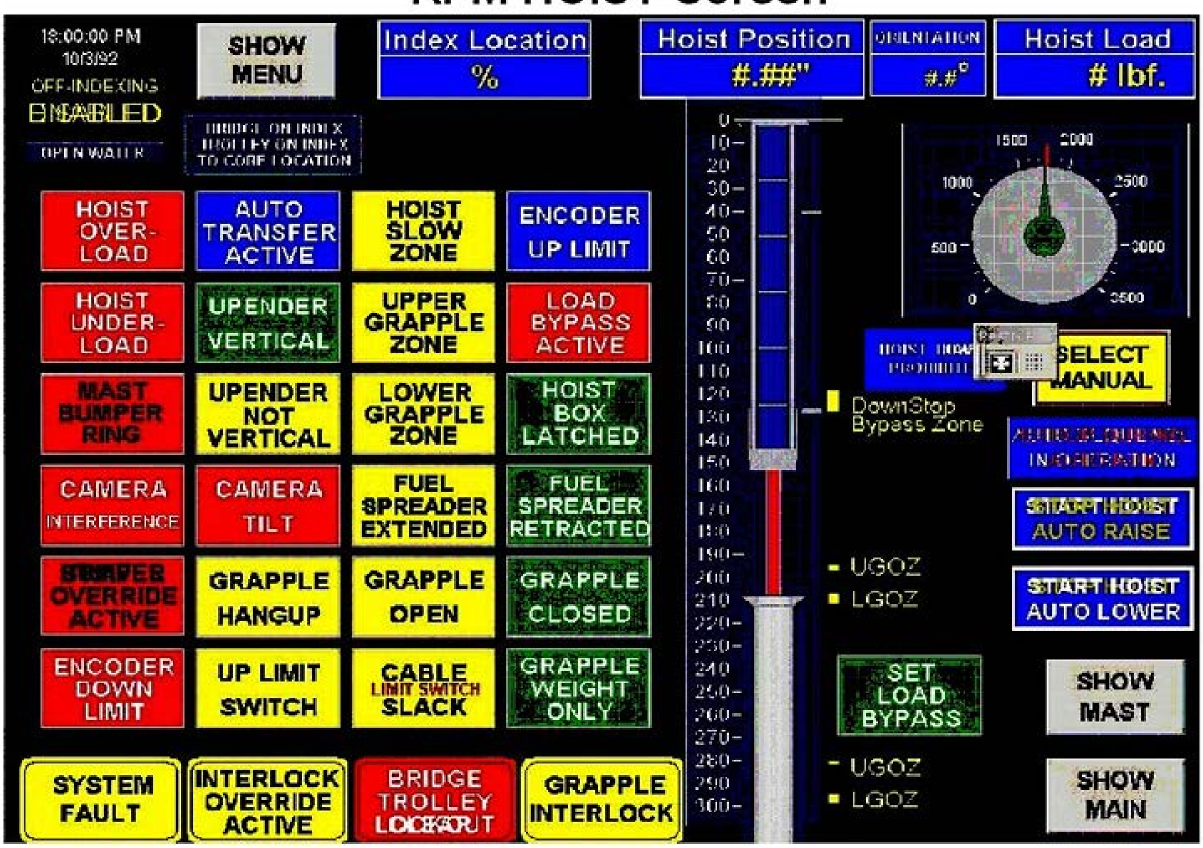
(23) GRAPPLE CLOSED (green); hoist grapple is closed.

(24) GRAPPLE WEIGHT ONLY (green); fuel bundle is down and only grapple weight remains. Also, if hoist load is detected below 310 lbs at any hoist elevation.

(25) SET LOAD BYPASS (green); hoist has an underload condition and Hoist Load Bypass needs to be activated.

(Slide #101)

RFM HOIST Screen



Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/12/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	051 G 2.4.47	
Level of Difficulty: 4	Importance Rating	4.2	

<u>Loss of Condenser Vacuum</u>: Emergency Procedures/Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

Question: 60

Given the following conditions at 100% power:

- Main Generator electric output is slowly lowering.
- RCS temperature and pressure are stable at normal operating parameters.
- Indicated NI power is stable.
- Calculated Thermal Power (XC-105) is slowly lowering.

Which of the following could be the cause of these indications?

A. Missouri River temperature slowly lowering.

B. Condenser pressure slowly rising.

C. 345 KV System voltage slowly lowering.

D. A steam leak upstream of the Turbine Stop Valves.

Answer: B

K/A Match:

Given Control Room information, applicant must be able to determine plant effects of a loss of vacuum.

A. Incorrect. Plausible because Missouri River temperature can affect vacuum which would lead to the conditions listed. Incorrect because River temperature would have to be rising.

B. **Correct**. The Turbine Control System at Fort Calhoun is operated in the MANUAL mode. This means that the operator sets the valve position and the Turbine Control Valves maintain position independent of steam pressure, Condenser vacuum, etc. Condenser pressure is rising and Main Generator output is lowering which are both symptoms of a Loss of Condenser Vacuum. This is reinforced by a loss of thermal efficiency (XC-105 is slowly lowering) without any change in RCS pressure or temperature.

C. Incorrect. Plausible because decreased electrical output is a symptom of a loss of vacuum. Incorrect because this is Main Generator load (watts) that is lowering not voltage.

D. Incorrect. Plausible because electric output is lowering this could be indicative of less steam going to the Turbine. Incorrect because XC-105 would be rising and RCS temperature and pressure would be lowering.

Technical Reference: AOP-26, Section I, Rev. 10
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-26, Turbine Malfunctions-Licensed Operator
EO 1.2 - **DESCRIBE** how the plant responds to malfunctions of the turbine or turbine support systems.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 5
55.43 ______

AOP-26, Section I

2.0 ENTRY CONDITIONS

A loss of vacuum is occurring, other than caused by seasonal river temperature increases, which may be indicated by any of the following:

A. Degraded Condenser vacuum.

B. Decreasing electrical output.

C. High Exhaust Hood temperature.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/20/15	Tier #	1	
Change: 0	Group/Category #	2	
	K/A #	059 AA1.01	
Level of Difficulty: 3	Importance Rating	3.5	

Accidental Liquid Radwaste Release: Ability to operate and/or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive-liquid monitor.

Question: 61

Given the following conditions:

- An Accidental Liquid Radwaste Release is in progress.
- Both Room 10 and Control Room RM-055 ratemeter keyswitches are in the KEYPAD position.

What is the plant response when the High Alarm Trip Setpoint is reached on RM-055, Liquid Radioactive Waste Effluent Monitor?

A. Any running Hotel Waste Tank Pump breaker trips.

B. Any running Monitor Tank Pump breaker trips.

C. HCV-691 & HCV-692, Release Control Valves, both close.

D. The release continues because the trip functions are disabled.

Answer: D

K/A Match:

Applicant must recognize operational implications of switch positions for Radiation Monitors.

Explanation:

A. Incorrect. Plausible because RM-055 will trip the Hotel Waste Tank Pump breaker. Incorrect because RM-055 ratemeter keyswitches are in the KEYPAD position.
B. Incorrect. Plausible because RM-055 trip the Monitor Tank Pump breaker. Incorrect because RM-055 ratemeter keyswitches are in the KEYPAD position.
C. Incorrect. Plausible because RM-055 closes HCV-691 and HCV-692. Incorrect because RM-055 ratemeter keyswitches are in the KEYPAD position.
D. **Correct**. When RM-055 ratemeter keyswitches are in the KEYPAD position the trip functions are disabled.

Technical Reference: LP 7-12-3, Slides #255 & #257, Rev. 1

(Attach if not previously provided including revision number) OI-RM-1, Precautions 3, 4, & 5 and Attachment 10, Step 1, Rev. 68

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-12-3, Radiation Monitoring System-Licensed Operator EO 4.0 - **EXPLAIN** the operations, actuations and applications of the individual radiation monitors. EO 7.1 - **DISCUSS** the purposes, prerequisites and precautions associated with OI-RM-1.

Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam	None	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	13	
	55.43		

Precautions 3, 4, & 5 from OI-RM-1

PRECAUTIONS

1. Radon gases, which are naturally emitted from the ground, normally rise and are dissipated into the atmosphere. During a temperature inversion, these gases are trapped close to the ground causing the Plant Process Monitors to read a higher background value. Although this is still background radiation, the Process Monitors may raise to the ALERT and High Alarm setpoints.

2. The ability to obtain 12 hr grab samples can be maintained by use of a portable vacuum pump when both RM-062 and RM-052 are inoperable at the same time. [AR 12912]

3. All automatic functions are inoperable when the Control Room ratemeter is placed in KEYPAD. All requirements of the Technical Specifications and the ODCM must be met prior to placing the Control Room ratemeter to KEYPAD.

4. All automatic functions are still operable when the local ratemeters is placed in KEYPAD.

5. Ratemeter alarm functions and trips are blocked when the check source is activated.

Step 1 from OI-RM-1, Attachment 10

Attachment 10 - RM-055 (Liquid Waste Disposal)

1. IF removing RM-055 from service,
   THEN perform the following:

   a. Ensure the ODCM requirement is met.

   b. Place RM-055 Control Room ratemeter keyswitch to KEYPAD.

   c. IF desired to prevent the trip functions,
      THEN place RM-055-1 local ratemeter keyswitch to KEYPAD (Rm 10).

EO 4.0, *4.1 **(Slide #257)**

RM-055

RM-055 OVERBOAARD DISCH WASTE HIGH RADIATION alarm on AI-33C will annunciate upon the ratemeter reaching it HIGH setpoint.

This condition will also result in the automatic isolation liquid waste releases.

RM-055 OVERBOARD DISCH WASTE TROUBLE alarm will annunciate on a ratemeter failure condition

EO 4.0, *4.1 **(Slide #255)**

RM-055

A HIGH alarm on the Control Room Ratemeter will terminate liquid waste releases by automatically closing valves HCV-691/692 and stopping Monitor Tank Pumps and tripping the breakers for the Hotel Pumps.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	069 AA2.02	
Level of Difficulty: 3	Importance Rating	3.9	

Loss of Containment Integrity: Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Verification of automatic and manual means of restoring integrity.

Question: 62

Given the following conditions:

- During surveillance testing HCV-1560A, Deaerated Water Supply to Containment Isolation Valve, was declared inoperable due to a ruptured diaphragm.
- AOP-12, Loss of Containment Integrity, has been entered.

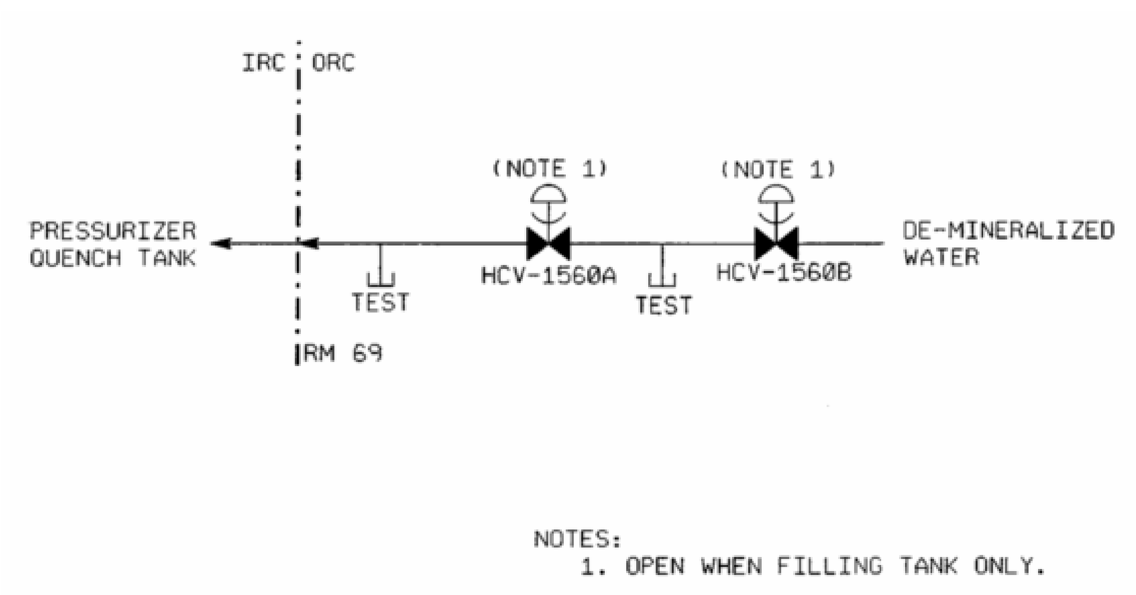
Using the attached figure, what additional action(s) is/are required prior to exiting AOP-12, Loss of Containment Integrity?

A. Verify HCV-1560B is OPERABLE and closed.

B. Handjack the operator for HCV-1560A closed.

C. Isolate air from HCV-1560B and depressurize its air regulator.

D. Remove the control power fuses from HCV-1560A.



Answer: C

K/A Match:

Requires procedural knowledge to restore Containment Integrity.

Explanation:

A. Incorrect. Plausible because verifying the valve OPERABLE and closed would ensure that Containment Integrity was reestablished, however, it is not part of AOP-12 direction and because this valve, if manipulated, could possibly exacerbate the Loss of Containment Integrity.

B. Incorrect. Plausible but AOP-12 directs disabling the redundant valve, which is HCV-1560B, not HCV-1560A

C. **Correct**. Per AOP-12, Step 8, if Containment Integrity is established by disabling the redundant valve for HCV-1560A, which is HCV-1560B. This is done by Step 8a or 8b.

D. Incorrect. Plausible because AOP-12 directs removal of the control power fuses for the redundant valve, this is HCV-1560B not HCV-1560A.

Technical Reference: AOP-12, Step 8, Rev. 8

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-8, Containment-Licensed Operator
EO 2.3 - Briefly **DESCRIBE** actions necessary if containment integrity is violated as per AOP-12 and Tech Spec 2.6.

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

Question History: Last NRC Exam 2012 NRC Exam, Question #22

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

Step 8 from AOP-12

8. **IF** the loss of Containment Integrity is due to an inoperable automatic isolation valve for reasons other than excessive leakage or faulty Control Room indication,

THEN establish Containment Integrity within 1 hour by performing Step a or b:

8.1 **IF** Containment Integrity is not restored within 1 hour,

THEN IMPLEMENT Step 6.1.

a. Disable the inoperable valve in the closed position by performing the following:

1) Isolate Instrument Air from the air operator for the inoperable valve.

2) Depressurize the regulator for the inoperable valve.

b. Direct EM to remove the Control Power Fuses for the inoperable valve.

a.1 Disable the redundant valve in the closed position by performing the following:

1) Isolate Instrument Air from the air operator for the redundant valve.

2) Depressurize the regulator for the redundant valve.

b.1 Direct EM to remove the Control Power Fuses for the redundant valve.

Bank Question:

During surveillance testing HCV-1560A, Dearated Water Supply to Containment Isolation Valve was declared inoperable for faulty control room indications. AOP-12, LOSS OF CONTAINMENT INTEGRITY has been entered.

Using the attached Figure Q22-1, what additional actions are required prior to exiting AOP-12?

A. Isolate air from HCV-1560A and depressurize its associated regulator
Direct EM to remove control power fuses for HCV-1560A.

B. Isolate air from HCV-1560B and depressurize its associated regulator.

C. Verify HCV-1560B is operable and closed.

D. Direct EM to remove control power fuses for HCV-1560A.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	076 AK2.01	
Level of Difficulty: 4	Importance Rating	2.6	

High Reactor Coolant Activity: Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors.

Question: 63

Given the following conditions:

- Plant is at 100% power.
- AOP-21, Reactor Coolant System High Activity, is in progress for several days for a minor fuel leak.
- Containment Pressure Reduction is in progress.
- RM-052, Containment/Auxiliary Building Vent Stack Swing Monitor, is aligned to the Auxiliary Building Vent Stack.
- RM-052 count rate has been slowly rising; however, Chemistry reports NO change in RCS activity.

Which of the following caused the rising count rate on RM-052, Containment/Auxiliary Building Vent Stack Swing Monitor?

A. Sample flow rate through FC-052, Flow Controller lowered from 2 SCFM to 1 SCFM.

B. The ratio of specific isotopes leaking from the fuel changed.

C. Motor operated valves MV1 and MV2 on the RM-052/RM-062 skid inlet piping are both open.

D. A temperature inversion caused RM-052 to read a higher background value.

Answer: D

K/A Match:

Applicant must have knowledge of environmental effects to Process Radiation Monitors.

Explanation:

A. Incorrect. Plausible because flow through FC-052 Flow Controller is less than design. Incorrect because a lower flow would yield a lower count rate on RM-052. Lower sample flow should result in lower indicated activity, but it also results in the sample remaining in the detector area longer, which allows greater isotopic decay, which is what is plausible.

B. Incorrect. Plausible if power level were changing then half-lives would change. Incorrect because the plant has been at 100% for several days, and these changes would be indicated in chemistry samples. With leaking fuel, the size of the leak and the energy of the isotopes being analyzed affects the detector sensitivity and accuracy. Throughout core life and differing power levels, the ratio of gases that are being analyzed would change. The detectors are calibrated to a given ratio and energy level. If there was a chemical change, power or core burnup change, or shift in the fuel defects being released into containment, then the detector could indicate a change in output because the sensitivity or accuracy is affected.

C. Incorrect. Plausible because if these valves were opened RM-052 would be taking a sample from the Vent Stack and Containment at the same time. Incorrect because these valves are electrically interlocked so they cannot be lined up simultaneously, and sampling both locations would provide a dilution for the containment sample.

D. **Correct**. RM-052 will sense the release of radioactivity from the Containment Pressure Reduction as well as the temperature inversion that is occurring. Per OI-RM-1, Precaution 1, buildup of radon gases will occur during a temperature inversion and cause Process Radiation Monitor readings to rise without a corresponding increase in RCS activity.

Technical Reference: OI-RM-1, Precaution 1, Rev. 68

(Attach if not previously provided including revision number) LP 7-12-3, Slides #184 and #193, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-3, Radiation Monitoring System-Licensed Operator
EO 4.0 - **EXPLAIN** the operations, actuations and applications of the individual radiation monitors.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 11
55.43 ____

OI-RM-1 Precaution

PRECAUTIONS

1. Radon gases, which are naturally emitted from the ground, normally rise and are dissipated into the atmosphere. During a temperature inversion, these gases are trapped close to the ground causing the Plant Process Monitors to read a higher background value. Although this is still background radiation, the Process Monitors may raise to the ALERT and High Alarm setpoints.

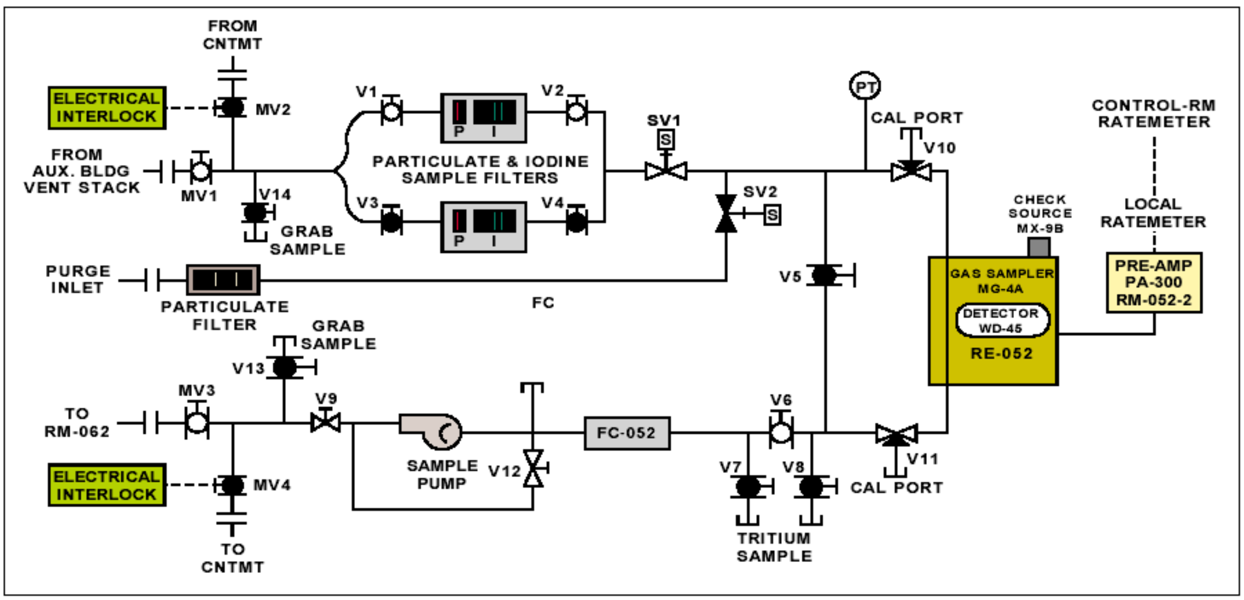
EO 4.0 **(Slide #193)**

RM-052

Sample flow is measured and controlled by FC-052. FC-052 is a package instrument containing a flow element, flow measurement system , flow output signal and control valve.

A manually set 0 - 5 VDC input is input to the controller from a potentiometer voltage A manually set 0 - 5 VDC input is input to the controller from a potentiometer voltage control. The instrument compares the sensed flow to the setpoint and automatically adjusts the position of an automatically metering solenoid valve to control flow. EAR 26003 determined that a set flow rate of 2 SCFM was sufficient to assure isokinetic flow conditions for all plant conditions.

(Slide #184)



EO 4.0 **(Slide #184)**

RM-052

Motor operated valves MV1 and MV2, located on the skid inlet piping, are electrically interlocked to ensure that both sample flow paths are not lined up simultaneously.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/12/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	A11 AA1.3	
Level of Difficulty: 4	Importance Rating	3.0	

RCS Overcooling - PTS: Ability to operate and/or monitor the following as they apply to the RCS Overcooling: Desired operating results during abnormal and emergency situations.

Question: 64

Given the following conditions after a trip from 100% power:

- An Uncontrolled Heat Extraction event is in progress on Steam Generator RC-2A.
- Reactor Coolant System (RCS) pressure is 2125 psia.
- Pressurizer level is 9% and slowly rising.
- Reactor Vessel Level Monitoring System (RVLMS) is 63%.
- Core Exit Thermocouple temperature is 430°F and stable.
- Steam Generator RC-2B wide range level is 30%.
- Steam Generator RC-2A wide range level is 0%.
- All automatic safety systems have functioned as expected.
- Reactor Coolant Pumps RC-3A and RC-3C are running.

Which of the following actions is required?

A. Perform HPSI Stop and Throttle Criteria to avoid Pressurized Thermal Shock.

B. Open the Reactor Head Vents to restore RVLMS level.

C. Initiate Pressurizer Spray flow to restore pressure control.

D. Slowly feed both Steam Generators to raise level.

Answer: C

K/A Match:

Applicant must evaluate plant conditions and choose the desired action to prevent Pressurized Thermal Shock.

Explanation:

A. Incorrect. Plausible because evidence of an imminent Pressurized Thermal Shock condition is present but the requirements to implement are not being met. Specifically, PZR level less than 10% indicates that Safety Injection flow should not be reduced (even though it is rising).

B. Incorrect. Plausible because RVLMS level is < 100%. Incorrect because opening the Reactor Head Vents would not restore Pressure Control.

C. **Correct**. Given the conditions listed, initiating Pressurizer Spray will reduce RCS pressure and restore control of subcooling.

D. Incorrect. Plausible because SG level is close to the SG 27% wide range limit for steaming the intact SG because dryout can result in the RCS going solid should temperature increase.

Technical Reference: EOP-05, Step 18, Step 18 CAUTION, & 29, Rev. 30
(Attach if not previously provided including revision number) EOP/AOP Floating Step A, Step 1 CAUTION, Rev. 7

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-15, EOP-05, Uncontrolled Heat Extraction-Licensed Operator EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of an UHE.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ___

Step 18 CAUTION from EOP-05

CAUTIONS

1. If the initial cooldown rate exceeds 100°F in any hour or is above the PT Curve on Attachment PC-12, RCS Pressure-Temperature Limits, there is a potential for Pressurized Thermal Shock of the Reactor Vessel. The Appendix E Curve on Attachment PC-12, RCS Pressure-Temperature Limits, can be met by **ALL** of the following methods:
   - Controlled addition of Feedwater to S/Gs
   - Control of heat removal from unaffected S/G(s)
   - Control of RCS repressurization from Charging and HPSI Pumps
   - Control of RCS repressurization using Main/Auxiliary Spray

2. Allowing the RCS to heatup after S/G dry out can result in the RCS going solid and exceeding RCS Pressure-Temperature Limits.

****

✖ 18. **IF** RC-2A is the least affected S/G, **THEN** prepare to steam RC-2A prior to reaching 27% WR in RC-2B by performing step a, b, or c:

18.1 **IF** RC-2B is the least affected S/G, **THEN** prepare to steam RC-2B prior to reaching 27% WR in RC-2A by performing step a, b, or c:

Step 1 CAUTION from Floating Step A

1. Verify **ALL** of the following stop and throttle criteria are satisfied:

   - RCS subcooling is greater than or equal to 20F
   - PZR level is greater than or equal to 10% and not lowering
   - At least one S/G is available for RCS heat removal
   - RVLMS indicates level is at or above the top of the Hot Leg (43%, ERF "I" display)

NOTE

Small changes in RCS Temperature and Inventory will cause pressure changes in a solid pressurizer.

29. Verify the RCS is not water solid by the following:

    a. Verify that RCS inventory or temperature changes do **NOT** produce a severe pressure response.

    b. Verify at least **ONE** of the following conditions exist:

    - Pressurizer level less than 100%
    - RVLMS less than 100%

29.1 **IF** the RCS is water solid **THEN** IMPLEMENT Attachment PC-14, Water Solid Operations.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/10/15	Tier #	1	
Change: 1	Group/Category #	2	
	K/A #	A16 AK1.2	
Level of Difficulty: 3	Importance Rating	3.0	

Excess RCS Leakage: Knowledge of the operational implications of the following concepts as they apply to the (Excess RCS Leakage): Normal, abnormal and emergency operating procedures associated with Excess RCS Leakage.

Question: 65

Given the following conditions:

- AOP-22, Reactor Coolant Leak, Section II, Reactor Coolant Leak less than 40 gpm, is in progress.
- After verifying Reactor Coolant System (RCS) pressure less than 1700 psia, PPLS is blocked.

Which of the following is the reason for performing this action in AOP-22, Reactor Coolant Leak?

A. Enable the Low Temperature Overpressure Protection circuitry.

B. Prevent lifting of the Power Operated Relief Valves if control of RCS T_{COLD} is lost.

C. Prevent initiation of Safety Injection because HPSI Stop and Throttle criteria is already met.

D. Maintain the normal Boration path available during Steam Generator depressurization.

Answer: A

K/A Match:

Applicant must know the operational implications of performing steps in AOP-22, Reactor Coolant Leak, including what those steps provide for.

Explanation:

A. **Correct**. With RCS pressure less than 1700 psia, the operator is preparing to enable the Low Temperature Overpressure Protection (LTOP) circuitry. Pressurizer Pressure Low Signal (PPLS) must be blocked.

B. Incorrect. Plausible because one of the actions to verify LTOP is to ensure that RCS temperature instruments T-113 and T-123 are OPERABLE. These instruments enable the LTOP network and if RCS temperature control is lost there is a potential for the PORVs to lift, however, the reason for blocking PPLS at 1700 psia is to enable LTOP.

C. Incorrect. Plausible because the only step in AOP-22 that blocks PPLS occurs in Section II (RCS leakage < 40 gpm). If leakage is > 40 gpm, entry into EOP-00, SPTAs is required. If PPLS were not blocked then SI flow would initiate but this is not required when leakage is < the capacity of 1 Charging Pump.

D. Incorrect. Plausible because if PPLS were not blocked the Emergency Boration flowpath would be aligned at 1600 psia.

Technical Reference: TDB-AOP-22, Steps 7 and 8, Rev. 34
(Attach if not previously provided including revision number) AOP-22, Section I, Step 7, Rev.34

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-22, AOP-22, Reactor Coolant Leak-Licensed Operator
EO 1.5 - Given the caution statements and/or notes listed in this AOP, **EXPLAIN** the reason for each.

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

Question History: Last NRC Exam 2005 NRC Exam, Question #26

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

10 CFR Part 55 Content:
55.41 10
55.43 ______

Steps 7 & 8 from AOP-22, Section II

7. Verify RCS pressure is less than 1700 psia.

At less than 1700 psia PPLS may be blocked which activates the Low Temperature Over Pressure (LTOP) protection.

---

8. Initiate LTOP by performing the following:

   a. Verify BOTH of the following RCS temperature instruments are operable:

      - T-113
      - T-123

   a.1 IF either RCS temperature instrument is inoperable, THEN disable the PORVs by closing BOTH PORV Block Valves:

      - HCV-150
      - HCV-151

Step 7 from AOP-22, Section I

2. **IF** RCS leakage rate is greater than 40 gpm
   AND the Reactor is critical,
   THEN perform the following:
   a. Trip the Reactor.
   b. GO TO EOP-00, Standard Post Trip Actions.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/10/15	Tier #	3	
Change: 1	Group/Category #	1	
	K/A #	2.1.4	
Level of Difficulty: 2	Importance Rating	3.3	

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.

Question: 66

Given the following conditions:

- A licensed operator has just returned from a routine eye exam with their personal Optometrist.
- They received a prescription for eye glasses for the first time.
- The Optometrist verified a corrected vision of 20/20 in both eyes wearing the new glasses.

What action is required before the operator may resume normal duties?

A. The operator must obtain a prescription set of respirator glasses for use with an SCBA mask.

B. The operator must receive from the NRC an amended license stating the new restriction for eye glasses.

C. The License Maintenance Coordinator must evaluate the results of the eye exam.

D. The operator must be evaluated by the Medical Review Officer (MRO) for an amendment to their license restrictions.

Answer: D

K/A Match:

Requires applicant knowledge of license maintenance requirements.

Explanation:

A. Incorrect. Plausible because the prescription was checked by a doctor. It is also plausible because the operators are required to have respirator glasses in the control room any time they are standing watch and require the glasses to be able to read procedures and indications, This would be correct if the doctor was the company's MRO and the restriction was evaluated against ANSI/ANS 3.4 requirements for licensed operator restrictions.

B. Incorrect. Plausible because this change will result in an amended license with a restriction for prescription eyewear. This is incorrect because the license does not have to be received from the NRC before the operator can resume license duties.

C. Incorrect. Plausible because the License Maintenance Coordinator does submit the license restriction amendment form to the NRC. Incorrect because the results must be reviewed by an MRO; the License Maintenance Coordinator does not evaluate the results for restrictions.

D. **Correct**. The operator must be evaluated by the MRO and the restriction amendment is required to be submitted prior to resuming licensed operator duties.

Technical Reference: SO-G-64, Step 3.6.5, Rev. 36
(Attach if not previously provided including revision number) Form NRC-396

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-13, License Maintenance-Licensed Operator
EO 1.2 - **STATE** the medical examination requirements.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43

3.5.2 Ensuring the following physical qualifications, as required, are met prior to assuming watch standing duties.

- NRC Licensed Operator Physical
- Fire Brigade

3.6 NRC Licensed Operator is responsible for:

3.6.1 Contacting the Access Authorization Group and setting a date for the physical upon notification from the Manager-Operations or designated alternate that an NRC Licensed Operator Physical is due.

3.6.2 Notifying the Manager-Operations or designated alternate that his/her physical was/was not completed as scheduled or requires follow-up.

3.6.3 Completing any required follow-up items such as consultation with personal physician, additional tests, further examinations, etc., as may be specified by the examining physician.

3.6.4 Ensuring they have either a set of spectacle glasses for use under a facemask stored in the Control Room or contact lenses if they are required to wear corrective lenses while performing licensed duties.

3.6.5 Completing SY-FC-102-206 Attachment 1 Medication Form when his/her mental, physical, or medical condition changes. The completed form shall be given to their immediate supervisor for review by the Medical Review Officer prior to assuming any licensed duties.

A. The operator shall **NOT** assume any duties which require a NRC license until an evaluation by the Medical Clinic has been made.

1. Examples of mental, physical, or medical condition changes that require notification include, but are not limited to the following:

   - Diabetes
   - Any hospitalizations, fainting, or dizziness
   - Major changes in vision since last physical (includes Lasik or other refractive surgery, or glasses being prescribed for the first time)
   - Injuries which result in prescriptions for pain medication OR limit mobility or dexterity (i.e., hand, foot/leg, back, or neck injuries)
   - Beginning or ending treatment for high blood pressure/hypertension
   - Beginning or ending treatment for nervous or mental disorders

Form NRC-396, "Certification of Medical Examination by Facility Licensee" Form NRC-396 is used to provide the Commission information required by Commission's regulations regarding an individual's medical fitness. The information provided is considered personal private information and is withheld from public disclosure under 10 CFR 2.390, "Public Inspections, Exemptions, and Requests for Withholding" (Ref. 17).

Section A, "Medical Exam Information," of the form (1) certifies that the applicant or licensee has been examined by a physician and that he or she has been found to meet the safeguards and fitness for duty requirements for licensed operators at the facility of record; (2) certifies that in reaching this determination, the guidance contained in ANSI/ANS-3.4-1996 (-1983), or an acceptable alternative method approved by the NRC, was followed and that documentation (medical evidence) is available for review by the NRC; and (3) identifies the type of licensed condition(s), if any, requested based on medical evidence. A brief explanation of the requested licensed condition and its relationship is also annotated. The following line items are provided for selection by the examining physician:

- No restrictions.
- Corrective lenses shall be worn when performing licensed duties.
- Hearing aid shall be worn when performing licensed duties.
- Shall take medication as prescribed to maintain medical qualifications.
- Shall use therapeutic device(s) as prescribed to maintain medical qualifications.
- Solo operation is not authorized.
- Shall submit medical status report every 3, 6, or 12 months.
- Shall not perform licensed duties requiring a respirator.
- Other restriction or exception.
- Restriction change from previous submittal. • Information only.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #	3	
Change: 1	Group/Category #	1	
	K/A #	2.1.37	
Level of Difficulty: 3	Importance Rating	4.3	

Conduct of Operations: Knowledge of procedures, guidelines, or limitations associated with reactivity management.

Question: 67

What are the Reactivity Management requirements with respect to CEA movements during normal operation?

A. Concurrence by Shift Manager prior to all insertions and withdrawals of CEAs.

B. Direct Control Room Supervisor oversight and peer checking for all insertions and withdrawals.

C. Direct Control Room Supervisor oversight and peer checking for withdrawals only.

D. Concurrence by Shift Manager and peer checking for withdrawals only.

Answer: B

K/A Match:

Applicant must be aware of all Reactivity Management requirements.

Explanation:

A. Incorrect. Plausible because SM/CRS concurrence is required for positive reactivity insertions (rod withdrawals), but not for rod insertions, however, direct oversight is also required.

B. **Correct**. As outlined in Fort Calhoun Station Standing Order. Permission is required prior to any reactivity addition (either positive or negative) to the Reactor.

C. Incorrect. Plausible because CRS oversight is required, however, it includes both insertion and withdrawal of CEAs.

D. Incorrect. Plausible because direct Shift Manager oversight is allowed but not required and peer checking is performed, however, it includes both insertion and withdrawal of CEAs.

Technical Reference: SO-O-1, Step 5.14.2.B, Rev. 107

(Attach if not previously provided including revision number) OP-AA-300, Reactivity Management, Steps 3.7, 3.8, & 4.5.10, Rev. 8

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-1, Standing Orders and Fort Calhoun Station Guidelines-LO EO 2.0 - **STATE** some of the activities, covered by Standing Orders, which require written procedures per Regulatory Guide 1.33.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 1
55.43

FORT CALHOUN STATION
STANDING ORDER

INFORMATION USE

SO-O-1
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5.14.2.B The SM/CRS's permission is required prior to any reactivity addition (either positive or negative) to the reactor. This includes such activities as withdrawing or inserting control rods, adding water to maintain 100% power, adding a heavy blend to lower power, rinsing in an Ion Exchanger or adding/removing load from the main turbine.

During certain emergencies, SM/CRS permission is not required when making negative reactivity changes. However, whenever possible, SM/CRS permission to conduct a reactivity change is still strongly encouraged. Examples of these types of operations include, but are not limited to, the following:

- When operating in an EOP or AOP
- Conditions are present that require a manual reactor trip
- Conditions are present that require an emergency boration

3.3. Plant Manager is the senior sponsor for the Reactivity Management Program at each Exelon Nuclear site.

3.4. Operations Director is responsible for the site integrated Reactivity Management Program ensuring that plant operations, maintenance, and engineering activities follow established Reactivity Management controls.

3.5. Nuclear Fuels Director is responsible for implementing Nuclear Fuel policies, programs, and products that support error-free reactivity operations at the sites.

3.6. Shift Operation Superintendent (SOS) is responsible for establishing consistent expectations for reactivity maneuvering oversight and developing and administering the site's integrated Reactivity Management program.

3.7. Shift Manager (SM) is responsible for communicating requirements for procedural adherence, conservative response to abnormal reactivity occurrences and proper respect for the reactor core to shift personnel.

3.8. Unit Supervisor (Control Room Supervisor) is responsible for ensuring Reactor Operators properly adhere to procedures, conservatively respond to abnormal reactivity events, and demonstrate proper respect for reactivity by using conservative operating practices. The safety and integrity of the core take precedence over power production.

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OP-AA-300
Revision 8
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From section 4.5, Responsibilities: Unit Supervisor (Control Room Supervisor) / Senior Reactor Operator

4.5.3. **REVIEWS and APPROVES** all planned reactivity changes in accordance with approved procedures or instructions developed by Reactor Engineering and communicated on the REMA Form.

4.5.4. **ENSURES** the pre-job brief for the work or activity addresses the potential reactivity effects.

4.5.5. **ENSURES** core parameters are maintained within prescribed bands.

4.5.6. **ENSURES** trainees manipulating reactivity controls are enrolled in an approved training program and directly supervised by a licensed individual.

NOTE: Control of the Integrated Control System (ICS) from the Unit Load Demand (ULD) station is considered an automatic method. (TMI only)

4.5.7. **ENSURES** positive reactivity additions are performed using only one manual method at a time.

4.5.8. **ENSURES** the level of control room activities does __not__ detract from the Reactor Operators' ability to properly monitor the reactor.

4.5.9. **ENSURES** operators who are in the process of reactivity manipulations are focused on the task and not involved with concurrent activities that would cause a distraction.

4.5.10. **DIRECTS** reactivity changes **and ENSURES** reactivity manipulations are made in a deliberate, carefully controlled manner while the reactor is monitored to ensure the desired response is attained.

– During manual control rod movement, the Unit Supervisor (Control Room Supervisor) should be positioned in proximity to the Reactor Operator, typically the location from which EOP actions are directed.

From OPD-3-09, Peer-Checking:

4.1 Manipulations requiring Peer Checking

4.1.1 Peer checks shall be used on any equipment having a "peer check" demarcation, which is orange tape/sticker on or around the component, or DCS control orange pushbuttons. The permanently marked peer checked components have been selected because of the significant impact they could have to reactivity control, plant safety and reliability. Attachment 1 - Components Marked for Peer Checks and Attachment 2 - Substation Components Marked for Peer Checks contain a listing of components marked for peer checking.

From OPD-3-09, Peer-Checking, Attachment 1:

CB-4

HC-102-1A PORV Test Switch A
HC-102-1 B PORV Test Switch B
Demin Water Makeup Flow Batching Switch FQS-269X
Boric Acid Makeup Flow Batching Switch FQS-269Y
HCV-438 A/C AC Inlet/Outlet Valves
HCV-438 B/D AC Inlet/Outlet Valves
Control Rod Group Selector Switch
Rod Control Mode Selector Switch
SGLS Block
Manual Rod Control Switch (raise/lower)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	1	
	K/A #	2.1.1	
Level of Difficulty: 3	Importance Rating	3.8	

Conduct of Operations: Knowledge of conduct of operations requirements.

Question: 68

Given the following condition:

- A Reactor Protection System Trip Unit is placed in BYPASS and is NOT directed by an approved procedure.

Which of the following groups, if notified, meets the MINIMUM notification requirements in accordance with SO-O-28, Requirements for Administratively Bypassing Reactor Protective System Trip Units?

A. The NRC Resident Inspector, AND
The System Engineering Manager.

B. The NRC Operations Center, AND
The Plant Manager.

C. The NRC Resident Inspector AND
Shift Operations Superintendent.

D. The NRC Operations Center, AND
Director – Site Operations.

Answer: C

K/A Match:

Applicant must be familiar with requirements of Operations Standing Orders.

Explanation:

A. Incorrect. Plausible because the NRC Resident Inspector must be notified, and because the Engineering Programs department should evaluate the PRA effect of bypassing RPS trip units outside of procedural guidance; incorrect because System Engineering Manager notification is not required by SO-O-28.

B. Incorrect. Plausible the Plant Manager must be notified, and because The NRC Resident must be notified; incorrect because the NRC Resident must be notified instead of NRC Operations Center.

C. **Correct**. As outlined SO-O-28, Step 5.1, the NRC Resident Inspector and the Plant Manager,

Director of Site Operations, or the Shift Operations Superintendent must be notified.

D. Incorrect. Plausible because the NRC Resident and Plant Manager notification would meet the minimum requirements for notification in accordance with the procedure, Incorrect because the NRC Operations Center is not required to be notified.

Technical Reference: SO-O-28, Step 5.1, Rev. 10
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-25, Reactor Protection System-Licensed Operator
EO 1.19 - **EXPLAIN** the notifications required by the Standing Orders if an RPS trip unit is placed in a tripped or bypassed condition.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

1.0 PURPOSE AND SCOPE

To specify notification requirements if an RPS trip unit bypass must be affected.

2.0 STATEMENT OF APPLICABILITY

This standing order applies to the use of any one of the twelve (12) bypass keys for the trip units on the RPS cabinets.

3.0 DEFINITIONS

None

4.0 RESPONSIBILITIES

Shift Manager - It is the responsibility of the Shift Manager on duty to ensure the requirements of this procedure are met.

5.0 PROCEDURE

5.1 The Manager-Fort Calhoun Station, Director Site Operations or Shift Operations Superintendent must be notified if a bypass must be affected on any RPS trip unit. No notification is required if such bypassing is required to perform a PRC approved procedure (for example a surveillance test or PRC approved operating or maintenance procedures).

5.2 The RPS trip unit bypass keys must be maintained in the Operations Key Depository in the Control Room when not in use to affect an authorized bypass.

5.3 Issue of RPS trip unit bypass keys will be in compliance with Standing Order SO-O-26 (Plant Keys).

5.4 The use of RPS trip unit bypass keys must be logged in the control room log.

5.5 The NRC Resident Inspector must be notified of unexpected use of RPS bypass keys.

Examination Outline Cross-reference:		Level	RO	SRO
Rev. Date: 08/30/15		Tier #	3	
Change: 1		Group/Category #	2	
		K/A #	2.2.13	
Level of Difficulty: 2		Importance Rating	4.1	

Equipment Control: Knowledge of tagging and clearance procedures.

Question: 69

Given the following conditions:

- The Plant is in MODE 5 for a Refueling outage.
- Electrical Maintenance needs to remove a 480 V breaker from its breaker cubicle and move it to the Maintenance Shop for refurbishment.
- The breaker is currently racked out with a DANGER tag on the breaker.

Which is the proper Clearance change for the breaker tag?

The DANGER Tag must …

A. …be attached to the breaker cubicle door.

B. …be moved to the Control Switch for the component served by the breaker.

C. …be released and removed from the Clearance.

D. …remain with the breaker so that is returned to the correct cubicle.

Answer: A

K/A Match:
Applicant must be familiar with tagging and clearance procedures including requirements to move tags.

Explanation:

A. **Correct**. Similar to the action required for a fuse block, the danger tag should be removed from the breaker and attached to the cubicle door to prevent a different untagged breaker from being inserted.
B. Incorrect. Plausible because the control switch would be the normal location for operating the breaker. Incorrect because the breaker could be closed locally once reinstalled and endanger personnel.
C. Incorrect. Plausible if thought this was acceptable. Incorrect because another Clearance would have to be established to tag out the breaker cubicle.
D. Incorrect. Plausible because the breaker was the component that was tagged. Incorrect because it does not prevent another breaker from being inserted in the cubicle.

Technical Reference: OP-FC-109-101, Rev. 0
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan OP-FC-109-101, FCS Clearance and Tagging-Licensed Operator
EO 11 - **EXPLAIN** the process for performing work under the protection of a Clearance.

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

Question History: Last NRC Exam 2014 NRC Exam, Question #70

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

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8. Plant Operations and the work group shall **EVALUATE** grounds that are lengthy to determine if both ends require tagging. Normally only one tag per ground cluster is required.

7.3.11. Fuse Standards

1. **When** a fuse is removed to provide C/O isolation, **then** the fuse shall be replaced with a non-conductive clip or other Tags Plus device that prevents insertion of a fuse when C/O is hanging.

2. **ATTACH** the tag to the fuse clip or place it as close to the clip as possible. **If** a non-conductive fuse block out device is <u>not</u> used, **then** the fuse shall be identified and stored for restoration when the C/O is removed, and a Tag will be placed such that another fuse cannot be accidentally installed.

3. **EXERCISE** care to insure that the size and rating of fuse removed during C/O activities is the same as the fuse that gets reinstalled during restoration.

4. Breaker control power fuses should only be tagged when they are included as part of the zone of protection.

5. When tagging a fuse block to provide C/O isolation, then the fuse block shall be controlled in one of the following manners.

   A. Place the fuse block (and fuses) in a bag identified with a copy of the Danger Tag or information traceable back to the C/O, store in a safe location, and place the actual Danger Tag on a wire terminated at the base of the stationary portion of the fuse block, covering the fuse block to prevent other fuses from being installed.

   B. Turn the removable portion of the fuse block 180 degrees to the OFF position and reinsert into the stationary portion of the fuse block, with the Danger Tag placed on handle of the fuse block.

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4. The duration of the authorization to manipulate SCT's should be limited by the Shift Management individual granting the authorization.

5.3.6. SCT's shall not be removed until the administrative requirements of this procedure have been met.

5.3.7. An SCT shall not be applied to a component bearing another SCT.

5.3.8. An SCT can be applied to a component bearing an Information Tag provided the component positions do **not** conflict. The SCT shall **not** be obstructed by the Information Tag

5.3.9. An SCT shall not be applied to a component bearing a Danger Tag. It is permissible to have a Danger Tag applied to the breaker compartment (i.e. the bus to line disconnect) and SCT's applied to the breakers control power fuses

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Revision 0
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5.2. Danger Tags

5.2.1. A Danger Tag will override all other Tag types in use at the facility.

5.2.2. A component with a Danger Tag attached to it shall **not** be physically removed from the system.

5.2.3. Danger Tags shall **not** be physically removed and/or equipment manipulated until the administrative requirements of this procedure have been met.

5.2.4. A Danger Tag can be applied to a component bearing another Danger Tag, or an Information Tag provided the component positions do **not** conflict. The Danger Tag shall **not** be obstructed by the Information Tag.

5.2.5. A Danger Tag shall **not** be applied to a component bearing a local leak rate test (LLRT) Tag, integrated leak rate test (ILRT) Tag, or Hydro Tag **if** the required component positions conflict.

5.2.6. A Danger Tag shall **not** be used to tag a breaker in the energized state.

5.2.7. A Danger Tag should **not** be used to tag a valve open to maintain a pressurized state, unless the pressure is being used to maintain isolation (e.g., air supply to a line stop bladder).

5.3. Special Condition Tags

5.3.1. SCT's have equal authority to Danger tags.

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3. C&T EEI Standards are not required if the component is removed from the Zone of Protection of the Clearance, either by physical removal or separation. EEI safety practices still need to be employed as listed in section 7.3.13.1 above. Examples of when Attachment 16 is not required:

   A. Transformer work with disconnects or bus work physically removed within line of sight of the transformer.

   B. Meggering of bolts on a pump with insulated flange connections.

   C. AOV maintenance and testing with an external air supply or INFO or Not Tagged air supply valve.

   D. Equipment not installed in the ZOP like a motor or pump in the shop, a breaker removed from its cubicle, or equipment electrically disconnected.

5. **CLEARANCE TAG STANDARDS**

5.1. General Standards

5.1.1. **No** device or equipment shall be operated while C/O or WTO Tags are attached with the following exceptions:

1. Qualified Operations personnel (or qualified FM Technician for WTO being used by FM) performing position verification while placing C/O's or WTO's.

2. Qualified Operations personnel to improve isolation. Prior to this manipulation, permission must be obtained from Shift Management and all Holders who have accepted the C/O or may be affected by the evolution. **If** the component is Danger Tagged, **then** the component shall only be moved in the direction of the "as applied position".

3. Qualified FM Technician to improve isolation of a FM WTO. Prior to this manipulation, permission must be obtained from the FM Supervisor. If the component is Danger Tagged, then the component shall only be moved in the direction of the "as applied position".

4. Qualified personnel manipulating Information Tagged equipment.

5. Qualified personnel manipulating SCT tagged equipment

5.1.2. All C/O Tags shall contain the following information as a minimum:

1. Component ID Number or noun name when component ID number is **not** available.
2. Component Description.
3. Position or status of component.

5.1.3. Clearance Tags should be placed on or as close to the component being controlled as possible to avoid confusion as to what point the Tag is controlling. Temporary attachment devices can be used or with proper Engineering approvals permanent attachment devices or holes can be employed.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/13/15	Tier #	3	
Change: 1	Group/Category #	2	
	K/A #	2.2.1	
Level of Difficulty: 3	Importance Rating	4.5	

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Question: 70

Given the following conditions:

- All CEAs are fully inserted and preparations are being made to perform a Reactor Startup by CEA withdrawal.
- According to the Estimated Critical Condition (ECC) calculation, the boron concentration should be reduced by 250 ppm.

According to OP-2A, Plant Startup, Attachment 1, CEA Withdrawal to Criticality Mode 2, which one of the following sequence of steps is performed?

A. 1.) Dilute to ECC boron concentration.
   2.) Withdraw the Non-Trippable CEAs.
   3.) Withdraw the Shutdown CEAs.
   4.) Withdraw the Regulating CEAs.

B. 1.) Dilute to ECC boron concentration.
   2.) Withdraw the Shutdown CEAs.
   3.) Withdraw the Non-Trippable CEAs.
   4.) Withdraw the Regulating CEAs.

C. 1.) Withdraw the Shutdown CEAs.
   2.) Dilute to ECC boron concentration.
   3.) Withdraw the Non-Trippable CEAs.
   4.) Withdraw the Regulating CEAs.

D. 1.) Withdraw the Non-Trippable CEAs.
   2.) Dilute to the ECC boron concentration.
   3.) Withdraw the Shutdown CEAs.
   4.) Withdraw the Regulating CEAs.

Answer: C

K/A Match:

Applicant must be familiar with reactivity sequencing (rods and boron) in pre-startup procedures. Question is higher cognitive level because if a Boration is required is performed before any CEAs are withdrawn. This ensures adequate SHUTDOWN MARGIN in the event of an uncontrolled dilution.

Explanation:

A. Incorrect. Plausible because if a Boration were required it would be performed first. Incorrect because the Shutdown CEAs are withdrawn before the RCS is diluted.

B. Incorrect. Plausible because if a Boration were required it would be performed first and the Rod withdrawal sequence is correct. Incorrect because the Shutdown CEAs are withdrawn before the RCS is diluted.

C. **Correct**. This is the correct sequence as outlined in OP-2A.

D. Incorrect. Plausible because the Rod withdrawal sequence is correct. Incorrect because the RCS is diluted after the Shutdown CEAs are withdrawn.

Technical Reference: OP-2A, Plant Startup, Attachment 2, Steps 1 through 14, Rev. 123

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 8-11-9, Reactor Startup- Licensed Operator
EO 5.2 - The **CRS/ATCO** will withdraw shutdown and non-trippable CEAs.

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

Question History: Last NRC Exam 2014 NRC Exam, Question #68

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

10 CFR Part 55 Content: 55.41 10
55.43 ______

PROCEDURE

1. Review the Estimated Critical Condition Worksheet for Critical CEA position AND their ± 0.5% Δρ values calculated in TDB-V.1b.

CAUTION
Only one method of positive reactivity addition shall be used at any one time.

2. IF required, THEN borate the RCS per OI-CH-4 until ECC Boron Concentration is obtained.
3. Place each Zero Power Mode Bypass Switch on the RPS cabinets to OFF and verify the following:
   - Lo Flow light goes out for each channel
   - TM/LP Bypass light goes out for each channel
   - The Zero Power Mode Bypassed annunciator clears (CB-4, A20)
4. Record Base Count Rate (CR_i) and Count Rate of 4 doublings (CR_f) obtained in OI-CH-11.

   CR_i__________CPS CR_f__________CPS
5. Continue to monitor source range count rate for four doublings (i.e. CR = CR_f) until 1/m plotting has commenced
6. Withdraw Shutdown Groups A and B to ARO using the Manual Group (MG) mode per OI-RR-1.

NOTE
Approximately one hour should be given prior to obtaining a RCS sample to allow proper mixing.

7. IF required, THEN dilute the RCS per OI-CH-4 until ECC Boron Concentration is obtained.
8. IF not withdrawn, THEN withdraw all Non-Trippable CEAs to ARO using the Manual Group (MG) mode per OI-RR-1.
9. Commence Reactor Engineer Criticality Log (Figure 2).
10. Verify all individual Shutdown and Non-Trippable CEAs positions indicate fully withdrawn by Individual CEA Position indicators.
11. RCS Hydrogen concentration is greater than or equal to 15 cc/kg.
12. Withdraw regulating CEA Groups 1, 2, 3, and 4 to four inches using Manual Individual Control.
13. Commence electronic 1/M Plots per Attachment 2A.
    a. **IF** electronic 1/M plot is unavailable, **THEN** manually plot the 1/M graph.

CAUTION
The Reactor shall be considered critical when there is sustained rising flux level (positive Startup Rate) with no CEA movement OR Reactor Power indicates greater than 10^{-4}%.

14. Using the Manual Sequential (MS) mode, take the Reactor critical by withdrawing the Regulating CEA Groups per OI-RR-1 as follows:

    IF during withdrawal, the reactor goes critical, THEN proceed to Step 15.

    a. Withdraw Group 1 to 90 inches.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	2	
	K/A #	2.2.43	
Level of Difficulty: 3	Importance Rating	3.0	

Equipment Control: Knowledge of the process used to track inoperable alarms.

Question: 71

Given the following conditions:

- An alarm card is being pulled due to a nuisance condition caused by a faulty pressure switch.
- A new pressure switch has been ordered and is expected to be installed in 7-10 days.

What actions must be performed along with pulling the alarm card?

A. An Annunciator Card Status Form must be filled out and forwarded to the Work Week Manager.

B. An Annunciator Card Status Form must be filled out and placed in the associated Annunciator Response Procedure book.

C. A blue flag (dot) must be posted on the annunciator window and the associated Alarm Response Procedure must be (or have been) reviewed.

D. An Annunciator Status Tag must be posted on the annunciator window and the associated Alarm Response Procedure must be reviewed for Compensatory Actions.

Answer: D

K/A Match:

Applicant must identify requirements for tracking an inoperable annunciator alarm.

Explanation:

A. Incorrect. Plausible because the Work Week Manager is informed per ARP-1 but only when a Loss of Annunciators is in progress. The Annunciator Card Status Form is completed by the Shift Manager/Control Room Supervisor.

B. Incorrect. Plausible because an Annunciator Card Status Form is filled out for alarm card deactivation but is not required to be placed with the associated ARP book.

C. Incorrect. Plausible because a blue flag (dot) is used but their purpose is to identify annunciators that are normally on during power operation.

D. **Correct**. As required per ARP-1, Section 5.4.

Technical Reference: ARP-1, Step 5.4.3, Rev. 29
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-11, Use of ARP-1-Licensed Operator
EO 1.0 - **DESCRIBE** operator actions for in annunciator in alarm.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	10
	55.43	

NOTE: The Annunciator Status Tag Number shall be used as the Control Number for Attachment 1, Annunciator Card Status Form.

NOTE: An Annunciator Status Tag is not required for ERFCS/DCS alarm points. The alarm point name (e.g., T1209A) shall be used for Attachment 4, ERFCS/DCS Alarm Point Status Form.

NOTE: Attachment 4, ERFCS/DCS Alarm Point Status Form, shall be used for any ERFCS/DCS alarm point that is removed from scan (ERFCS) or placed in Manual (DCS). SO-O-32 requirements still apply for all ERFCS/DCS points.

5.4.3 Initiate Attachment 1, Annunciator Card Status Form, for a de-activated annunciator card or Attachment 4, ERFCS/DCS Alarm Point Status Form for a disabled ERFCS/DCS point, as follows.

5.4.3.A Complete Section 1 for each annunciator card or alarm point that is to be deactivated.

5.4.3.A.1) Section 1 may be completed by an Operator or Maintenance technician.

5.4.3.A.2) The remainder of Attachment 1 is to be completed by the Shift Manager or his designee.

5.4.3.B An Annunciator Status Tag is attached to the associated annunciator window, as applicable. The ERFCS alarm point removed from scan or DCS alarm point placed in Manual shall be annotated on Form FC-125, Log of Computer Points Removed from Scan in accordance with SO-O-32, Plant Computer.

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ANNUNCIATOR RESPONSE PROCEDURE PAGE 13 OF 20

assessment for the duration of the maintenance is completed and maintenance will not take more than 90 days. A 50.59 Review is not required to be performed unless the 90 day period is exceeded or expected to be exceeded. See SO-M-100 for risk assessment guidance.

5.4.3.C.3) An FC-154A, 10 CFR 50.59 Screening, will be required for the deactivation of an annunciator card(s) whether or not compensatory actions are required unless the compensatory actions are part of an approved procedure.

5.4.3.C.4) If it is determined that an FC-154B, 10 CFR50.59 Evaluation, is required, the evaluation concludes that the activity would be allowed per plant procedures without obtaining a License Amendment.

5.4.3.D Complete Section 3 - Alarm Card De-Activated

5.4.3.D.1) Review the applicable ARP for defined compensatory actions associated with the deactivation of the alarm card.

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ANNUNCIATOR RESPONSE PROCEDURE PAGE 3 OF 20

4.4 Alternative actions which are directed by other procedures in response to specific conditions or combinations of annunciators in alarm are not prohibited by the ARPs.

4.5 Annunciator windows that are normally on during power operation are designated with a bold B on the annunciator lampbox drawing. Normally on annunciators are designated on the individual response pages by having the window outlined with a double line; in addition, normally on annunciators have a small blue dot located in the lower right hand corner of the annunciator window.

5.2.9 An Operator should not simultaneously depress the alarm acknowledgment and reset buttons on any operating panel. Such practice could potentially mask the receipt of other unidentified alarms and prevent the implementation of appropriate Operator actions.

5.2.10 If an annunciator window is normally expected to be in alarm but found to be in a clear condition, then the Operator should consult the applicable ARP as an aid to event diagnosis.

5.3 Loss of Annunciator Power

5.3.1 Perform the following in the event of loss of power to one or more annunciator panels:

5.3.1.A Walkdown the control boards and take actions per the applicable operating procedure or instruction to stabilize the plant and control equipment.

5.3.1.B IF approximately 75% of the annunciators associated with safety systems are lost, THEN implement the Emergency Plan.

5.3.1.C Initiate a Work Request.

5.3.1.D Notify the Work Week Manager to initiate repairs.

5.3.1.E Initiate a Condition Report.

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Attachment 1 - Annunciator Card Status Form

SECTION 1

Control Number: ____________ (Annunciator Status Tag Number)	Date: / /	Requestor:
Panel:	Annunciator:	Window:

SECTION 2

Alarm Card De-Activation: (Any one answered YES allows card de-activation). (1)	YES	NO
1. Compensatory actions are part of an approved procedure.		
2. Maintenance is being performed and a maintenance risk assessment has been completed per SO-M-100, Attachment 7.1.		
3. An FC-154A, 50.59 Screen, for the de-activation of the alarm card and the identified compensatory action(s), as warranted, have been initiated. (Attach completed 50.59 Screening(s) to this form.)		
4. An FC-154B, 50.59 Evaluation, has been completed and allows the activity per plant procedures without obtaining a License Amendment.		

SECTION 3

Alarm Card De-Activated:				
List all referenced material reviewed to determine need for compensatory actions (T.S., UFSAR, EOPs and AOPs, ARPs, Radiological Controls): N/A If compensatory actions are part of an approved procedure.				
Compensatory actions required per Shift Manager/Control Room Supervisor:				
IF YES, THEN initiate Attachment 2, Compensatory Action Sheet.				
FC-68J, Procedure Change Request submitted to incorporate compensatory actions as warranted or identify none required.				
Fill in the date and time the Annunciator Card was pulled: Date: / / Time:				
Reason Annunciator Card is to be pulled or left locked in (check one):			Identify one of the following for restoration:	
Nuisance annunciator			Procedure	
Maintenance			Work Process	
Problem annunciator			MWR No.	

_______________________________ / / _______ ______

Shift Manager / Control Room Supervisor Date Time

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	3	
	K/A #	2.3.4	
Level of Difficulty: 3	Importance Rating	3.2	

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Question: 72

Given the following conditions:

- A room in the Auxiliary Building has general area radiation levels of 110 mrem/hour.
- A valve on the far wall of the room has a contact radiation level of 900 mrem/hour.
- The radiation level 30 centimeters from the valve is 75 mrem/hour.

Which of the following is the correct posting for the room?

A. "CAUTION, RADIATION AREA"

B. "CAUTION, HIGH RADIATION AREA"

C. "CAUTION, LOCKED HIGH RADIATION AREA"

D. "GRAVE DANGER - VERY HIGH RADIATION AREA"

Answer: B

K/A Match:

Applicant must be able to identify displayed signage based on exposure limits and radiation levels.

Explanation:

A. Incorrect. Posting used for an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

B. **Correct**. Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate in excess of 100 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

C. Incorrect. Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate greater than or equal to 1000 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

D. Incorrect. Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or one meter from any surface that the radiation penetrates.

Technical Reference: RP-AA-18, Steps 5.2.3 to 5.2.6, Rev. 1
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator
EO - **STATE** the Requirements of Section 5, Administrative Controls: High Radiation and Restricted High Radiation Area Controls

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 12
55.43 ______

5.1. Postings for Access to RCAs

5.1.1. For entries to RCAs (e.g., power block, radwaste, etc.), the following is the minimum level of posting: "Caution - Radioactive Material."

5.1.2. For some locations based upon radiological conditions, "Caution - Radiation Area" may be used in lieu of the "Caution - Radioactive Material" posting specified above.

5.1.3. Additional information may be required on the minimum posting based on current survey information or at the discretion of the Radiation Protection Department (e.g., "No Eating, Drinking or Smoking Permitted," etc.).

5.2. Postings within the RCA

5.2.1. General Provisions for Postings

1. RCAs shall be conspicuously posted so as to warn personnel approaching the area from any direction and reflect the radiological condition of an area.

5.2.2. "Caution - Radioactive Material" – Posting used for an area or room in which there is used or stored an amount of licensed radioactive material exceeding ten times the quantity of such material specified in 10 CFR 20 Appendix C.

5.2.3. "Caution - Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

5.2.4. "Caution - High Radiation Area" or "Danger – High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate in excess of 100 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

5.2.5. "Caution - Locked High Radiation Area" or "Danger – Locked High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent rate greater than or equal to 1000 mrem/hr at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

5.2.6. "Grave Danger - Very High Radiation Area" – Posting used for an area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or one meter from any surface that the radiation penetrates.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	3	
	K/A #	2.3.13	
Level of Difficulty: 3	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.

Question: 73

After exiting a Locked High Radiation Area, what steps must be taken to ensure the door is secure in accordance with RP-AA-460, Controls for High and Locked High Radiation Areas?

A. Contact Security to ensure the door is adequately secured and notify Radiation Protection when completed.

B. Person exiting the room should check the door and document the check.

C. Operator must verify door is shut and locked, with the Shift Manager providing independent verification.

D. Radiation Protection and an independent verifier will ensure the door is closed, locked, and physically challenged.

Answer: D

K/A Match:
Applicant must be knowledgeable of the entry and access requirements for locked high radiation areas in the plant, including restoration of access control.

Explanation:

A. Incorrect. Plausible because Security is routinely called to verify access throughout the plant particularly for doors that are normally alarmed. Incorrect because Radiation Protection must verify that the door is locked.
B. Incorrect. Plausible because that person is documenting the check. Incorrect because it does not meet requirements.
C. Incorrect. Plausible because an independent verification is required and the Shift Manager could be that person. Additionally, the Shift Manager is one of the people who maintains administrative control of the keys per Technical Specification 5.11.2. Incorrect because Radiation Protection must verify that the door is locked.
D. **Correct**. Per procedure, Radiation Protection and a person other than the individual that entered the locked high radiation area must verify the doors closed, locked, and physically challenged.

Technical Reference: RP-AA-460, Step 4.4.5.5, Rev. 26
(Attach if not previously provided including revision number) Technical Specification 5.11.2, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 19-24-3, Radiation Protection Plan Familiarization- LO EO 1.10 - **STATE** the programs to be implemented by the Radiation Protection Department and **DESCRIBE** the overall implementation as per the RPP to include the following program: ALARA.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 12
55.43

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5. When all personnel have exited the area, Access Control Guard **ENSURES** the following.

A. RP is **NOTIFIED** that all personnel have exited the area.

B. **REMAINS** at the LHRA Access until RP Personnel arrive.

C. RP Personnel **and** Access Control Guard shall **VERIFY** via individual physical challenge that the LHRA Access is closed **and** locked.

D. **DOCUMENT** physical challenges to LHRA closure via Attachment 10 or CGE **and** on Attachment 7 or CGE.

TECHNICAL SPECIFICATIONS

5.0 **ADMINISTRATIVE CONTROLS**

5.10 Record Retention

5.10.1 Records shall be retained as described in the Quality Assurance Program.

5.11 Radiation Protection Program

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

5.11.1 In lieu of the "control device" required by paragraph 20.1601(a) of 10 CFR Part 20, and as an alternative method allowed under § 20.1601(c), each high radiation area (as defined in § 20.1601) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by required issuance of a Radiation Work Permit.* Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Manager-Radiation Protection (MRP) in the Radiation Work Permit.

5.11.2 The requirements of 5.11.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr** but less than 500 rads/hr*** (Restricted High Radiation Area). In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Manager on duty and/or the MRP with the following exception:

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	4	
	K/A #	2.4.3	
Level of Difficulty: 3	Importance Rating	3.7	

Emergency Procedures/Plan: Ability to identify post-accident instrumentation.

Question: 74

Which of the following instruments are required by Technical Specification LCO 2.21, Table 2-10, Post-Accident Monitoring Instrumentation?

1. LT-387, Containment Wide Range Level
2. RM-091A, Containment Wide Range Area Radiation Monitor
3. PIC-785, Containment Narrow Range Pressure

A. 1, 2, and 3

B. 1 and 2 only

C. 1 and 3 only

D. 2 and 3 only

Answer: B

K/A Match:
Applicant is required to identify Containment Post-Accident Monitoring Instrumentation.
Explanation:
A. Incorrect. Containment Narrow Range Water Level and Wide Range Radiation Monitor are correct, but Containment Narrow Range Pressure is incorrect.
B. **Correct**. As identified in Technical Specification LCO 2.21, Table 2-10.
C. Incorrect. Containment Narrow Range Water Level is correct. Containment Narrow Range Area Pressure is incorrect.
D. Incorrect. Containment Wide Range Area Radiation Monitor is correct. Containment Narrow Range Pressure is incorrect.

Technical Reference: (Attach if not previously provided including revision number)
LP 7-11-8, Slide #68 & #76, Rev. 1
LP 7-12-3, Slide #288 & #303, Rev. 1
Technical Specification LCO 2.21, Table 2-10, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator
EO 1.0 - **STATE** what plant equipment is covered by the Limiting Conditions for Operations.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 10
55.43 ____

Major Component Description **(Slide #68)**

Containment Instrumentation

Wide Range and Narrow Range Containment Pressure Indication is available on the Post Accident Monitoring Panels AI-65A and AI-65B.

AI-65A

(a) PIC-785 (-5 to +5 psig) NR

(b) PIC-783 (-5 to +195 psig) WR

(c) PR-783/785, two-pen recorder provided for NR & WR pressures.

AI-65B

(a) PIC-786 (-5 to +5 psig) NR

(b) PIC-784 (-5 to +195 psig) WR

(c) PR-784/786, two-pen recorder provided for NR & WR pressures.

Major Component Description **(Slide #76)**

Level Indication and Recorders

LI-599-1 and LI-600-1, Containment Sump Water Level are located on the Post Accident Monitoring Panels AI-65A and AI-65B (5-37").

LI-387-1 and LI-388-1, Post Accident Water Level, are located on AI-65A and AI-65B (0-27'6").

LR-387/599 and LR-388/600, Containment Sump Water Level, recorders are located on AI-65A and AI-65B (two-pen recorders).

EO *7.2 **(Slide #303)**

RM-091A and B

Post-Accident High Range Monitors, RM-091A and B are located on the 1045' elevation in Containment.

The detectors are ionization chambers with a range of 1 to 10 E7 R/hr.

RM-091A is powered from 120 VAC instrument bus AI-40A.

RM-091B is powered from 120 VAC instrument bus AI-40B

EO 4.0, **(Slide #288)**

Normal Range Area Radiation Monitors

Control Room ratemeters are arranged in groups on AI-33B based on their location in the plant.

RM-070 thru RM-075 are located in Containment.

TECHNICAL SPECIFICATIONS

TABLE 2-10

Post-Accident Monitoring Instrumentation Operating Limits

	Instrument	**Minimum Operable Channels**	**Action**
1.	Containment Wide Range Radiation Monitors (RM-091A & B)	2	(a)
2.	Wide Range Noble Gas Stack Monitor RM-063 (Noble Gas Portion Only)	1	(a)
3.	Main Steam Line Radiation Monitor (RM-064)	1	(a)
4.	Not Used		
5.	Containment Water Level		
	Narrow Range (LT-599 & LT-600)	1	(d)
	Wide Range (LT-387 & LT-388)	2	(b)(c)
6.	Containment Wide Range Pressure	2	(b)(c)
7.	Reactor Coolant System Subcooled Margin Monitor	2	(e)(f)
8.	Core Exit Thermocouples (i)	2/Core Quadrant	(g)(h)
9.	Reactor Vessel Level (HJTC) (j)	2	(k)(l)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/12/15	Tier #	3	
Change: 0	Group/Category #	4	
	K/A #	2.4.32	
Level of Difficulty: 3	Importance Rating	3.6	

Emergency Procedures/Plan: Knowledge of operator response to loss of all annunciators.

Question: 75

Given the following condition:

- Plant is in MODE 2 at 1% power when a Loss of All Annunciators occurs.

Which of the following is required per ARP-1, Annunciators?

A. Implement the Emergency Plan.

B. Perform a controlled Reactor Shutdown in accordance with OP-3A, Plant Shutdown.

C. Establish a Dedicated Operator for each Control Board section in accordance with SO-O-1, Conduct of Operations.

D. Trip the Reactor and enter EOP-00, Standard Post Trip Actions.

Answer: A

K/A Match:

Applicant must be familiar with procedure for Loss of Annunciator Power.

A. **Correct**. As outlined in ARP-1, Step 5.3, Loss of Annunciator Power.

B. Incorrect. There is no requirement to shutdown the Reactor for a loss of annunciators.

C. Incorrect. Plausible because Dedicated Operator is defined in SO-O-1. Incorrect because a dedicated operator is used to maintain operability of required Technical Specification equipment, and in this condition, no Technical Specification actions are required.

D. Incorrect. Plausible if thought that the plant was in an unanalyzed condition. Incorrect as this is not required by ARP-1.

Technical Reference: ARP-1, Step 5.3, Rev. 29

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-11, Use of ARP-1-Licensed Operator
EO 1.0 - **DESCRIBE** operator actions for in annunciator in alarm.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 10
55.43 ____

FORT CALHOUN STATION **INFORMATION USE** ARP-1
ANNUNCIATOR RESPONSE PROCEDURE PAGE 10 OF 20

5.2.9 An Operator should not simultaneously depress the alarm acknowledgment and reset buttons on any operating panel. Such practice could potentially mask the receipt of other unidentified alarms and prevent the implementation of appropriate Operator actions.

5.2.10 If an annunciator window is normally expected to be in alarm but found to be in a clear condition, then the Operator should consult the applicable ARP as an aid to event diagnosis.

5.3 Loss of Annunciator Power

5.3.1 Perform the following in the event of loss of power to one or more annunciator panels:

5.3.1.A Walkdown the control boards and take actions per the applicable operating procedure or instruction to stabilize the plant and control equipment.

5.3.1.B IF approximately 75% of the annunciators associated with safety systems are lost, THEN implement the Emergency Plan.

5.3.1.C Initiate a Work Request.

5.3.1.D Notify the Work Week Manager to initiate repairs.

5.3.1.E Initiate a Condition Report.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 1	Group/Category #		1
	K/A #	011 EA2.01	
Level of Difficulty: 3	Importance Rating		4.7

Large Break Loss of Coolant Accident: Ability to determine or interpret the following as they apply to a Large Break LOCA: Actions to be taken, based on RCS temperature and pressure - saturated and superheated.

Question: 76

Given the following conditions:

- A Plant trip occurred due to a Loss of Coolant Accident.
- EOP-03, Loss of Coolant Accident, is in progress.
- Loop 1 T_{COLD} is 348ºF and Loop 2 T_{COLD} is 349ºF.
- Representative Core Exit Thermocouple (CET) temperature is 420ºF.
- Reactor Vessel Level Monitoring System indicates 43%.
- Pressurizer pressure is 300 psia.
- Pressurizer level is 40%.
- Steam Generator narrow range levels are 55%.
- Containment pressure is 6 psig and lowering.
- Recirculation Actuation Signal has occurred.

It is desired to place the Shutdown Cooling (SDC) System in service.

Per EOP-03, Loss of Coolant Accident, which of the following must be performed prior to implementing HR-25, SDC with RAS?

A. Reduce Pressurizer pressure using Auxiliary Spray per Attachment PC-11, Pressure Control.

B. Restore RCS subcooling and Pressurizer level to greater than 45% per Attachment IC-14, RCS Void Elimination.

C. Lower Steam Generator pressure to maintain both Loops T_{COLD} less than 300°F using Attachment HR-12, Secondary Heat Removal Operation.

D. Raise Steam Generator narrow range levels to between 85% and 89% per Attachment HR-11, Main Feed Control (DCS).

Answer: B

K/A Match:

As the SRO, applicant must be cognizant of conditions required to enter Shutdown Cooling during a Large Break LOCA when superheated conditions exist in the core. Question tests if the SRO, using specific procedure content knowledge, can determine the actions required by one EOP before performing another EOP.

Explanation:

A. Incorrect. Plausible because RCS pressure must be maintained ≤ 300 psia and PC-11 would be the correct procedure to perform that action. Incorrect because lowering pressure would increase superheat. Subcooling must be ≥ 20°F in order to meet SDC entry requirements.

B. **Correct**. Superheat conditions currently exist. Implementing IC-14 will eliminate voids in the head and restore RVLMS level. When that is accomplished, RCS subcooling can be restored. Pressurizer level must be ≥ 45% constant or rising. Using IC-14 eliminates voids in the Steam Generators (Step 1) or the Reactor Vessel (Step 2).

C. Incorrect. Plausible because RCS T_{COLD} must be less than 350°F to place SDC in service. Incorrect because Reactor Vessel voiding is occurring, inadequate subcooling exists, and Pressurizer level is not ≥ 45%, and RCS pressure must be less than 300 psia.

D. Incorrect. Plausible because raising level would ensure that RCS T_{COLD} temperatures remain less than 350°F. Incorrect because RCS is superheated at this time.

Technical Reference: (Attach if not previously provided including revision number)
- EOP-03, Step 62, Rev. 38
- EOP/AOP Attachments-IC, Attachment IC-14, Rev. 1
- EOP/AOP Attachments-HR, Attachments HR-11 & HR-12, Rev. 1
- EOP/AOP Attachments-PC, Attachment PC-11, Step 1.1 CA, Rev. 0

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-13, Loss of Coolant Accident-Licensed Operator
EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of a LOCA.
EO 3.0 - **DEMONSTRATE** the knowledge required to implement the Floating Steps of EOP-03, LOCA, when needed.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
- 55.41 ____
- 55.43 5

EOP-03, Step 62

62. WHEN ALL of the following SDC entry conditions are established:

- PZR level is greater than or equal to 45% and constant or rising
- RCS subcooling is greater than or equal to 20°F
- RCS pressure is less than or equal to 300 psia
- RCS T_C less than 350°F

THEN initiate SDC operation PER **ONE** of the following attachments:

- Attachment HR-24, SDC without RAS
- Attachment HR-25, SDC with RAS
- Attachment HR-29, Cooled SI Flow with RAS

Time: ___________

PC-11, Step 1.1 CONTINGENCY ACTION

1.1 IF SIAS has initiated

AND pressurizer heaters are required,

THEN perform the following:

1) Place **ALL** Backup Heater control switches in "OFF":

- "225 KW BACKUP HTRS BANK 1 GROUP 1/2/3"
- "150 KW BACKUP HTRS BANK 2 GROUP 4/5"
- "150 KW BACKUP HTRS BANK 3 GROUP 8/9"
- "225 KW BACKUP HTRS BANK 4 GROUP 10/11/12"

(continue)

b. Depressurize the RCS using Aux Spray by operating the following valves as necessary:

- HCV-240, PZR Auxiliary Spray Isolation Valve
- HCV-249, PZR Auxiliary Spray Isolation Valve
- HCV-238, Loop 1 Charging Isolation Valve
- HCV-239, Loop 2 Charging Isolation Valve

c. **IF** HPSI stop and throttle criteria are met, **THEN** control Pressurizer level using **ANY** or all of the following:

- Charging
- Letdown
- HPSI flow

PER Attachment IC-11, Inventory Control.

IC-14, Step 1 NOTE and CAUTION

NOTE

RCS voiding may be indicated by **ANY OR ALL** of the following:

- PZR level rising significantly more than expected while operating PZR spray
- RVLMS indicates Reactor Vessel voiding
- Erratic S/G Δp
- If the RCS cannot be depressurized to SDC entry pressure

****

CAUTION

Void elimination actions may require significant changes in RCS pressure. Do not exceed the pressure-temperature limits of Attachment PC-12, RCS Pressure Temperature Limits, while attempting to eliminate voids.

****

1. **IF** voiding is suspected in the tubes of an isolated S/G,
   THEN eliminate voiding by performing any or **ALL** of the following:

   a. Steam the isolated S/G using Steam Dump and Bypass by performing the following:

IC-14, Step 2

2. **IF** S/Gs are **NOT** isolated **AND** voiding is suspected,
   THEN eliminate RCS voiding by performing the following:
   a. Ensure Letdown is isolated.
   b. Stop RCS depressurization.
   c. Raise RCS pressure to collapse the void using any or **ALL** of the following:
      - PZR Heaters
      - Charging Pumps
      - HPSI Pumps

      PER Attachment PC-11, Pressure Control.
   d. Lower RCS pressure by performing the following:
      1) Deenergize PZR heaters.
      2) Stop the Charging Pumps.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 1	Group/Category #		1
	K/A #	026 G 2.2.37	
Level of Difficulty: 3	Importance Rating		4.6

Loss of Component Cooling Water: Equipment Control: Ability to determine operability and/or availability of safety related equipment.

Question: 77

Given the following conditions:

- AOP-11, Loss of Component Cooling Water (CCW), was entered due to rising CCW temperature.
- Subsequently, CCW Heat Exchanger AC-1C has been removed from service due to a tube leak.

Which of the following describes the operability of the CCW System with respect to a Design Basis Accident?

The CCW System is _____(1)_____. The MAXIMUM amount of time to restore CCW Heat Exchanger AC-1C is _____(2)_____ per Technical Specification LCO 2.4(2)b.

A. (1) OPERABLE
(2) 7 days

B. (1) INOPERABLE
(2) 7 days

C. (1) OPERABLE
(2) 14 days

D. (1) INOPERABLE
(2) 14 days

Answer: C

K/A Match:

As the SRO, applicant will analyze the condition of a system and determine OPERABILITY status based on equipment conditions. It is an SRO ONLY job function to determine OPERABILITY at FCS.

Explanation:

A. Incorrect. Plausible because the CCW System remains OPERABLE. Incorrect because the LCO allows 14 days to restore the CCW Heat Exchanger.

B. Incorrect. Plausible if thought that having one CCW Heat Exchanger out of service rendered the system inoperable. If two (2) CCW Heat Exchangers were inoperable the Reactor would be placed in a HOT SHUTDOWN condition within 12 hours.

C. **Correct**. Per Technical Specification Bases, the CCW System remains OPERABLE with 3 CCW Heat Exchangers available during a Large Break LOCA or Main Steam Line Break inside Containment. This assumption includes that all Containment Air Cooling Units are available and operating.

D. Incorrect. Plausible because the LCO allows 14 days to restore. Incorrect because the system remains OPERABLE. If two (2) CCW Heat Exchangers were inoperable the Reactor would be placed in a HOT SHUTDOWN condition within 12 hours.

Technical Reference: Technical Specification LCO 2.4(2) & Bases, Amendment #283

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-11, Loss of CCW-Licensed Operator
EO 1.6 - **DESCRIBE** the two Technical Specification LCOs that are challenged by a loss of CCW.

Question Source:

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

Question History: Last NRC Exam None

Question Cognitive Level:

Memory or Fundamental Knowledge	X
Comprehension or Analysis	

10 CFR Part 55 Content:

55.41	
55.43	2

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.4 Containment Cooling (Continued)

b. During power operation one of the components listed in (1)a.i. or ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. For cases involving Raw Water pump inoperability, if the river water temperature is below 60 degrees Fahrenheit, one Raw Water pump may be inoperable indefinitely without applying any LCO action statement. When the river water temperature is greater than 60 degrees Fahrenheit, an inoperable Raw Water pump shall be restored to operability within 7 days or the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable Raw Water pump is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. (This does not include: 1) One Raw Water pump which may be inoperable as described above if the river water temperature is below 60 degrees Fahrenheit or, 2) SI-3A and SI-3B being simultaneously inoperable; or 3) VA-3A and VA-3B, or VA-7C and VA-7D, being simultaneously inoperable. Only two raw water pumps may be out of service during power operations. Either containment spray pump, SI-3A or SI-3B, must be operable during power operations. One train of the containment air cooling and filtering systems (VA-3A and VA-7C), or (VA-3B and VA-7D), must be operable during power operations). If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

b. During power operation one component cooling heat exchanger may be inoperable. If the operability of the heat exchanger is not restored within 14 days, the reactor shall be placed in a hot shutdown condition within 12 hours. If two component cooling heat exchangers are inoperable, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable heat exchanger(s) is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION

2.4 Containment Cooling (Continued)

The component cooling system pumps and heat exchanger, the spray pumps and the shutdown heat exchangers are located in the auxiliary building.[9] [10] The raw water (RW) pumps are located in the intake structure.[11] When river level is low, the RW pump minimum submergence level (MSL) of 976 Feet 9 inches in the intake cells can be affected by the accumulation of debris and/or ice on the traveling screens and/or trash racks possibly leading to RW pump degradation. Thus, when river level is low, the intake cells are monitored to ensure that appropriate actions are taken in the event that adequate water levels cannot be maintained in the intake cells.

Intake cell levels are also adversely affected by the flows associated with the non-safety related circulating water (CW) pumps since the large flow rates associated with the CW pumps create significant head losses even with relatively clean intake cell conditions. However, the CW pumps have a much higher MSL requirement (983 feet 0 inches) and would become unstable and trip or be manually shutdown well before intake cell levels decrease to the RW pump MSL. The head loss associated with CW pump flow would then be recovered and intake cell levels would rise.

Analyses show that after a high heat load accident such as a large break LOCA or Main Steam Line Break inside containment, three in service component cooling heat exchangers will maintain CCW return temperature in an analyzed range. This assumes all of the containment air cooling units are operating which would create the maximum heat load on the CCW system. In order to ensure that three heat exchangers would be in service after a DBA in conjunction with an assumed single failure, four are required to be operable.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 2	Group/Category #		1
	K/A #	038 EA2.08	
Level of Difficulty: 4	Importance Rating		4.4

Steam Generator Tube Rupture: Ability to determine or interpret the following as they apply to a SGTR: Viable alternatives for placing plant in safe condition when condenser is not available

Question: 78

Given the following conditions:

- Plant was in MODE 2 when a Steam Generator Tube Rupture occurred on RC-2B.
- EOP-04, Steam Generator Tube Rupture, is in progress.
- Secondary Side Distributed Control System (DCS) is NOT responding to operator input.
- An inadvertent Steam Generator Isolation Signal has occurred.
- Reactor Coolant System (RCS) T_{HOT} is 540°F.
- RCS pressure is 1100 psia.
- Steam Generator RC-2A pressure is 985 psia.
- Steam Generator RC-2B has been isolated in accordance with EOP-04.

Which of the following procedures should be entered and what action is required?

A. HR-12, Secondary Heat Removal Operation, and operate the Steam Dump Bypass Valves.

B. HR-12, Secondary Heat Removal Operation, open HCV-1041C, RC-2A MSIV Bypass Valve and locally operate HCV-1040, Atmospheric Dump Valve.

C. HR-13, Local MS-291, MS-292 Operation, and simultaneously open BOTH MS-291 and MS-292, Air Assisted Main Steam Safety Valves.

D. HR-13, Local MS-291, MS-292 Operation, and alternate between MS-291 AND MS-292, Air Assisted Main Steam Safety Valves.

Answer: B

K/A Match:

As the SRO, applicant must select procedure/attachment for accommodating a Steam Generator Tube Rupture when the Condenser is not available.

Explanation:

A. Incorrect. Plausible because HR-12 is the correct procedure to enter. Incorrect because Secondary DCS is not functioning and there is no option for locally controlling Steam Dump Bypass Valves.

B. **Correct**. The Steam Generator Isolation Signal (SGIS) isolates steam and feed to both SGs. Valve control can be obtained by opening the power supply to these valves and then manually operating. With Steam Generator RC-2B isolated and Secondary DCS unavailable, opening RC-2A MSIV Bypass Valve and locally operating HCV-1040 is the correct method.

C. Incorrect. Plausible because HR-13 does provide guidance for local operation of MS-291 and MS-292. Plausible because this will provide the fastest cooldown, this is desirable with RCS temperature at 540°F. Incorrect because MS-292 should not be opened.

D. Incorrect. Plausible because HR-13 is the procedure for local operation of MS-291 and MS-292. With Secondary DCS unavailable local operation is one of the options. Plausible because it desired to have a controlled cooldown and alternating operation between the main steam safety valves will provide a more controlled cooldown. Incorrect because Steam Generator RC-2B is already isolated and MS-292 should not be opened.

Technical Reference: EOP/AOP Attachments-HR, Attachment HR-12, Steps 4, 9, 10, & 11, Rev. 1

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-14, Steam Generator Tube Rupture-Licensed Operator
EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of a SGTR.
EO 3.0 - **DEMONSTRATE** the knowledge required to implement the Floating Steps of EOP-04, SGTR, when needed.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ____
55.43 5

HR-12, Steps 4, 9, 10, & 11

4. **IF** Steam Dump and Bypass is available, **THEN** control RCS temperature with a single valve, by performing the following (SD&B Control Display, DCS):	4.1 **IF** Steam Dump and Bypass is **NOT** available, **THEN** GO TO Step 9.
9. **IF** HCV-1040 is available, **THEN** control RCS temperature by performing the following:	9.1 **IF** both MSIVs are closed, **THEN** GO TO Step 10.
10. **IF** the MSIVs are closed **AND** HCV-1040 is required for heat removal, **THEN** unisolate the least affected S/G(s) by performing the following: Open either or **ALL** of the following: • HCV-1041C, RC-2A MSIV Bypass Valve • HCV-1042C, RC-2B MSIV Bypass Valve (continue)	10.1 **IF BOTH** MSIVs were closed for Condenser isolation, **THEN** GO TO Step 11. a.1 **IF** local operation is required, **THEN** perform the following for S/G(s) being unisolated: 1) **IF** RC-2A is being unisolated, **THEN** open MCC-4A1-C04, "HCV-1041C MAIN STEAM BYPASS VALVE" (Upper Electrical Penetration Room). 2) **IF** RC-2A is being unisolated, **THEN** open HCV-1041C, "STEAM GENERATOR RC-2A MS ISOLATION VALVE HCV-1041A BYPASS VALVE" (Room 81).
11. Operate at least one of the following Air Assisted Main Steam Safety Valves: • MS-291 • MS-292	11.1 **(LOCAL)** Operate at least one of the following Air Assisted Main Steam Safety Valves PER Attachment HR-13, Local MS-291, MS-292 Operation: • MS-291 • MS-292

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 2	Group/Category #		1
	K/A #	055 G 2.4.35	
Level of Difficulty: 3	Importance Rating		4.0

Station Blackout: Emergency Procedures/Plan: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

Question: 79

Given the following conditions:

- EOP-07, Station Blackout, was entered following a Loss of Offsite Power.
- Diesel Generator DG-1 was previously cleared for maintenance.
- Diesel Generator DG-2 failed to start after the following was performed per MVA-18, Emergency Start of Diesel Generator DG-2:
    - The NORMAL START pushbutton was pressed in the Control Room.
    - The EMERGENCY START pushbutton was pressed in the Control Room.
- An Auxiliary Operator is being dispatched to the DG-2 Room with MVA-18.

(1) What initial local action is directed to the Auxiliary Operator? and
(2) Once DG-2 is running, what procedure provides the direction to energize Bus 1A4?

A. (1) Take local control of DG-2 and attempt a NORMAL ENGINE START (D2-63) followed by an EMERGENCY ENGINE START (D2-74).
(2) EOP-07, Station Blackout.

B. (1) Take local control of DG-2 and attempt a NORMAL ENGINE START (D2-63) followed by an EMERGENCY ENGINE START (D2-74).
(2) MVA-18, Emergency Start of Diesel Generator DG-2.

C. (1) Verify DG starting air pressure > 150 psig then pull up on the “T” handle for DIESEL GENERATOR DG-2 PRIMARY STARTING AIR SOLENOID VALVE.
(2) MVA-18, Emergency Start of Diesel Generator DG-2.

D. (1) Verify DG starting air pressure > 150 psig then pull up on the “T” handle for DIESEL GENERATOR DG-2 PRIMARY STARTING AIR SOLENOID VALVE.
(2) EOP-07, Station Blackout.

Answer: A

K/A Match:

As the SRO, applicant must be familiar with local actions used during EOPs including appropriate procedure references following those actions. The question is SRO ONLY level because it requires specific procedure content knowledge of the EOP with respect to restoration of power to buses once the DG is locally started.

Explanation:

A. **Correct**. The CONTINGENCY ACTION at Step 16 uses MVA-18 to attempt a normal then emergency start of DG-2 from the Control Room. MVA-18 then sends the AO to the DG-2 Room to attempt a local DG start. MVA-18 delineates steps that the local operator takes in an attempt to start DG-2. Those 1st steps are NORMAL ENGINE START then EMERGENCY ENGINE START using switches on the Engine Control Panel. If the diesel fails to start via this method, and starting air pressure is > 150 psig then the Primary Air Solenoid "T" handle is pulled followed by the Secondary Air Solenoid "T" handle in an attempt to start the Diesel. Once the diesel is started MVA-18 directs the operator to MVA-19, Emergency Diesel Generator Long Term Actions. MVA-19 monitors DG operation and repositions some DG controls for long-term operation. The crew must return to EOP-07 to close DG-2 Output Breaker 1AD2.

B. Incorrect. Plausible because this is the initial local starting method. Incorrect because MVA-18 won't shut DG-2 Output Breaker 1AD2.

C. Incorrect. Plausible if thought that MVA-18 closed the DG-2 Output Breaker. Starting air pressure must be greater than 150 psig to start the DG but this method is the last alignment attempted.

D. Incorrect. Plausible because EOP-07 entry is correct. Incorrect because using the air solenoid T handle to start the DG is the last alignment attempted.

Technical Reference: (Attach if not previously provided including revision number)
EOP-07, Steps 16 and 17, Rev. 18
EOP/AOP Attachments-MVA, Attachment MVA-18, Steps 1 to 5 , Rev. 1
EOP/AOP Attachments-MVA, Attachment MVA-18, Steps 13 to 17, Rev. 1

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:
Lesson Plan 7-18-17, Station Blackout-Licensed Operator
EO 1.1 - **EXPLAIN** the major strategy used to mitigate the consequences of a SBO.
EO 2.4 - **GIVEN** a copy of Attachment MVA-17 or 18, **EXPLAIN** the steps necessary to emergency start a Diesel Generator.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

EOP-07, Steps 16 & 17

16. Verify DG-2 is running.

16.1 **IF** DG-2 fails to start, **THEN** start DG-2 PER Attachment MVA-18, Emergency Start of Diesel Generator DG-2.

17. Synchronize and close breaker 1AD2 by performing the following:

    a. Verify DG-2 frequency is greater than 60 Hz **AND** voltage is greater than 4160 V.

    b. Place "D2/BUS 1A4 SYNC SWITCH" to "ON".

    c. Close Breaker 1AD2.

    c.1 **IF** breaker 1AD2 does **NOT** close **AND** no fault exists on bus 1A4, **THEN** perform the following:

MVA-18, Steps 1-5 and 13-17

1. Start DG-2 by pressing the "DIESEL NORMAL START" push button.

1.1 **IF** DG-2 did **NOT** start, **THEN** press the "EMERGENCY START" push button.

2. IF DG-2 is running, THEN GO TO Step 16.

3. Start DG-2 by performing the following (Engine Control Panel):

a. Place the "ENGINE CONTROL D2-62" switch in "LOCAL".

b. Press the "NORMAL ENGINE START D2-63" push button.

3.1 **IF** DG-2 did **NOT** start, **THEN** start DG-2 by performing the following (Engine Control Panel):

a. Place the "ENGINE CONTROL D2-62" switch in "EMERG".

b. Press the "EMERGENCY ENGINE START D2-74" push button.

4. **IF** DG-2 is running, **THEN** GO TO Step 15.

NOTE

A minimum starting air pressure of 150 psig is required to start DG-2.

5. **IF** "PRIMARY STARTING AIR SYSTEM PRESSURE" **OR** "SECONDARY STARTING AIR SYSTEM PRESSURE" is greater than 150 psig (Engine Control Panel), **THEN** GO TO Step 13.

13. WHEN "PRIMARY STARTING AIR SYSTEM PRESSURE" OR "SECONDARY STARTING AIR SYSTEM PRESSURE" is greater than 150 psig (Engine Control Panel), THEN ensure the overspeed trip is reset (West End of DG-2).

14. Start DG-2 by performing the following (Engine Control Panel):

a. Place the "ENGINE CONTROL D2-62" switch in "LOCAL".

b. Press the "NORMAL ENGINE START D2-63" push button.

(continue)

****

CAUTION

This step may damage the Air Start Motors if the manual override is not released after a few seconds.

****

14.1 **IF** DG-2 did **NOT** start,
AND the Primary Air Receivers are pressurized,
THEN manually override SA-192, "DIESEL GENERATOR DG-2 PRIMARY STARTING AIR SOLENOID VALVE", by pulling up "T" handle located on top of the valve.

14. (continued)

14.2 **IF** DG-2 did **NOT** start,
AND the Secondary Air Receivers are pressurized,
THEN manually override SA-191, "DIESEL GENERATOR DG-2 SECONDARY STARTING AIR SOLENOID VALVE", by pulling up "T" handle located on top of the valve.

15. Ensure the "ENGINE CONTROL D2-62" switch in "EMERG" to return DG-2 start function to the Control Room.

16. Check that DG-2 is operating at greater than or equal to 900 RPM.

17. IMPLEMENT Attachment MVA-19, Emergency Diesel Generator Long Term Actions.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 2	Group/Category #		1
	K/A #	057 AA2.16	
Level of Difficulty: 3	Importance Rating		3.1

Loss of Vital AC Instrument Bus: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: Normal and abnormal PZR level for various modes of plant operation.

Question: 80

Given the following conditions:

- A Plant Cooldown is in progress.
- Shutdown Cooling is in service.
- Reactor Coolant System (RCS) T_{COLD} is 280°F.
- Pressurizer pressure is 250 psia.
- Instrument Bus 2 voltage is 107 VAC.
- LI-106, Cold Shutdown Pressurizer Level, failed low.
- Automatic pump STARTS/STOPS have been disabled.

Which of the following procedures is implemented, and what instrument is used to determine actual Pressurizer level in conjunction with the TDB Correction Curve?

A. AOP-16, Loss of Instrument Bus Power OR AOP-19, Loss of Shutdown Cooling. Use Pressurizer Level Control Channel LI-101Y.

B. AOP-16, Loss of Instrument Bus Power AND AOP-19, Loss of Shutdown Cooling. Use Pressurizer Level Control Channel LI-101Y.

C. AOP-16, Loss of Instrument Bus Power OR AOP-19, Loss of Shutdown Cooling. Use Cold Shutdown Reactor Coolant Level Indicator LI-197.

D. AOP-16, Loss of Instrument Bus Power AND AOP-19, Loss of Shutdown Cooling. Use Cold Shutdown Reactor Coolant Level Indicator LI-197.

Answer: B

K/A Match:

As the SRO, applicant must be knowledgeable of procedure requirements and substitute instrumentation when an instrument failure occurs. SRO ONLY because of the off-normal conditions in the stem requires determination of compensatory instrumentation to use for indication of pressurizer level.

Explanation:

A. Incorrect. Plausible because LI-101Y is the instrument to use. Incorrect because a Loss of Shutdown Cooling has occurred, and loss of an instrument bus. When the instrument bus is lost on shutdown cooling, loss of shutdown cooling occurs because HCV-341 fails closed, and FCV-326 fails open.

B. **Correct**. LI-106 is powered from Instrument Bus IA-42B. Guidance for loss of this bus is found in AOP-16, Loss of Instrument Bus, Section VII, Loss of Instrument Bus AI-42B. TDB.III-20 lists elevations for various level instruments. LI-101Y must be used if LI-106 is inoperable. Loss of Shutdown Cooling, AOP-19 is required to be entered due to the loss of cooling capability.

C. Incorrect. Plausible if thought that a Loss of Shutdown Cooling is in progress. LI-197 has insufficient overlap with LI-106; therefore, LI-101Y would be used to determine actual Pressurizer level. Incorrect because both AOP-19 and AOP-16 are required to be entered.

D. Incorrect. Plausible because the correct procedures are selected. Incorrect because LI-197 only has about ~5% overlap with LI-106.

Technical Reference:	TDB-V.11, IA-42B Instrument Bus Loads, Rev. 51
(Attach if not previously provided including revision number)	TDB-III-20, RCS Elevations vs. LI-106, LI-197 and LIS-119, Rev. 19
	AOP-16, Section VII, Step 8 NOTE, Rev. 20
	AOP-19, Step 7, Rev. 18

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-32, Loss of 4160V Bus or 480V Bus Power-Licensed Operator EO 1.2 - **DESCRIBE** how the plant responds to a loss of a 4160 Volt or 480 Volt bus in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

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

Question History: Last NRC Exam 2007 NRC Exam

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

TDB-V.11, IA-42B Instrument Bus Loads

System	RCS	
LIA-132	Quench Tank Level	Fails Low
PIA-131	Quench Tank Press.	Fails Low
TIA-136	Press. Relief Line Temp.	Fails Low

System	RCS-LEVEL	
LT-197	RCS Level	Annunc. & Zero Indication
LT-106	Pressurizer Level (Wide Range) Xmtr	Fails Low
LI-106	Pressurizer Level (Wide Range)	Blank Display

System	RCS-PRESSURE	
PC-105A	Przr. Press.	-
PT-105	Wide Range RCS Pressure	No Signal to PC-105A

NOTE

Upon loss of Instrument Bus AI-42B, **ALL** of the following instrumentation or equipment associated with the **Core Heat Removal Safety Function** is inoperable:

- AI-270 panel, preventing auto start of oil pumps for RC-3A/B/C/D
- RC-3B and RC-3D, Seal Leak Off Flow Switches
- "LPSI/SHTDN CLG FLOW CONTROLLER FCV-326"
- "COLD SHTDN RC LEVEL LI-197"
- "SHUTDOWN COOLING INLET/OUTLET TEMP TR-346" recorder
- PC-118A, SDC Low Range PZR Pressure Channel
- SDC Flow Control Valve FCV-326 fails open
- SDC HX Temperature Control Valve, HCV-341 fails closed

AOP-16, Section VII, Step 8 NOTE

Bank Question:

The RCS was being cooled down with RCS cold leg temperature at 350°F and pressurizer pressure at 1400 psia when a loss of an instrument bus made pressurizer level indicator, LI-106, inoperable. All automatic pump starts/stops are disabled.

How should actual pressurizer level be determined in this situation.

A. Enter the ARP for "LI-106 LEVEL LO". Direct the Operator to use LI-101Y and the TDB correction curve to determine actual pressurizer level.

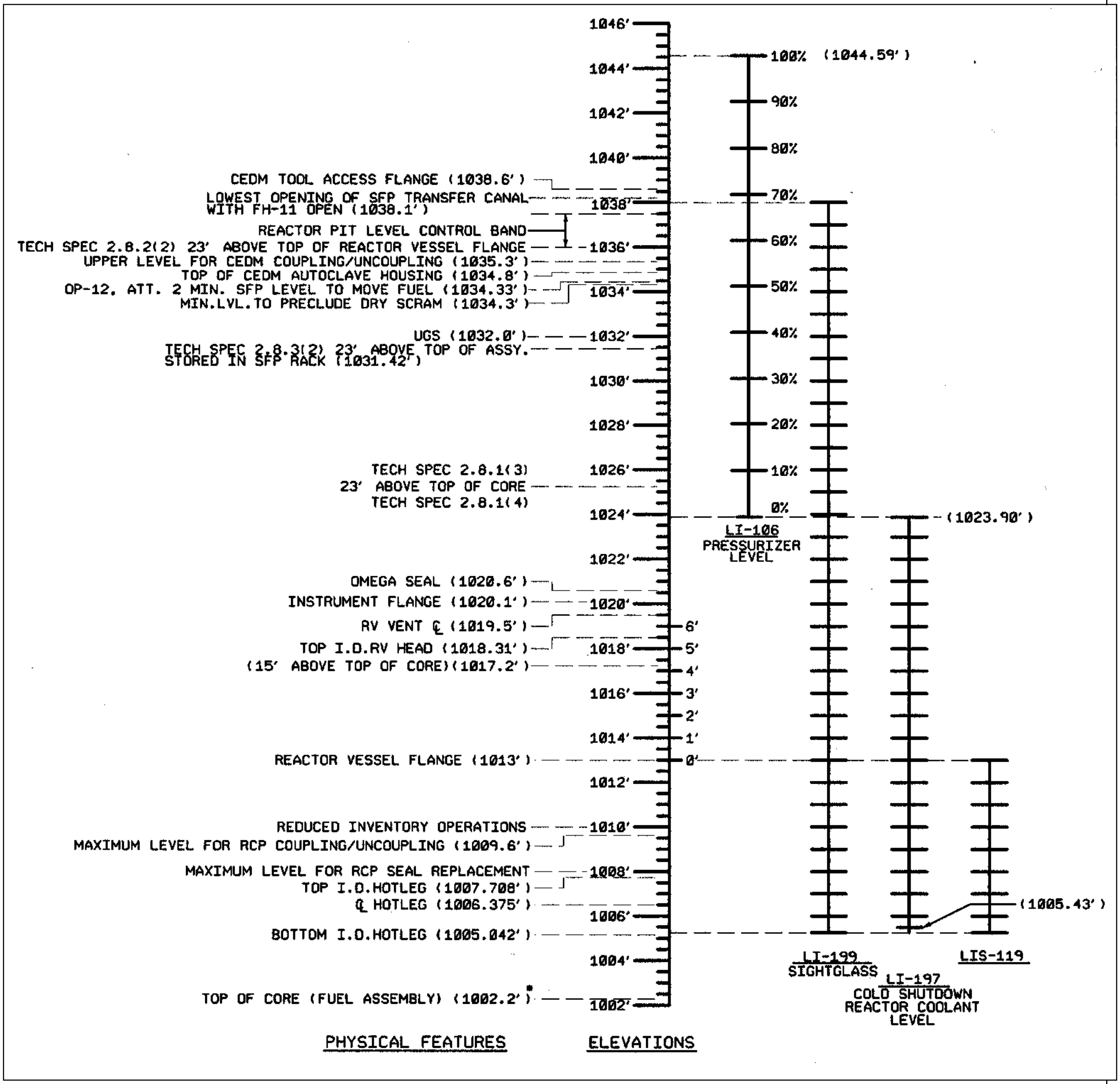
B. Enter the ARP for "LI-106 LEVEL LO". Direct the Operator to use LI-197 and the TDB correction curve to determine actual pressurizer level.

C. Enter AOP-16, "Loss of Instrument Bus Power". Direct the Operator to use LI-101Y and the TDB correction curve to determine actual pressurizer level.

D. Enter AOP-16, "Loss of Instrument Bus Power". Direct the Operator to use LI-197 and the TDB correction curve to determine actual pressurizer level.

TDB.III-20

RCS Elevations vs. LI-106, LI-197 and LIS-119



AOP-19, Step 7

7. Verify RCS Water Level is above the centerline of the Hot Leg using at least two of the following level indications:

- RVLMS (29%)
- LI-197 (1006.5 feet)
- LI-199 (1006.5 feet, Containment)
- LIS-119 (1006.5 feet)
- LI-106
- LI-101X
- LI-101Y

7.1 **IF** RCS Water Level is **NOT** above the centerline of the Hot Leg,
THEN stop the operating LPSI or CS Pump.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/28/15	Tier #		1
Change: 1	Group/Category #		1
	K/A #	077 G 2.2.36	
Level of Difficulty: 4	Importance Rating		4.2

Generator Voltage and Electric Grid Disturbances: Equipment Control: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Question: 81

Given the following conditions:

- Plant is at 100% power.
- Diesel Generator DG-1 starting air pressure has just been discovered at 145 psig.
- OPPD Transmission and Distribution personnel working in the station switchyard are preparing to perform maintenance on the 161 KV line.
- During the maintenance, it is forecasted that 161 KV system voltage will be 161.0 KV.

(1) What is required for DG-1?
(2) What is required concerning the planned 161 KV maintenance?

A. (1) Declare DG-1 inoperable.
(2) Do NOT allow the maintenance to be conducted on the 161KV line until DG-1 is restored.

B. (1) Ensure DG-1 starting air remains above 125 psig
(2) Allow the maintenance to be conducted on the 161 KV line as long as the forecasted time is less than 8 hours.

C. (1) Ensure DG-1 starting air remains above 125 psig.
(2) Do NOT allow the maintenance to be conducted on the 161KV line until DG-1 is restored.

D. (1) Declare DG-1 inoperable.
(2) Allow the maintenance to be conducted on the 161 KV line as long as the forecasted time is less than 8 hours.

Answer: A

K/A Match:

As the SRO, applicant must identify the electrical system operability and maintenance requirements for the diesel generator and 161 KV Grid. The question is SRO ONLY level because it tests on specific procedure content knowledge (beyond that required of an RO) of an AOP-31 requirement and diesel operability.

Explanation:

A. **Correct**. Per AOP-31, 161 KV Grid Malfunctions, Section I, 161KV Grid Instabilities, with one Diesel Generator inoperable and actual or predicted 161KV voltage less than 161.3 KV, a reactor shutdown would be required. Per TS 2.7, diesel generator #1 is inoperable less than 190 psig.

B. Incorrect. Plausible because per Technical Specification LCO 2.7, diesel generator operability can be extended as long as starting air pressure is maintained greater than 150 psig and restored greater than 190 psig within 48 hours. Plausible because the allowable maintenance time for 161KV system work restores the system to operable within a normal shift. Incorrect because the starting air system pressure is 145 psig, which is less than both the limit and the modification for operability. Incorrect because the anticipated voltage would make the system inoperable and reportable to the NRC.

C. Incorrect. Plausible because per Technical Specification LCO 2.7, diesel generator operability can be extended as long as starting air pressure is maintained greater than 150 psig and restored greater than 190 psig within 48 hours. Incorrect because the starting air system pressure is 145 psig, this is less than both the limit and the modification for operability. The correct action for 161KV maintenance is addressed, but the incorrect diesel generator operability is determined.

D. Incorrect. Plausible because the diesel generator is declared inoperable less than 190 psig. Incorrect because with a diesel inoperable and 161KV system voltage less than 161.3 KV, a plant shutdown would be required.

Technical Reference: AOP-31, Steps 1, 2, & 9, Rev. 14

(Attach if not previously provided including revision number) Technical Specification LCO 2.7, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-31, 161 KV Grid Malfunctions-Licensed Operator
EO 1.8 - **DESCRIBE** what 161KV grid conditions require plant shutdown.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

AOP-31, Steps 1 & 2

NOTE

The 161 KV Grid is inoperable below 161.3 KV.

1. **IF** actual or predicted 161 KV Grid voltage is less than 161.3 KV, **THEN** perform the following:

   A. **IF** either Diesel Generator is inoperable, **THEN** place the Reactor in Hot Shutdown within 6 hours PER OP-4, Load Change and Normal Power Operation.

   B. Notify the NRC Operations Center within 4 hours of the 161 KV Grid inoperability.

2. Verify voltages on 4160 V Buses 1A3 and 1A4 are greater than 3750 V.

   2.1 **IF** voltage on 4160 V Buses 1A3 or 1A4 is less than 3750 V, **THEN** GO TO Step 9.

AOP-31, Step 9

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.7 Electrical Systems (Continued)

(2) Modification of Minimum Requirements

The minimum requirements may be modified to the extent that one of the following conditions will be allowed after the reactor coolant has been heated above 300°F. However, the reactor shall not be made critical unless all minimum requirements are met. If any of the provisions of these exceptions are violated, the reactor shall be placed in a hot shutdown condition within the following 12 hours. If the violation is not corrected within an additional 12 hours, the reactor shall be placed in a cold shutdown condition within an additional 24 hours.

a. Both unit auxiliary power transformers T1A-1 and T1A-2 (4.16 kV) may be inoperable for up to 72 hours.

b. Either house service transformer T1A-3 or T1A-4 (4.16kV) may be inoperable for up to 7 days. The NRC Operations Center shall be notified by telephone within 4 hours after transformer inoperability. Additionally, within 24 hours from discovery of either house service transformer inoperability, declare the required feature(s) associated with the inoperable house service transformer inoperable, when its redundant required feature (including the steam driven auxiliary feedwater pump FW-10) is inoperable.

c. Both house service transformers T1A-3 and T1A-4 (4.16kV) may be inoperable for up to 72 hours. The loss of the 161kV incoming line renders both transformers inoperable. The NRC Operations Center shall be notified by telephone within 4 hours after inoperability of both transformers.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.7 Electrical Systems (Continued)

o. One of the required inverters may be inoperable for up to 24 hours provided the reactor protective and engineered safeguards systems instrument channels supplied by the remaining three required inverters are all operable and the 120V a-c instrument bus associated with the inoperable inverter is powered from its bypass source.

(3) Modification of Minimum Requirements for Diesel Fuel Oil, Diesel Lube Oil, and Starting Air

The minimum requirements may be modified to the extent that any of the following conditions will be allowed after the reactor coolant has been heated above 300°F. However, the reactor shall not be made critical unless all minimum requirements are met.

a. If the inventory of diesel fuel oil in FO-1 is less than 16,000 gallons and/or FO-10 is less than 10,000 gallons, but the combined inventory in FO-1 and FO-10 is greater than a 6 day supply (23,350 gallons), then restore the required inventory within 48 hours.

b. If one or more diesel generators has lube oil inventory < 500 gallons and > 450 gallons, then restore the lube oil inventory to within limits within 48 hours.

c. If the total particulates of fuel oil stored in FO-1 or FO-10 is not within limits, then restore fuel oil total particulates to within limits within 7 days.

d. If the properties of new fuel oil stored in FO-1 or FO-10 is not within limits, then restore stored fuel oil properties to within limits within 30 days.

e. If one or more diesel generators has the required starting air receiver bank with pressure < 190 psig and > 150 psig, then restore starting air receiver bank pressure to > 190 psig within 48 hours.

f. If the Required Action and associated Completion Time of a, b, c, d or e are not met or one or more diesel generators have diesel fuel oil, lube oil, or a required starting air subsystem not within limits for reasons other than a, b, c, d, or e, then declare the associated DG inoperable immediately.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 1	Group/Category #		2
	K/A #	003 AA2.01	
Level of Difficulty: 4	Importance Rating		3.9

Dropped Control Rod: Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position.

Question: 82

Given the following conditions:

- A Shutdown Group CEA B15 has dropped in the core.
- AOP-02, CEA Malfunctions, has been implemented.
- During recovery of CEA B15, BOTH CEA Position Indication Systems were lost.
- Reactor power is stable at 65%.

Which of the following is required?

A. Maintain Reactor power at 65% and make NO CEA adjustments until either the Primary or Secondary Position Indicating System is restored.

B. Maintain Reactor power at 65% and make NO CEA adjustments until BOTH the Primary and Secondary Position Indicating Systems are restored.

C. Declare all CEAs inoperable, verify Shutdown Margin is satisfied, and place the Reactor in HOT SHUTDOWN per AOP-5, Emergency Shutdown.

D. Enter Technical Specification LCO 2.15.1(4), Instrumentation and Control Systems and place the Reactor in HOT SHUTDOWN per OP-4, Load Change and Normal Power Operations.

Answer: D

K/A Match:

As the SRO, applicant must have knowledge of the Technical Specification requirements for CEA Position Indication Systems.

Explanation:

A. Incorrect. Plausible because no CEA adjustments should be made when both position indicating systems are out of service. Incorrect because a reactor shutdown is required.

B. Incorrect. Plausible because no CEA adjustments should be made with both position indicating systems are out of service. Incorrect because a reactor shutdown is required.

C. Incorrect. Plausible because the procedure entry is correct for a misaligned Shutdown Group CEA that cannot be restored to within 12 inches of all CEAs in its Group per AOP-02, Section II. Incorrect because the rod has been recovered and an Emergency Shutdown is not required, and all rods are not inoperable because of the loss of position indication.

D. **Correct**. Loss of both Position Indication Systems does not require a Reactor Trip; however, the plant must be placed in HOT SHUTDOWN within 12 hours as outlined in Technical Specification LCO 2.15.1(4).

Technical Reference: AOP-02, Section II, Step 16 CA & Section V, Step 1 CA, Rev. 10a

(Attach if not previously provided including revision number) Technical Specification LCO 2.15.1(4), Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-2, CEDM Malfunctions-Licensed Operator
EO 1.6 - **DESCRIBE** the Technical Specification LCO challenged by a CEA or Control System malfunction.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 2

AOP-02, Section V, Step 1

1. Verify **ONE** of the following is operable:
   - Primary CEA position indication
   - Secondary CEA position indication

1.1 **IF** both the Primary and Secondary CEA position indication systems are inoperable, **THEN** perform the following:

a. Maintain all CEAs fully withdrawn using the core mimic display on the DCS.

b. Enter Technical Specification 2.15.1(4).

c. Place the Reactor in Hot Shutdown within Twelve hours PER OP-4, Load Change and Normal Power Operation.

16. **WHEN** power level is less than or equal to 70% ΔT Power,

THEN realign the CEA, within one hour to within 12 inches of all CEAs in its group by performing the following:

a. Borate as necessary to maintain power level steady.

b. Place the "ROD CONTROL MODE SELECTOR SWITCH" in "MANUAL INDIVIDUAL".

c. Select the group containing the misaligned CEA using the "CONTROL ROD GROUP SELECTOR SWITCH".

d. Select the misaligned CEA which is to be moved using the "ROD SELECTOR SWITCH" for the misaligned group.

16.1 **IF** the misaligned CEA can **NOT** be restored to within 12 inches of all CEAs in their group within one hour,

THEN initiate a Reactor shutdown by performing the following:

a. Declare the CEA inoperable.

b. Verify Shutdown margin is satisfied within one hour PER Technical Specification 2.10.2, Reactivity Control Systems and Core Physics Parameters Limits.

c. Place the Reactor in Hot Shutdown within an additional five hours PER OP-4, Load Change and Normal Power Operations.

d. GO TO Section 5.0, Exit Conditions.

AOP-02, Section II, Step 16

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.15.1 Instrumentation and Control Systems (Continued)

If after 24 hours from time of initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applied to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

TECHNICAL SPECIFICATIONS

TABLE 2-5

Instrumentation Operating Requirements for Other Safety Feature Functions

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition
1	CEA Position Indication Systems	1[(a)]	None	None
2	Pressurizer Level	1	None	Not Applicable

NOTES:

(a) If one channel of CEA position indication is inoperable for one or more CEAs, requirements of specification 2.15.1 are modified for item 1 to "Perform TS 3.1, Table 3-3, Item 4 within 15 minutes following any CEA motion in that group." Specifications 2.15.1(1), (2), and (3) are not applicable.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		1
Change: 1	Group/Category #		2
	K/A #	036 AA2.02	
Level of Difficulty: 3	Importance Rating		4.1

Fuel Handling Incidents: Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling incident.

Question: 83

Given the following conditions:

- Refueling Operations are underway in the Spent Fuel Pool.
- An irradiated Fuel Assembly is being moved from the Upender to its storage location.
- During transition, bubbles are seen rising from the Fuel Assembly.
- AOP-08, Fuel Handling Incident, has been entered.

As the Fuel Handling Supervisor, where will you direct this Fuel Assembly be stored?

Stored in…

A. …a new fuel storage rack.

B. …the Region 2 storage location designated in the core offload sequence.

C. …the horizontal position in the Upender.

D. …the lowered position of the New Fuel Elevator.

Answer: C

K/A Match:
As the SRO, applicant must be familiar with appropriate locations for storing a damaged Fuel Assembly. This question requires specific procedure content knowledge beyond that required of an RO for addressing a damaged fuel assembly placement.

Explanation:

A. Incorrect. Plausible because this is a valid storage location for new fuel. Incorrect because this is an irradiated fuel assembly.
B. Incorrect. Plausible because the Spent Fuel Pool does have designated storage areas for immediately storing damaged fuel. Incorrect because the fuel should only be stored in Region 1.
C. **Correct**. As outlined in the precautions of OI-FH-1, Fuel Handling Operations, any incident that requires the immediate uncoupling of an irradiated Fuel Assembly can be stored in Region 1, any empty cell in the Spent Fuel Pool, or stored in the horizontal position in the Upender.
D. Incorrect. Plausible because stored in the lowered position of the New Fuel Elevator is allowed but only for new Fuel.

Technical Reference: OI-FH-1, Precaution 21, Rev. 92
(Attach if not previously provided including revision number) AOP-08, Entry Conditions, Rev. 10a

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-8, Fuel Handling Incident-Licensed Operator
EO 1.3 - **DESCRIBE** the major recovery actions of this AOP.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	
	55.43	7

PRECAUTIONS (continued)

21. The following positions are provided for guidance in the event of an incident which requires the immediate uncoupling of a fuel assembly from FH-12 [AR 17967]:

FH-12

- Any empty cell in Region 1
- Any empty cell in the Spent Fuel Pool
- Stored in the horizontal position in the Upender

New Fuel

- New Fuel Storage Rack
- Stored in the lowered position of the New Fuel Elevator
- Any empty cell in Region 1

AOP-08, Entry Condition I

2.0 ENTRY CONDITIONS

A fuel assembly has been damaged which may be indicated by any of the following:

A. Area radiation monitors increase.

B. "RM-050 CNTMT PARTICULATE HIGH RADIATION" alarm (AI-33C; A33C).

C. "RM-051 CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

D. "RM-052 STACK/CNTMT NOBLE GAS HIGH RADIATION" alarm (AI-33C; A33C).

E. "RM-062 AUX BLDG VENT STACK HIGH RADIATION" alarm (AI-33C; A33C).

F. Containment air particulate high radiation indication upscale.

G. Ventilation Isolation Actuation Signal (VIAS).

H. While handling fuel, the Hoist Load Indicator shows low Hoist weight.

I. Possible damage to Fuel Assembly is observed.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/01/15	Tier #		1
Change: 0	Group/Category #		2
	K/A #	060 G 2.4.21	
Level of Difficulty: 3	Importance Rating		4.6

Accidental Gaseous Radwaste Release: Emergency Procedures/Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Question: 84

Given the following plant parameters before and after an event:

Parameter	**Value before trip**	**Value when transitioning from EOP-00**	**Value 15 minutes after transitioning from EOP-00**
Reactor Power	100%	8x10-6 %	300 cps
Pressurizer Level	60%	10%	0%
RVLMS	100%	100%	100%
RCS T_{COLD}	542.7°F	450°F	420°F
Containment Area Radiation	6 mrem/hr	500 mrem/hr	15 Rem/hr
Containment Sump level	20 inches	4 ft	10 ft
Containment Pressure	0.1 psig	50 psig	0.3 psig
Steam Generator RC-2A Pressure	830 psia	800 psia	650 psia
Steam Generator RC-2B Pressure	830 psia	450 psia	30 psia
4160 buses energized	All	1A3	1A3

All automatic Engineered Safety Features (ESF) equipment operated as designed.

Which of the following presents the greatest challenge?

A. Core uncovery resulting in fuel melt.

B. Loss of core heat removal capability when the SIRWT empties.

C. Radioactive release from Containment.

D. Overloading Diesel Generator(s) results in Station Blackout.

Answer: C

K/A Match:

As the SRO, the applicant must be familiar with Emergency Plan entry conditions and upgrade criteria during loss of fission product barriers. Knowledge of Emergency Plan Fission Product Barrier Criteria and assessment of Safety Functions is specific SRO knowledge.

Explanation:

A. Incorrect. Plausible because Pressurizer level is lowering, which could be interpreted as a challenge to the core remaining covered. Incorrect because the Steam Generator (SG) has nearly completed blowing down, which stops the RCS shrink due to the uncontrolled heat extraction. Core cooling is maintained by one Train of Safety Injection (SI) and heat removal is available with one intact SG.

B. Incorrect. Plausible because Containment Sump level is rising and one train of SI and Containment Spray are transferring SIRWT inventory to the Containment. Also plausible because one SG and one Train of SI are not available for heat removal. Incorrect because Containment Sump level is available to be used for heat removal with one train of post-RAS equipment available, as well as one SG available for heat removal.

C. **Correct**. Containment pressure indications reveal a sudden pressure reduction in Containment following an initial rise associated with a Loss of Coolant Accident. The indications are representative of a Loss of Containment in accordance with TBD-EPIP-OSC-1F, concurrent with a Loss of the Reactor Coolant System Barrier. This presents a release of fission products downwind of the site.

D. Incorrect. Plausible because one loss of one Diesel Generator would result in a Station Blackout. Incorrect because the equipment needed to combat the events in progress is designed to be met by one vital bus energized. With vital 4160 volt Bus 1A3 energized, safety injection, heat removal, feedwater to the SGs, and instrumentation needed to monitor plant conditions is provided.

Technical Reference: TDB-EPIP-OSC-1F, Fission Product Barrier Degradation, Rev. 1

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 1070-103, Classification Bases - Licensed Operator
EO 1.8 - **EXPLAIN** how the Three Fission Product Barrier Criteria is used in the classification process.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ______
55.43 5

Basis Information
PWR Emergency Action Levels

CONTAINMENT BARRIER EXAMPLE EALs: (1 **OR** 2 **OR** 3 **OR** 4 **OR** 5 **OR** 6)

The Containment Barrier includes the containment building, its connections up to and including the outermost containment isolation valves. This barrier also includes the main steam, feedwater, and blowdown line extensions outside the containment building up to and including the outermost secondary side isolation valve.

1. Containment Pressure

Rapid unexplained loss of pressure (i.e., not attributable to containment spray or condensation effects) following an initial pressure increase indicates a loss of containment integrity. Containment pressure and sump levels should increase as a result of the mass and energy release into containment from a LOCA. Thus, sump level or pressure not increasing indicates containment bypass and a loss of containment integrity. The 60 PSIG for potential loss of containment is based on the containment design pressure. Containment Hydrogen concentration of 3% is based on the EOPs using 3% Hydrogen as the point to take action for hydrogen control to ensure hydrogen concentration does not reach 4% which is considered the combustible concentration or explosive mixture. Existence of an explosive mixture means a hydrogen and oxygen concentration of at least the lower deflagration limit curve exists. As described above, this EAL is primarily a discriminator between Site Area Emergency and General Emergency representing a potential loss of the third barrier.

The second potential loss EAL represents a potential loss of containment in that the containment cooling fans or the containment spray system (e.g., containment sprays, ice condenser fans, etc., but not including containment venting strategies) are either lost or performing in a degraded manner, as indicated by containment pressure greater than the setpoint at which the equipment was supposed to have actuated.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/05/15	Tier #		1
Change: 0	Group/Category #		2
	K/A #	067 G 2.2.44	
Level of Difficulty: 3	Importance Rating		4.4

Plant Fire on Site: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question: 85

Given the following conditions:

- Plant is at 100% power when the following alarms are received:
  - CB-20/A15 – AI-148 FIRE DETECTION ALARM OR TROUBLE.
  - CB-20/A15 – AI-152 FIRE DETECTION ALARM OR TROUBLE.
- FP-1A, Electric and FP-1B, Diesel Fire Pumps are both RUNNING.
- Sprinkler/Deluge System discharge is indicated.

Subsequently:

- A Security Guard has reported smoke in the Auxiliary Building.
- The Fire Incident Commander has located the fire in the northwest corner of the 1007 ft. elevation of the Auxiliary Building in Room 29.
- The running Charging Pump tripped on low suction pressure.
- Volume Control Tank (VCT) pressure is 0 psig.
- The Shift Manager has determined that a Plant Shutdown is required based on fire conditions.

The next action the Unit Supervisor should direct the crew to perform per AOP-06-1, Fire Emergency - Auxiliary Building Radiation Controlled Areas and Containment, is to…

A. …establish control at the Auxiliary Shutdown Panels AI-185 and AI-179.

B. …dispatch an operator to locally close LCV-218-2, Volume Control Tank Outlet Valve, to restore Charging Pump suction.

C. …trip the reactor and establish Emergency Boration using the High Pressure Safety Injection pumps.

D. …dispatch an operator to locally open LCV-218-3, Charging Pump Suction Valve from the SIRWT, to restore Charging Pump suction.

Answer: D

K/A Match:

As the SRO, applicant must be familiar with all aspects of the Fire Emergency procedure including equipment that must be operated or secured to minimize the effects of the fire. This question requires assessment of conditions and specific knowledge of the procedure to select the appropriate action to address the loss of makeup capability during a fire.

Explanation:

A. Incorrect. Plausible because AOP-06, Fire Emergency contains direction to establish control from the Auxiliary Shutdown Panels (ASP) when the fire affects Control Room control and indication, and the ASPs do include instrumentation and control for the Charging System. Incorrect because transferring control will not improve Charging and Letdown system controls.

B. Incorrect. Plausible because local operation of Charging and Letdown Valves, including closing LCV-218-2, is desirable to restore Charging and Letdown. Incorrect because LCV-218-2 is in the VCT room, which is affected by the fire, and because with VCT pressure at 0 psig, the SIRWT suction path can be accomplished without closing LCV-218-2.

C. Incorrect. Plausible because HPSI Pumps can be used to establish Emergency Boration if Charging Pumps are unavailable. Incorrect because this requires depressurization of the RCS to shutoff head of the HPSI Pumps, which requires a Reactor Trip; this action is not required when other means of establishing Emergency Boration are available.

D. **Correct**. AOP-06-1, Fire Emergency - Auxiliary Building Radiation Controlled Areas and Containment, is used to establish a Charging path. Because the VCT is depressurized, flow can be restored by opening LCV-218-3.

Technical Reference: AOP-06-1, Section 2.0, CAUTIONS and NOTES, Steps 1 through 5, Rev. 3

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-6, Emergency Fire Procedure- Licensed Operator
EO 1.3 - **DESCRIBE** the major recovery actions of this AOP.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

AOP-06-1, Section 2.0, CAUTIONS and NOTES

B. **CAUTIONS**

1. Equipment designated with an asterisk (*****) is **NOT** credited in the Fire Safe Shutdown Analysis for this fire area. This equipment requires close monitoring for proper operation if used.

2. The Fire Safe Shutdown Analysis assumes that RCS makeup from the CVCS is lost and the Safety Injection system is used for inventory control in this fire. If inventory control cannot be established via CVCS, EOP-20, Functional Recovery Procedure, must be implemented to establish inventory control with the Safety Injection system.

C. **NOTES**

1. The actions in section 4.0, Instructions/Contingency Actions are intended to increase the likelihood that the VCT can be isolated from charging pump suction and flow established from the SIRWT or BASTs. Prompt action will increase the probability that the VCT is isolated prior to possible failure of LCV-218-2.

2. The Fire Safe Shutdown Analysis does not credit boric acid pumps CH-4A and CH-4B or check valve CH-166 to allow boric acid supply to the charging pumps with LCV-218-2 open. However, this combination may be available for a fire in Room 29.

3. Depressurizing the VCT may allow for charging pump suction alignment to the SIRWT or BASTs with LCV-218-2 open.

AOP-06-1, Steps 1-5

1. <u>Ensure</u> HCV-208 is open.	1.1 **IF ANY** RCP is operating **AND** HCV-208 is **NOT** open, **THEN** <u>perform</u> the following: a. **IF** the Reactor is critical, **THEN** <u>trip</u> the Reactor. b. <u>Stop</u> all RCPs. c. **IF** the Reactor was tripped, **THEN** <u>IMPLEMENT</u> EOP-00, <u>Standard Post Trip Actions</u>.
2. <u>Isolate</u> letdown by closing **ALL** of the following: • TCV-202***** • HCV-204 • HCV-206***** • HCV-241*****	2.1 **IF** letdown can **NOT** be isolated by Containment isolation valves, **THEN** <u>close</u> **BOTH** of the following: • LCV-101-1***** • LCV-101-2*****
3. <u>Place</u> **ALL** charging pumps in "PULL-TO-LOCK". • CH-1A • CH-1B • CH-1C	

<u>NOTE</u>

The fire in room 29 may result in the complete loss of charging capabilities if LCV-218-2 fails to close. Use of CH-4A or CH-4B may close check valve CH-166, supplying the charging pumps from the boric acid storage tanks. The Fire Safe Shutdown Analysis assumed that the CVCS is not available for boration or inventory control.

4. <u>Align</u> charging pump suction to the SIRWT: a. <u>Open</u> LCV-218-3*****. b. <u>Close</u> LCV-218-2*****. c. <u>Operate</u> charging pumps as necessary to maintain Pressurizer level.	4.1 **IF** charging pump suction from the SIRWT is **NOT** available, **THEN** <u>align</u> charging pump suction to the BASTs: a. <u>Open</u> HCV-268*****. b. <u>Start</u> at least **ONE** of the following: • CH-4A***** • CH-4B*****

c. Ensure **ALL** of the following valves are closed:

- HCV-257*****
- HCV-264*****

d. Operate charging pumps as necessary to maintain Pressurizer level.

c. Ensure **ALL** of the following valves are closed:

- HCV-257*****
- HCV-264*****

d. Operate charging pumps as necessary to maintain Pressurizer level.

5. **IF** the Reactor is critical **AND** charging is available for emergency boration, **THEN** IMPLEMENT AOP-05, Emergency Shutdown

5.1 **IF** the Reactor is critical **AND** charging is **NOT** available for emergency boration, **THEN** maintain steady state Plant operation until charging is recovered.

5.2 **IF** Pressurizer level can **NOT** be restored to or maintained at greater than 45%, **THEN** perform the following:

a. Trip the Reactor.

b. IMPLEMENT EOP-00, Standard Post Trip

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		1
	K/A #	003 A2.01	
Level of Difficulty: 3	Importance Rating		3.9

Reactor Coolant Pump System: Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off.

Question: 86

Given the following conditions:

- Plant is at 100% power.
- Reactor Coolant Pump (RCP) RC-3A parameters are as follows:
  - P3177 - RC-3A Middle Seal Inlet Pressure reads 450 psig.
  - P3178 - RC-3A Upper Seal Inlet Pressure reads 450 psig.
  - P3176 - RC-3A Upper Seal Outlet Pressure (Seal Bleedoff Pressure) reads 50 psig.
  - RC-3A Controlled Bleed Off Flow is 0.1 gpm.
  - RC-3A Seal Bleed Off temperature is 200°F.
  - RC-3A Seal Cavity temperature is 180°F.

(1) Which of the following identifies the condition of Reactor Coolant Pump RC-3A, and
(2) What action is required?

A. (1) RCP has out of specification Seal Cavity and Bleed Off temperatures.
(2) Implement AOP-05, Emergency Shutdown, and monitor RCP-3A for possible AOP-22, Reactor Coolant Leak, entry conditions.

B. (1) RCP has out of specification Seal Cavity and Bleed Off temperatures.
(2) Implement AOP-35, Reactor Coolant Pump Malfunctions, Contact System Engineer and continue operation with increased monitoring.

C. (1) RCP has a clogged pressure breakdown device.
(2) Implement AOP-35, Reactor Coolant Pump Malfunctions, Contact System Engineer and continue operation with increased monitoring.

D. (1) RCP has a clogged pressure breakdown device.
(2) Implement AOP-05, Emergency Shutdown, and monitor RCP-3A for possible AOP-22, Reactor Coolant Leak, entry conditions.

Answer: D

K/A Match:

As the SRO, applicant must evaluate RCP seal conditions and select procedures and define required actions once those procedures are entered. AOP-5 entry conditions are as follows: "Shift Manager decides the implementation of AOP-05 based on conditions requiring an emergency plant shutdown." This question requires assessment of conditions and specific knowledge of the procedure to select the appropriate procedure during reactor coolant pump malfunctions.

Explanation:

A. Incorrect. Plausible because the procedural actions are correct. Out of specification temperatures are 250°F for Seal Bleed Off and 200°F for the Seal Cavity.

B. Incorrect. Plausible because Seal Bleed Off temperature and Controlled Bleed Off flow are approaching out of specification temperatures per OI-RC-9, and entry into AOP-35 is appropriate. Incorrect because AOP-05 requires reactor shutdown.

C. Incorrect. Plausible because with Controlled Bleed Off flow less than 0.5 gpm a clogged pressure breakdown device is the problem. Incorrect because this action applies to one failed seal.

D. **Correct**. With Controlled Bleed Off flow less than 0.5 gpm, and middle seal inlet pressure less than 500 psig infers that a clogged pressure breakdown device is the problem. AOP-05 entry is required as well as monitoring for possible entry into AOP-22, Reactor Coolant Leak.

Technical Reference: AOP-35, Attachment 1, Rev. 7

(Attach if not previously provided including revision number) AOP-05, Entry Conditions, Rev. 12a

OI-RD-9, Table 1, Rev. 78

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-35, RCP Malfunctions-Licensed Operator
EO 1.2 - **DESCRIBE** how the plant responds to malfunctions of the reactor coolant pump or reactor coolant pump support systems.
EO 1.3 - **DESCRIBE** the major recovery actions of this AOP.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

Attachment A - Response to Degraded RCP Seal Conditions

CONDITIONS	ACTIONS
One seal has failed (Seal is considered to have failed if the delta P is less than 200 psid.)	1. Continued operation with increased monitoring. 2. Contact System Engineer.
Two seals have failed.	1. Immediate shutdown per AOP-05. 2. Monitor remaining seal and AOP-22 entry conditions.
Partial blockage of lower pressure breakdown device (Inlet pressure to the middle seal has dropped to 1000 to 500 psig versus a nominal value of 1350 psig and CBO flow has decreased to less then 0.75 gpm.)	1. Contact System Engineer.
Pressure breakdown device is plugged. Inlet to middle seal has dropped below 500 psig and CBO flow is less than 0.5 gpm.	1. Immediate shutdown per AOP-05. 2. Monitor seal parameters and AOP-22 entry conditions.
Failure of three seals.	1. Trip plant and affected RCP.
Bleedoff temperature over 250°F.	1. Immediate shutdown per AOP-05.
Seal cavity temperature over 200°F.	1. Trip plant and affected RCP.

AOP-05, Entry Conditions

2.0 ENTRY CONDITIONS

Shift Manager decides the implementation of this procedure to be appropriate based on conditions requiring an emergency plant shutdown.

OI-RC-9, Table 1

Table 1 - Reactor Coolant Pump RC-3A Normal Operating Parameters

Page 1 of 2

ERF COMPUTER ADDRESS	**DESCRIPTION**	**EXPECTED VALUE AT 532°F/2100 PSIA**	**ALARM VALUE**
L3101	RCP RC-3A Upper Res Level	85% (75-95%)	(1)
L3102	RCP RC-3A Lower Res Level	95% (80-110%)	(1)
P3116	RC-3A Upper Seal Outlet Press	50 psig (40-60 psig)	150 psig
P3117	RCP RC-3A Middle Seal Press	1450 psig (1350-1550 psig)	L-1200 psig/H-1600 psig
P3118	RC-3A Upper Seal Inlet Press	700 psig (600-800 psig)	L-500 psig/H-900 psig
T3103	RC-3A Mtr Lower Guide Brg Temp	130°F (120-140°F)	200°F (2)
T3104	RCP RC-3A Mtr Stator Temp	65°C (55-75°C)	120°C
T3105	RC-3A Mtr Drn Trst Brg Temp	135°F (125-145°F)	200°F (2)
T3106	RC-3A Mtr Upper Trst Brg Temp	170°F (160-180°F)	200°F (2)
T3107	RC-3A Mtr Upper Guide Brg Temp	150°F (140-160°F)	200°F (2)
T3108	RC-3A Mtr ARRD Brg Temp	170°F (160-180°F)	200°F (2)
T3113	RCP RC-3A Lower Seal Temp	120°F (105-135°F)	150°F
T3114	RCP RC-3A Bleedoff Temp	150°F (135-165°F)	L-100°F/H-180°F
T2800	CCW Pump Discharge Temp	70°F (53-90°F)	120°F
F3115	Controlled Bleedoff Flow	1.0 gpm (0.75-1.25 gpm)	L-0.60 gpm/H-2.00 gpm

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		1
	K/A #	022 A2.05	
Level of Difficulty: 4	Importance Rating		3.5

Containment Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS.

Question: 87

Given the following conditions:

- Plant is in MODE 1.
- VA-7C, Containment Ventilation Cooling Fan, is out-of-service for breaker repair.
- The following indications are observed:
  - FT-416, VA-1A Flow Transmitter for VA-3A is reading 200 gpm and lowering.
  - TIC-420, VA-1A Temp for VA-3A is reading 120°F and rising.
  - Component Cooling Water (CCW) Surge Tank level is 40" and lowering.
  - Containment Sump level is normal.

(1) Which of the following is the impact on operations for these indications, and
(2) What action is required?

A. (1) A Component Cooling Water leak is in progress and HCV-400A & HCV-400C must be closed.
(2) Restore VA-7C or place the Reactor in HOT SHUTDOWN.

B. (1) An inadvertent CIAS has caused HCV-400A & HCV-400C to close causing a loss of Containment Cooling heat sink.
(2) Enter Technical Specification 2.0.1 and place the Reactor in HOT SHUTDOWN.

C. (1) An inadvertent CIAS has caused HCV-400A & HCV-400C to close causing a loss of Containment Cooling heat sink.
(2) Restore VA-7C or place the Reactor in HOT SHUTDOWN.

D. (1) A Component Cooling Water leak is in progress and HCV-400A & HCV-400C must be closed.
(2) Enter Technical Specification 2.0.1 and place the Reactor in HOT SHUTDOWN.

Answer: A or D

<u>K/A Match:</u>

As the SRO, applicant must determine the reason for rising temperature and lowering flow on Containment Cooling System Heat Exchanger VA-1A. Once assessed, determine applicable Technical Specification LCO 2.4(2) Required Action.

Explanation:

A. **<u>Correct</u>**. HCV-400A, VA-1A CCW Inlet Valve is one of 2 valves located outside Containment that supplies CCW to VA-1A. A CCW leak is in progress. With one Containment Cooling Unit out of service on the same Train, restore VA-7C within 24 hours or place the Reactor in HOT SHUTDOWN within 12 hours.

B. Incorrect. Plausible because if VA-7D was inoperable there would have been a requirement to enter Technical Specification LCO 2.0.1. If FT-416, CCW Flow Transmitter from Containment Cooling Coil VA-1A fails low (or sees low flow) <u>and</u> a CIAS is present, both HCV-400A & HCV-400C, Inlet and Outlet Valves to VA-1A will close after a short time delay.The loss of heat sources into the CCW system from an inadvertent CIAS that isolates the Containment Coolers would result in a cooldown of the CCW system, which would cause shrink and lowering level and pressure in the CCW surge tank.

C. Incorrect. Plausible because the Technical Specification Required Action is correct. If FT-416, CCW Flow Transmitter from Containment Cooling Coil VA-1A fails low (or sees low flow) <u>and</u> a CIAS is present, both HCV-400A & HCV-400C, Inlet and Outlet Valves to VA-1A will close after a short time delay. Incorrect because if the CIAS had occurred there would be no flow indication from FT-416. The loss of heat sources into the CCW system from an inadvertent CIAS that isolates the Containment Coolers would result in a cooldown of the CCW system, which would cause shrink and lowering level and pressure in the CCW surge tank.

D. Incorrect. Plausible because a CCW leak is occurring. Incorrect because loss of train of Containment Cooling does not require entry into Technical Specification LCO 2.0.1.

Technical Reference:	AOP-11, Step 8.d, Rev. 16
(Attach if not previously provided including revision number)	Technical Specification LCO 2.4(2), Amendment #283
	LP 7-17-11, Slides #103, #106, #178, Rev. 1
	ARP-CB-1/2/3/A1, Windows A-1U and A-1L
	OI-CC-1, Prerequisite 4, Rev. 83

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator
EO 7.0 - **STATE** what plant equipment is covered by the LCOs.

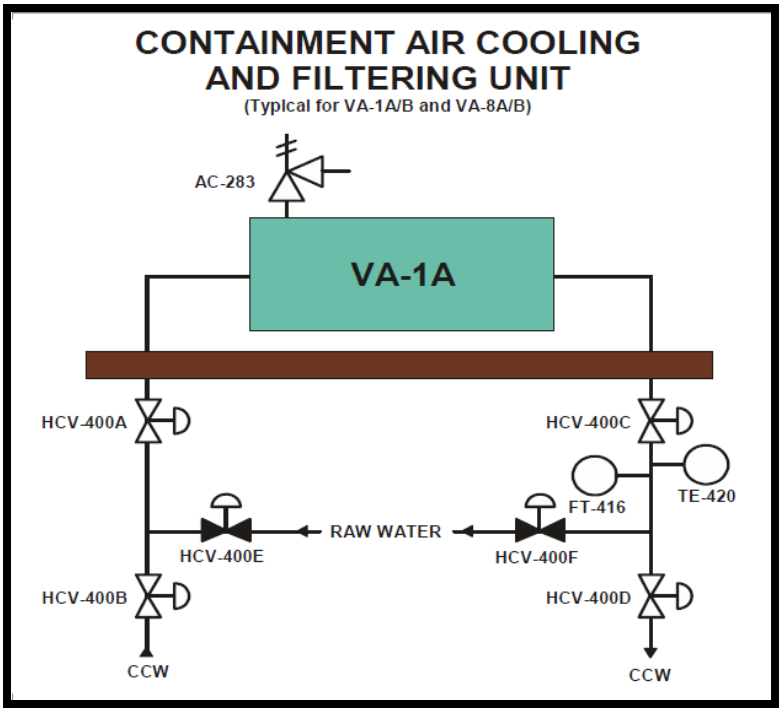
Question Source:	Bank #		
	Modified Bank #		(Note changes or attach parent)
	New	X	

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 _5_

(Slide #103)



TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.4 Containment Cooling (Continued)

b. During power operation one of the components listed in (1)a.i. or ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. For cases involving Raw Water pump inoperability, if the river water temperature is below 60 degrees Fahrenheit, one Raw Water pump may be inoperable indefinitely without applying any LCO action statement. When the river water temperature is greater than 60 degrees Fahrenheit, an inoperable Raw Water pump shall be restored to operability within 7 days or the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable Raw Water pump is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. (This does not include: 1) One Raw Water pump which may be inoperable as described above if the river water temperature is below 60 degrees Fahrenheit or, 2) SI-3A and SI-3B being simultaneously inoperable; or 3) VA-3A and VA-3B, or VA-7C and VA-7D, being simultaneously inoperable. Only two raw water pumps may be out of service during power operations. Either containment spray pump, SI-3A or SI-3B, must be operable during power operations. One train of the containment air cooling and filtering systems (VA-3A and VA-7C), or (VA-3B and VA-7D), must be operable during power operations). If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

NOTE

The following sub-steps may be performed in any order. The actions chosen should be based on those most likely to isolate the source of CCW leakage.
****

CAUTIONS

1. System effects must be evaluated before isolating equipment cooling water.

2. All RCPs must be stopped before closing RCP Lube Oil/Seal Coolers CCW Valves, HCV-438A/B/C/D. Isolation of CCW to operating RCPs will result in RCP Seal damage.

****

8. ATTEMPT TO isolate the CCW leak by performing the following:

   a. Place HCV-474, "SI/CS PUMP CLRS AC INLET VALVE" in "CLOSE".

8. (continued)

   b. Place **ANY** or all of the following switches in "CLOSE":

      - "CONT RM AIR COND VA-46A CCW VALVES HCV-2898A/B"
      - "CONT RM AIR COND VA-46B CCW VALVES HCV-2899A/B"
      - "SPRAY PUMP SI-3A BEARING COOLER CCW VALVES HCV-2813A/B"
      - "LPSI PUMP SI-1A BEARING COOLER CCW VALVES HCV-2808A/B"
      - "HPSI PUMP SI-2A BEARING COOLER CCW VALVES HCV-2810A/B"

8.

c. <u>Place</u> **ANY** or all of the following switches in "CLOSE":

- "HPSI PUMP SI-2C BEARING COOLER CCW VALVES HCV-2812A/B"
- "SPRAY PUMP SI-3B BEARING COOLER CCW VALVES HCV-2814A/B"
- "SPRAY PUMP SI-3C BEARING COOLER CCW VALVES HCV-2815A/B"
- "LPSI PUMP SI-1B BEARING COOLER CCW VALVES HCV-2809A/B"
- "HPSI PUMP SI-2B BEARING COOLER CCW VALVES HCV-2811A/B"

8.

d. <u>Place</u> **ANY** or all of the following switches in "CLOSE":

- "CNTMT CLG COIL VA-1A AC VLVS CONTROL SW HCV-400B/D"
- "CNTMT CLG COIL VA-1B AC VLVS CONTROL SW HCV-401B/D"
- "CNTMT CLG COIL VA-8A AC VLVS CONTROL SW HCV-402B/D"
- "CNTMT CLG COIL VA-8B AC VLVS CONTROL SW HCV-403B/D"

Panel: **CB-1/2/3**	Annunciator: **A1**	Window: **A-1L**

COMPONENT COOLING WATER FROM CONTAINMENT AIR COOLING AND FILTERING UNIT VA-1A HIGH TEMPERATURE

SAFETY RELATED

CC WATER FROM COIL VA-1A TEMP HI

Tech Spec References: 2.4

Initiating Device _TIC-420_ Setpoint _>120°F_ Power _AI-40A_____

OPERATOR ACTIONS

1. Check the CCW temperature on TIC-420 from VA-1A, indicates greater than 120°F (CB-1,2,3).

2. IF CCW outlet temperature is high, THEN raise CCW flow with HCV-400C, CNTMT CLG COIL VA-1A OUTLT ISOL VLV CNTRLR.

3. IF CCW flow indicated on FI-416 (CB-1,2,3) does not rise, THEN ensure the CCW supply and return valves for VA-1A are open:

   - HCV-400A/B, CNTMT CLG COIL VA-1A INLET VALVES
   - HCV-400C/D, CNTMT CLG COIL VA-1A OUTLET VALVES

4. IF CCW high temperature alarm does not clear, THEN place additional CCW Heat Exchanger(s) in service in accordance with OI-CC-1.

IF high temperature is due to a loss of CCW, THEN GO TO AOP-11.

Panel: **CB-1/2/3**	Annunciator: **A1**	Window: **A-1U**

COMPONENT COOLING WATER FROM CONTAINMENT
AIR COOLING AND FILTERING UNIT VA-1A LOW FLOW

CC WATER FROM COIL VA-1A NO FLOW

SAFETY RELATED

Tech Spec References: 2.4

Initiating Device _FC-416A_ Setpoint_ 0 - 559 gpm_ Power _PS-1A/AI-40A_

OPERATOR ACTIONS

1. Check CCW flow from VA-1A on FI-416 indicates less than or equal to setpoint (CB-1,2,3).

2. IF CIAS signal is present, THEN ensure the following CCW valves automatically close after time, 25 to 30 sec., delay:

   - HCV-400A, CNTMT CLG COIL VA-1A INLET ISOL VALVE
   - HCV-400C, CNTMT CLG COIL VA-1A OUTLET ISOL VALVE

3. IF CCW flow from VA-1A is low, THEN ensure HCV-400B, CNTMT CLG COIL VA-1A AC INLET VALVE, and HCV-400D, CNTMT CLG COIL VA-1A AC OUTLET VALVE, are open.

   3.1 Open HCV-400A.

   3.2 Raise flow by throttling open CNTMT CLG COIL VA-1A OUTLT ISOL VLV CNTRLR HCV-400C.

4. IF a loss of CCW is indicated, THEN GO TO AOP-11.

EO *1.2 **(Slide #106)**

Components Cooled by CCW

Containment Air Cooling and Filtering Units

A low flow signal closes the valves if a CIAS signal is present and there is low flow out of the cooler.

Low flow does not close the valves if a CIAS occurs until a time delay of approximately 50 seconds has elapsed.

Holding the switch in CIRC overrides the low flow signal so the operator can re-open the valves.

Low flow does not close the valves unless a CIAS is also present.

EO *1.2, 2.2 **(Slide #178)**

Normal System Operation

Level Control Valve LCV-2801 is manually lined up to fill the CCW surge tank from the Demineralized (Deaerated) Water System.

NOTE: The capability to operate LCV-2801 in AUTO exists, but it is not used. If in AUTO, LCV-2801 would cycle to maintain CCW surge tank level between 41” to 44”.

OI-CC-1, Prerequisite 4

PREREQUISITES

1. Procedure Revision Verification

   Revision No.______________ Date:___________

2. Checklist OI-CC-1-CL-A is completed per OP-1.

3. Deaerated Water System is in operation per OI-DW-4.

4. AC-2 Component Cooling Water Surge Tank level is 41 to 52 inches and pressure is 38.5 to 42 psig.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		1
	K/A #	026 G 2.1.23	
Level of Difficulty: 3	Importance Rating		4.4

Containment Spray System: Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation

Question: 88

Given the following conditions:

- A Main Steam Line Break and Loss of Coolant Accident have occurred in Containment.
- Containment pressure is 18 psig and slowly lowering.
- Both Trains of PPLS, CPHS, SIAS, CSAS and CIAS have actuated.
- Recirculation Actuation Signal (RAS) has actuated.
- Containment temperature is 220°F.
- Both High Pressure Safety injection (HPSI) Pumps discharge pressures and flows have become erratic.

(1) Which of the following has occurred, and
(2) What action is required to restore HPSI flow?

A. (1) Containment Sump blockage has occurred.
(2) Refer to EOP-20, Functional Recovery, RCS Inventory Control, and place Containment Spray Pumps in PULL-TO-LOCK.

B. (1) All 3 Containment Spray Pumps are at runout conditions.
(2) Refer to EOP-20, Functional Recovery, RCS Inventory Control, and place Containment Spray Pumps in PULL-TO-LOCK.

C. (1) Containment Sump blockage has occurred.
(2) Refer to Attachment HR-29, Cooled SI Flow with RAS, then align HPSI Pumps to their Cooled Suction Valves, HCV-349/HCV-350 and start a LPSI pump.

D. (1) All 3 Containment Spray Pumps are at runout conditions.
(2) Refer to Attachment HR-29, Cooled SI Flow with RAS, then align HPSI Pumps to their Cooled Suction Valves, HCV-349/HCV-350 and start a LPSI pump.

Answer: A

K/A Match:

As the SRO, applicant must be familiar with conditions that require securing Containment Spray Pumps when conditions affecting core cooling are apparent. The question is SRO ONLY because the SRO is responsible to coordinate the actions of the Functional Recovery procedures.

Explanation:

A. **Correct**. When both HPSI Pumps are experiencing indications of cavitation (fluctuating discharge pressure/flow/amperage, abnormal noise) and an RAS has occurred, all LPSI and CS Pumps must be placed in PULL-TO-LOCK per EOP-20, Functional Recovery for RCS Inventory Control, Step 11.

B. Incorrect. Plausible because excessive flow from the CS Pumps could be causing the conditions listed. Incorrect because normal containment spray actuation does not include automatic starting of all containment spray pumps.

C. Incorrect. Plausible because Containment Sump blockage is occurring. Incorrect because aligning the HPSI cooled suction to a LPSI pump following RAS will not provide adequate flow.

D. Incorrect. Plausible because t excessive flow from the CS Pumps could be causing the conditions listed. Incorrect because normal containment spray actuation does not include automatic starting of all containment spray pumps. Incorrect because aligning the HPSI cooled suction to a LPSI pump following RAS will not provide adequate flow

Technical Reference: (Attach if not previously provided including revision number)

EOP-20, RCS Inventory Control, Step 11, Rev.

EOP-Heat Removal, Attachment HR-29, Step 1 NOTE & Step 8, Rev. 1

EOP/AOP Attachments, Floating Step F, Steps 5, Rev. 7

EOP-03, Step 32.e, Rev. 38

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-18, Functional Recovery-Licensed Operator
EO 1.4 - **DESCRIBE** the overall strategy of EOP-20.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41 ____
55.43 5

EOP-20, RCS Inventory Control, Step 11

*11. **IF** RAS is present,
AND a loss of SI suction is indicated on at least two pumps by **ANY** of the following:

- Erratic indication of SI flow
- Erratic indication of SI Pump discharge header pressure
- Erratic indication of SI Pump motor current
- SI Pump Trip Annunciator
- Abnormal SI Pump noise

THEN <u>perform</u> **ALL** of the following:

a. <u>Place</u> all LPSI and CS pump Control Switches in "PULL-TO-LOCK".

b. **IF** HPSI Pump performance improves, **THEN** <u>reduce</u> SI flow to meet the minimum required <u>PER</u> Attachment HR-30, <u>Total SI Pump Flow to Match Decay Heat vs. Time After Trip</u>.

b.1 **IF** HPSI Pump performance does **NOT** improve, **THEN** <u>perform</u> the following:

1) <u>Throttle</u> SI flow to 50 gpm per pump.

2) **IF** HPSI Pump performance does **NOT** improve, **THEN** <u>place</u> the affected HPSI Pump(s) Control Switches in "PULL-TO-LOCK".

Floating Step F, Step 5

9. **IF** cooled SI flow is required,
AND RAS has occurred,
THEN IMPLEMENT Attachment HR-29,
Cooled SI Flow With RAS.

Attachment HR-29, Step 1 NOTE

****

CAUTION

Containment Spray must have been terminated prior to implementing this Attachment. Flow rates of less than 200 gpm per pump may cause CS Pump damage.

****

1. Place control switches for **BOTH** of the Containment Spray Valves in "OVERRIDE":
   - HC-344
   - HC-345

Attachment HR-29, Step 8

2. Open **BOTH** of the HPSI Pump Cooled Suction Valves:
   - HCV-349
   - HCV-350

3. Ensure **ANY** or all of the HPSI Pumps, SI-2A/B/C, are operating.

4. Start one Containment Spray Pump, SI-3A/B/C.

5. Throttle **BOTH** of the CCW Outlet Valves to establish desired cooldown rate:
   - HCV-484
   - HCV-485

EOP-03, Step 32

✱ 32. **IF** SIRWT level falls to 16 inches, **THEN** verify that STLS initiates RAS by performing the following:	32.1 **IF** RAS is **NOT** actuated by STLS, **THEN** perform step a or b:
a. Verify **ALL** of the following STLS relays have tripped: • 86A/STLS • 86B1/STLS • 86B/STLS • 86A1/STLS	a. **IF** any of the STLS relays have **NOT** tripped, **THEN** manually initiate using the AI-30A/B key on the STLS Test Switches. • 86A/STLS Test Switch • 86B/STLS Test Switch
b. Verify **ALL** of the following RAS relays have tripped: • 86A/RAS • 86B1/RAS • 86B/RAS • 86A1/RAS	b. Manually establish RAS flow path by performing the following: 1) Open both SI Pump Suction Containment Isolation Valves. • HCV-383-3 • HCV-383-4 (continued)
c. Verify **BOTH** of the following valves are open: • HCV-383-3, SI Pump Suction Containment Isolation Valve • HCV-383-4, SI Pump Suction Containment Isolation Valve	
d. Verify **ALL** of the following valves are closed: • LCV-383-1, SI Pump Suction SIRWT Isolation Valve • LCV-383-2, SI Pump Suction SIRWT Isolation Valve • HCV-385, SIRWT Recirc Valve • HCV-386, SIRWT Recirc Valve • HCV-480, AC-4A CCW Inlet Valve • HCV-481, AC-4B CCW Inlet Valve • HCV-484, AC-4A CCW Outlet Valve • HCV-485, AC-4B CCW Outlet Valve	
e. Ensure **BOTH** LPSI pumps stop. Time: __________	e.1 **IF** LPSI pumps are not stopped, **THEN** IMPLEMENT Floating Step B, LPSI Stop and Throttle.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		1
	K/A #	064 A2.02	
Level of Difficulty: 3	Importance Rating		2.9

Emergency Diesel Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Load, VARS, pressure on air compressor, speed droop, frequency, voltage, fuel oil level, temperatures.

Question: 89

Given the following conditions while in MODE 2:

- Diesel Generator DG-1 was run for Surveillance on the last Shift.
- During Log review the following was noted:
  - FO-1, Diesel Generator Fuel Oil Storage Tank, level was 14,000 gallons.
  - FO-10, Auxiliary Boiler Fuel Oil Storage Tank, level was 11,000 gallons.
- Shortly after Shift Turnover, the following alarm was received:
  - ARP-DG1/D1L, Window B-1 – F.O. LEVEL HI-LOW.

(1) Which of the following is the reason for the alarm, and
(2) What action is required?

A. (1) Level in DG-1 Auxiliary Day Tank is low.
(2) Declare DG-1 inoperable immediately.

B. (1) Level in DG-1 Auxiliary Day Tank is low.
(2) Restore FO-1 to its required level within 48 hours.

C. (1) Level in DG-1 Engine Base Fuel Oil Tank is low.
(2) Declare DG-1 inoperable immediately.

D. (1) Level in DG-1 Engine Base Fuel Oil Tank is low.
(2) Restore FO-1 to its required level within 48 hours.

Answer: B

K/A Match:

As the SRO, applicant must be knowledgeable of alarms servicing The Diesel Generators including Technical Specification Required Action for insufficient fuel oil inventory.

Explanation:

A. Incorrect. Plausible because the source of the alarm is the Auxiliary Day Tank. Incorrect because DG-1 is not inoperable in its current condition. If this amount of lube oil was present (260 gallons) and 48 hours had passed since this was detected, DG-1 would be declared inoperable.

B. **Correct**. There is no level alarm associated with the Engine Base Fuel Oil Tank. FO-1 level must be restored within 48 hours. FO-1 is less than the 16,000 gallons required but the combined inventory in FO-1 and FO-10 is greater than the 6 day supply required (23,350 gallons).

C. Incorrect. Plausible if thought that there was an alarm on the Engine Base Fuel Oil Tank. DG-1 is not inoperable until 48 hours have past and FO-1 level is not restored to 16,000 gallons.

D. Incorrect. Plausible because FO-1 should be restored in 48 hours. Incorrect because there is no level alarm on the Engine Base Fuel Oil Tank. This tank level can be read directly in gallons at the diesel.

Technical Reference: (Attach if not previously provided including revision number)

- Technical Specification LCO 2.7(1) & 2.7(3), Amendment #283
- OP-ST-SHIFT-0001, Page 44, Rev. 121
- ARP-DG1/D1L, Window B-1, Rev. 21
- LP 4-23-7, Slides #207, #224, #317, & #319, Rev. 2

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-13-5, Emergency Diesel Generator-Licensed Operator
EO 1.17 - **STATE** the limiting condition for operation for the EDG and the modification of minimum requirements specific to the EDGs.

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

Question History: Last NRC Exam 2012 NRC Exam, Question #90

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

10 CFR Part 55 Content:
- 55.41 ______
- 55.43 2

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION

2.7 Electrical Systems

Applicability

Applies to the availability of electrical power for the operation of plant components.

Objective

To define those conditions of electrical power availability necessary to provide for safe reactor operation and the continuing availability of engineered safety features.

Specifications

(1) Minimum Requirements

The reactor shall not be heated up or maintained at temperatures above 300°F unless the following electrical systems are operable:

a. Unit auxiliary power transformers T1A-1 or T1A-2 (4,160 V).

b. House service transformers T1A-3 and T1A-4 (4,160 V).

c. 4,160 V engineered safety feature buses 1A3 and 1A4.

d. 4,160 V/480 V Transformers T1B-3A, T1B-3B, T1B-3C, T1B-4A, T1B-4B, T1B-4C.

e. 480 V distribution buses 1B3A, 1B3A-4A, 1B4A, 1B3B, 1B3B-4B, 1B4B, 1B3C, 1B3C-4C, 1B4C.

f. MCC No. 3A1, 3A2, 3B1, 3C1, 3C2, 4A1, 4A2, 4B1, 4C1 and 4C2.

g. 125 V d-c buses No. 1 and 2 (Panels EE-8F and EE-8G).

h. 125 V d-c distribution panels AI-41A and AI-41B.

i. 120V a-c instrument buses A, B, C, and D (Panels AI-40-A, B, C and D).

j. Two (2) 125 V d-c bus No. 1 required inverters: (A and C), or (A and associated swing inverter), or (C and associated swing inverter) AND;

Two (2) 125 V d-c bus No. 2 required inverters: (B and D), or (B and associated swing inverter), or (D and associated swing inverter).

k. Station batteries No. 1 and 2 (EE-8A and EE-8B) including one battery charger on each 125 V d-c bus No. 1 and 2 (EE-8F and EE-8G).

l. Two emergency diesel generators (DG-1 and DG-2).

m. One diesel fuel oil storage system containing a minimum volume of 16,000 gallons of diesel fuel in FO-1, and a minimum volume of 10,000 gallons of diesel fuel in FO-10.

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION
2.7 Electrical Systems (Continued)

o. One of the required inverters may be inoperable for up to 24 hours provided the reactor protective and engineered safeguards systems instrument channels supplied by the remaining three required inverters are all operable and the 120V a-c instrument bus associated with the inoperable inverter is powered from its bypass source.

(3) Modification of Minimum Requirements for Diesel Fuel Oil, Diesel Lube Oil, and Starting Air

The minimum requirements may be modified to the extent that any of the following conditions will be allowed after the reactor coolant has been heated above 300°F. However, the reactor shall not be made critical unless all minimum requirements are met.

a. If the inventory of diesel fuel oil in FO-1 is less than 16,000 gallons and/or FO-10 is less than 10,000 gallons, but the combined inventory in FO-1 and FO-10 is greater than a 6 day supply (23,350 gallons), then restore the required inventory within 48 hours.

b. If one or more diesel generators has lube oil inventory < 500 gallons and > 450 gallons, then restore the lube oil inventory to within limits within 48 hours.

c. If the total particulates of fuel oil stored in FO-1 or FO-10 is not within limits, then restore fuel oil total particulates to within limits within 7 days.

d. If the properties of new fuel oil stored in FO-1 or FO-10 is not within limits, then restore stored fuel oil properties to within limits within 30 days.

e. If one or more diesel generators has the required starting air receiver bank with pressure < 190 psig and > 150 psig, then restore starting air receiver bank pressure to > 190 psig within 48 hours.

f. If the Required Action and associated Completion Time of a, b, c, d or e are not met or one or more diesel generators have diesel fuel oil, lube oil, or a required starting air subsystem not within limits for reasons other than a, b, c, d, or e, then declare the associated DG inoperable immediately.

EO 6.3 **(Slide #224)**

Fuel Oil Subsystem - Major Component Description

Auxiliary Fuel Oil Day Tank (FO-2)

The tank is filled from the discharge of the Fuel Oil Transfer Pumps.

Four level switches provide automatic control of the fuel oil transfer pumps:

1. Normal low level switch starts first pump at 9.5 inches above the bottom of the tank.
2. Backup low level switch starts second pump and initiates an alarm at 7.5 inches above the tank bottom.
3. Normal high level switch stops both transfer pumps at 1-3/4 inches below the top of the tank.
4. Backup high level switch provides additional stop signal to both transfer pumps and initiates an alarm at 1-5/8 inches below the top of the tank.

EO 11.1, *11.4 **(Slide #317)**

System Checks

Fuel Oil Subsystem

Check is performed once daily.

Diesel Generator Fuel Oil Storage Tank (FO-1) level manometer on the south wall of No. 1 diesel room.

Minimum Technical Specification level of tank is 16,000 gallons.

The Water Plant Intake Logs have a 16,500 gallon minimum.

When the tank level is 16,500 gallons, prompt Shift Manager to order fuel oil.

EO 6.2 **(Slide #207)**

Fuel Oil Subsystem – General System Description

Fuel Oil Tanks

<u>NOTE</u>: Both diesel engines are supplied from the in-ground fuel oil tank. Each unit has its own engine base tank and wall mounted auxiliary day tank.

1. FO-1 – 18,000 gallon in-ground Diesel Generator Fuel Oil Storage Tank.
2. FO-10 – 18,000 gallon in-ground Auxiliary Boiler Fuel Oil Storage Tank. (FO-10 can be used to replenish FO-1 in an emergency).
3. 550 gallon engine base fuel oil tank.
4. 300 gallon wall mounted auxiliary day tank.

(Slide #319)

Engine Control Panel	Annunciator: **D1L**	Window: **B-1**

AUXILIARY FUEL OIL TANK HIGH OR LOW LEVEL

SAFETY RELATED

F.O. LEVEL
HI-LOW

Tech Spec References: 2.7

Initiating Device LSH-2 (LCA-3418B) Setpoint 1.625 inches below Day Tank top
Initiating Device LSL-2 (LCA-3418C) Setpoint < 7.5 inches above the Day Tank bottom Power DP1-D1

OPERATOR ACTIONS

1. Check DG-1 Wall Mounted Fuel Oil Day Tank (FO-2-1) level on LI-2134.

2. IF level is high, THEN ensure FO-4A-1 and FO-4B-1, Fuel Oil Transfer Pumps, are automatically shutdown.

3. IF level is low, THEN fill FO-2-1 per OI-DG-1.

4.1 Inspect DG-1 fuel oil system for leakage.

4.2 Notify the Control Room of any fuel oil leakage.

OP-ST-SHIFT-0001, Page 44, DIESEL GENERATOR FUEL INVENTORY

APPLICABLE MODES

Modes 1, 2, 3, 4, and 5

PROCEDURE REFERENCE

TDB-X, Attachment 1, Section 2

TECH SPEC REFERENCE

3.2, Table 3-5, Item 9

ACCEPTANCE CRITERIA

- LI-2107 (Diesel Gen Fuel Oil Storage Tank) is ≥ 16,000 gallons
- LI-2105 (Aux Boiler Fuel Oil Storage Tank) is ≥ 13,000 gallons
  (RCS ≥ 210°F OR Mode 4/5 with RCS pressurized)
- LI-2105 (Aux Boiler Fuel Oil Storage Tank) is ≥ 3,000 gallons (Mode 4 or 5)(*)
- Both Base Tanks are full
- If either tank indicator reads(RCS>210F or Mode 4/5 with RCS Pressurized)):
  LI-2107 < 16,000
  LI-2105 < 13,000
  But the combined level indication is greater than a 6 day supply (25,000 gallons), then restore the required inventory within 48 hours.
- If either tank indicator reads(Mode 4/5 RCS depressurized):
  LI-2107 < 16,000
  LI-2105 < 3,000
  Then restore the required inventory within 48 hours.

REMARKS

1) IF either tank level indicator reads:

   LI-2107 ≤ 16,500 gallons or
   LI-2105 ≤ 16,000 gallons (RCS ≥ 210°F)
   LI-2105 ≤ 4,000 gallons (Mode 4 or 5) (*)

   THEN verify tank levels by direct measurement (tank sounding) and notify the Shift Manager to order fuel oil.

<u>Bank Question</u>:

A local operator has just finished his rounds and reports the following:

- Diesel fuel oil inventory in FO-1 is 13,000 gallons
- Diesel fuel oil inventory in FO-10 is 10,000 gallons
- Diesel generator DG-1 lube oil inventory is 545 gallons
- Diesel generator DG-2 lube oil inventory is 555 gallons
- Neither diesel generator is undergoing any maintenance.

Based on the above report, which of the following technical specifications applies?

A. Enter T.S. 2.7(3)(a) and restore required inventory within 48 hours

<u>B. Declare diesel generator DG-1 inoperable, enter T.S. 2.7(2)(j), and restore to operable within 7 days</u>

C. Declare diesel generator DG-2 inoperable, enter T.S. 2.7(2)(j), and restore to operable within 7 days

D. Declare both diesel generators inoperable, enter T.S. 2.0.1, be in hot shutdown within 6 hours, and cold shutdown within the following 36 hours

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		1
	K/A #	078 G 2.2.44	
Level of Difficulty: 3	Importance Rating		4.4

Instrument Air System: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Question: 90

Given the following conditions:

- Instrument Air (IA) pressure is 45 psig and lowering.
- The Reactor was just tripped and EOP-00, Standard Post Trip Actions, is in progress.

Which of the following is required?

A. Isolate Instrument Air to Containment to maintain Containment Integrity.

B. Stop all but one Main Feedwater Pump to minimize draining of Condensate Storage Tank.

C. Stop all Main Feedwater Pumps to prevent a Reactor Coolant System cooldown.

D. Align Boration to the VCT and establish adequate SHUTDOWN MARGIN.

Answer: C

K/A Match:

As the SRO, applicant must know the operational actions when Instrument Air pressure is lost and understand what impact those actions have on Reactor Safety. Question is SRO ONLY because it tests specific procedure content knowledge.

Explanation:

A. Incorrect. Plausible because Containment Integrity is an issue when Instrument Air continues to deplete. Instrument Air to Containment is isolated earlier in the AOP (Step 9) in an attempt to locate the source of the leak. Once it is determined that the leak is not inside containment, those valves are reopened. Incorrect because Containment Air Valves must 1st be positioned to their "Desired Integrity" position. Isolating instrument air to containment is performed in AOP-17 to attempt to find the source of the air leak, but then IA is restored if the leak is not in containment. A loss of containment integrity is not the reason that the instrument air containment isolation valves are closed.

B. Incorrect. Plausible because draining of the Condensate Storage Tank is a concern during a Loss of Instrument Air. Incorrect because this action minimizes Secondary Pump operation but does not stop all Main Feedwater Pumps.

C. **Correct**. Lowering IA pressure will cause the Main Feedwater Regulating Valves to fail as is when pressure reaches the 70 to 80 psig threshold. At < 50 psig, EOP-00 is entered and MFW Pumps secured to minimize/prevent an RCS cooldown due to failure of the MFW Regulating Valves.

D. Incorrect. Plausible because Boration must be aligned. Incorrect because Boration is directed via the SIRWT.

Technical Reference: AOP-17, Step 13 Contingency Actions, Rev. 15
(Attach if not previously provided including revision number) AOP-17, Steps 9, 10, 16, & 20, Rev. 15

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-17-17, Loss of Instrument Air-Licensed Operator
EO 1.3 - **DESCRIBE** the major recovery actions of this AOP.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 5

AOP-17
Page 12 of 50

INSTRUCTIONS	CONTINGENCY ACTIONS

****

CAUTION

Extended operation with Compressed Air pressure less than 80 psig will result in depletion of Air Accumulator reserves.

****

INSTRUCTIONS	CONTINGENCY ACTIONS
13. **IF** Instrument Air pressure is greater than or equal to 50 psig, **THEN** restore Instrument Air by performing the following:	13.1 **IF** Instrument Air pressure is less than 50 psig, **THEN** initiate a Reactor Shutdown by performing the following:
a. Determine the source of air leakage.	a. Trip the Reactor.
	b. Stop all Main Feed Pumps to prevent an RCS cooldown.
b. Evaluate the need to shutdown the Reactor PER **ONE** of the following procedures: • OP-4, Load Change and Normal Power Operations • AOP-05, Emergency Shutdown	c. IMPLEMENT EOP-00, Standard Post Trip Actions. d. IMPLEMENT the Emergency Plan.

INSTRUCTIONS	CONTINGENCY ACTIONS
13. (continued)	
c. Direct Maintenance to repair the source of air leakage.	
d. Ensure Service Air and Air Dryers have been returned to normal PER OI-CA-1, Compressed Air System Normal Operation.	
e. **IF** air pressure is greater than or equal to 98 psig, **THEN** GO TO Section 5.0, Exit Conditions.	e. Close **ONE** of the following valves to isolate LCV-1190 and prevent draining of Condensate Storage Tank to Condenser Hotwell (Turbine Mezzanine; West Side): • FW-269, "CONDENSATE MAKEUP VALVE LCV-1190 INLET VALVE" • FW-270, "CONDENSATE MAKEUP VALVE LCV-1190 OUTLET VALVE"

INSTRUCTIONS CONTINGENCY ACTIONS

NOTES

1. YCV-1045A and YCV-1045B may open as their air accumulators bleed down, causing FW-10, Steam AFW Pump, to start.
2. Open and close operation of HCV-1107B and HCV-1108B is possible for a minimum of three cycles.

14. Feed S/Gs with AFW to the AFW Nozzles to maintain S/G levels 35-85% NR (73-94% WR) PER Attachment HR-12, Secondary Heat Removal.

15. Ensure HCV-208, RCP Bleedoff to RCDT Isolation Valve, is open.

16. Align Charging Pump Suction to the SIRWT by performing the following:

    a. Open LCV-218-3, Charging Pump Suction SIRWT Isolation Valve.

    b. Close LCV-218-2, VCT Outlet Valve.

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

Containment Isolation Valves which automatically fail closed on a loss of Instrument Air must be closed to meet the Containment Integrity requirements of Technical Specifications 2.0.1, General Requirements, and 2.6, Containment System.

INSTRUCTIONS	CONTINGENCY ACTIONS
20. Verify that the Containment Isolation valves as listed in Attachment F, Failure Positions of Containment Isolation Valves, are in their "Desired Integrity" position.	20.1 **IF** the Containment Isolation Valve(s) are **NOT** in the "Desired Integrity Positions", **THEN** perform the following: a. Manually operate the Containment Isolation Valve(s) as required to ensure compliance with the Technical Specifications. b. IMPLEMENT AOP-12, Loss of Containment Integrity.

INSTRUCTIONS	CONTINGENCY ACTIONS
8. (continued)	
c. GO TO Section 5.0, Exit Conditions.	
9. **IF** Instrument Air pressure continues to lower, **THEN** close **ANY** or all of the Instrument Air Containment Isolation Valves, PCV-1849A/B, to attempt isolation of the rupture.	
10. **IF** Instrument Air pressure continues to lower, **THEN** perform the following:	10.1 **IF** Instrument Air pressure returns to a normal 98-108 psig, **THEN** investigate the leak by performing the following:
a. Open PCV-1849B, Instrument Air Containment Isolation Valve.	a. REFER TO Attachment A, Failure Position of Valves Inside Containment, for failure positions of valves upon loss of air pressure.
b. Open PCV-1849A, Instrument Air Containment Isolation Valve.	b. Consider entering the Containment to investigate the leak.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/17/15	Tier #		2
Change: 2	Group/Category #		2
	K/A #	002 G 2.2.42	
Level of Difficulty: 2	Importance Rating		4.6

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

Question: 91

According to Technical Specifications, if the allowable cooldown rate is exceeded, what corrective action must be taken before the plant can be returned to power?

A. A report must be submitted to the NRC. The NRC must give approval for plant restart.

B. Vessel weld material samples (coupons) must be removed from capsules and tested for fracture toughness properties.

C. The plant must be maintained (soak) in HOT SHUTDOWN for at least 36 hours to allow thermal stresses to be relieved.

D. An analysis must be performed to determine the effects of the out of limits condition on the fracture toughness properties of the RCS.

Answer: D

K/A Match:

As the SRO, applicant must be familiar with Technical Specification Required Actions for exceeding the cooldown rate.

Explanation:

A. Incorrect. Plausible because a report would be submitted. Incorrect because approval for plant restart is not required.
B. Incorrect. Plausible if thought that vessel disassembly is required to test weld material samples following a cooldown violation. Incorrect because this action is not required.
C. Incorrect. Plausible because a requirement to be in COLD SHUTDOWN within 36 hours is part of the Required Actions. Incorrect because performing a soak is not required.
D. **Correct**. As outlined in Technical Specification LCO 2.1.2 Required Actions.

Technical Reference: Technical Specification LCO 2.1.2, Amendment #283

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator EO 7.1 - **DISCUSS** the basis for the LCOs.

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

Question History: Last NRC Exam 2005 NRC Exam, Question #92

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

10 CFR Part 55 Content:	55.41	
	55.43	2

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

1. Immediately initiate action to restore the temperature or pressure to within the limit.

2. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

3. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		2
	K/A #	034 A2.03	
Level of Difficulty: 4	Importance Rating		4.0

Fuel Handling System: Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Mispositioned fuel element.

Question: 92

Given the following conditions:

- You have assumed the shift as the Fuel Handling Supervisor.
- OP-12, Fueling Operations, is in progress for Core reloading.
- The off-going fuel handlers turnover that the last Fuel Assembly moved would NOT engage the Fuel Alignment Pins with the holes in the Core Support Plate.

(1) If NOT corrected, what is the impact of this mispositioned Fuel Assembly on proper vessel reassembly, and
(2) What action is required to complete the Core reloading?

A. (1) The Fuel Assembly is positioned too high, causing the Upper Guide Structure to NOT fully seat without damaging the Fuel Assembly.
(2) Return the Fuel Assembly to the Upender, then insert the Fuel Assembly Guide with FH-1 to align and insert the Fuel Assembly.

B. (1) The CEDM will NOT latch correctly onto the Fuel Assembly because the top of a Fuel Assembly will be laterally displaced after the Upper Guide Structure is installed.
(2) Return the Fuel Assembly to the Upender, then insert the Fuel Assembly Guide with FH-1 to align and insert the Fuel Assembly.

C. (1) The Fuel Assembly is positioned too high, causing the Upper Guide Structure to NOT fully seat without damaging the Fuel Assembly.
(2) Approve inserting the Fuel Assembly with a 90 degree orientation change.

D. (1) The CEDM will NOT latch correctly onto the Fuel Assembly because the top of a Fuel Assembly will be laterally displaced after the Upper Guide Structure is installed.
(2) Approve inserting the Fuel Assembly with a 90 degree orientation change.

Answer: A

K/A Match:

As the SRO, applicant must be aware of the reassembly process of the Reactor and knowledgeable of the procedures that allow Fuel Assemblies to be moved or repositioned.

Explanation:

A. **Correct**. Per OP-12, Attachment 8, Special Fuel Handling Techniques, the Fuel Assembly Guide is used to insert mispositioned fuel assemblies. Fuel assemblies that do not fully insert would cause the Upper Guide Structure to damage the Fuel Assembly due to being vertically displaced from the intended position.

B. Incorrect. Plausible because the corrective action using the Fuel Assembly Guide is correct. Incorrect because the Upper Guide Structure, once installed, aligns the CEDM with the CEA. The CEDMs are not latched until the Upper Guide Structure and Reactor Head are in place, and this would eliminate any mispositioning associated with latching the CEAs.

C. Incorrect. Plausible because the consequence of the mispositioned Fuel Assembly not being fully inserted would cause the Upper Guide Structure to damage the Fuel Assembly. Incorrect but plausible because performing an orientation change may allow the assembly to fully insert into the Core Support Plate, but this cannot be approved by the Fuel Handling Supervisor; this is only approved by the Reactor Engineer.

D. Incorrect. Plausible because performing an orientation change may allow the assembly to fully insert into the Core Support Plate, but cannot be approved by the Fuel Handling Supervisor. Plausible because the CEDM would be misaligned if the Upper Guide Structure did not ensure alignment between the CEA and the CEDM. Incorrect because both the consequence of the misalignment and the action taken to correct the problem are both wrong.

Technical Reference: OP-12, Attachment 1, PRECAUTIONS & Attachment 8, Step 6, Rev. 70
(Attach if not previously provided including revision number) LP 7-11-21, Slide #66, #67, #92, #109 & #110, Rev. 0

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-11-13, Fuel Handling Machine-Licensed Operator
EO 2.1 - **DISCUSS** the prerequisites and precautions associated with fuel handling equipment and the refueling machine.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ____
55.43 7

6. Use of the Fuel Assembly Guide

6.1 Repeated attempts to load a Fuel Assembly in the Core Support Plate have failed due to failure of the Fuel Alignment Pins to engage holes in the Core Support Plate.

6.2 A Core location for the Guide has been selected.

6.3 The bowed Fuel Assembly is stored in the Upender.

CAUTION

1. Hoist load must be closely monitored while positioning the Fuel Assembly Guide.
2. The Fuel Assembly Guide can be used with the approval of the Reactor Engineer.

6.4 With the EMPTY HOIST BYPASS selector switch in ON, place the Fuel Assembly Guide in the selected Core location.

6.5 Insert the bowed Fuel Assembly per OI-FH-1 or other alternate procedures in this attachment.

6.6 At a convenient step in the Fuel Movement Sequence, remove the Fuel Assembly Guide from the Core.

From OP-12, Attachment 8, Step 6

OP-12, Attachment 1, PRECAUTIONS

> **NOTE**
>
> 1. With the exception of Appendix A, all changes to this procedure must be made per SO-G-30. Appendix A changes may be made using Form O from the Nuclear Material Accountability procedure NMA-3.
>
> 2. The Reactor Engineer must approve all changes to Appendix A.

19. If a change to Appendix A requires a Fuel Assembly to be temporarily stored in a Core Location, the Reactor Engineer must be notified to determine an alternate core location using the k-Infinity Map in Appendix A .**[AR 14254]**

20. A New Fuel Assembly may be stored in the Elevator while other Spent Fuel Movements are completed.

21. The Reactor Engineer or his designee shall provide the Control Room a sequence of Fuel Movements for the Shift (Appendix A).

EO 4.3 **(Slide #66)**

Major Component Description

Core Support Assembly

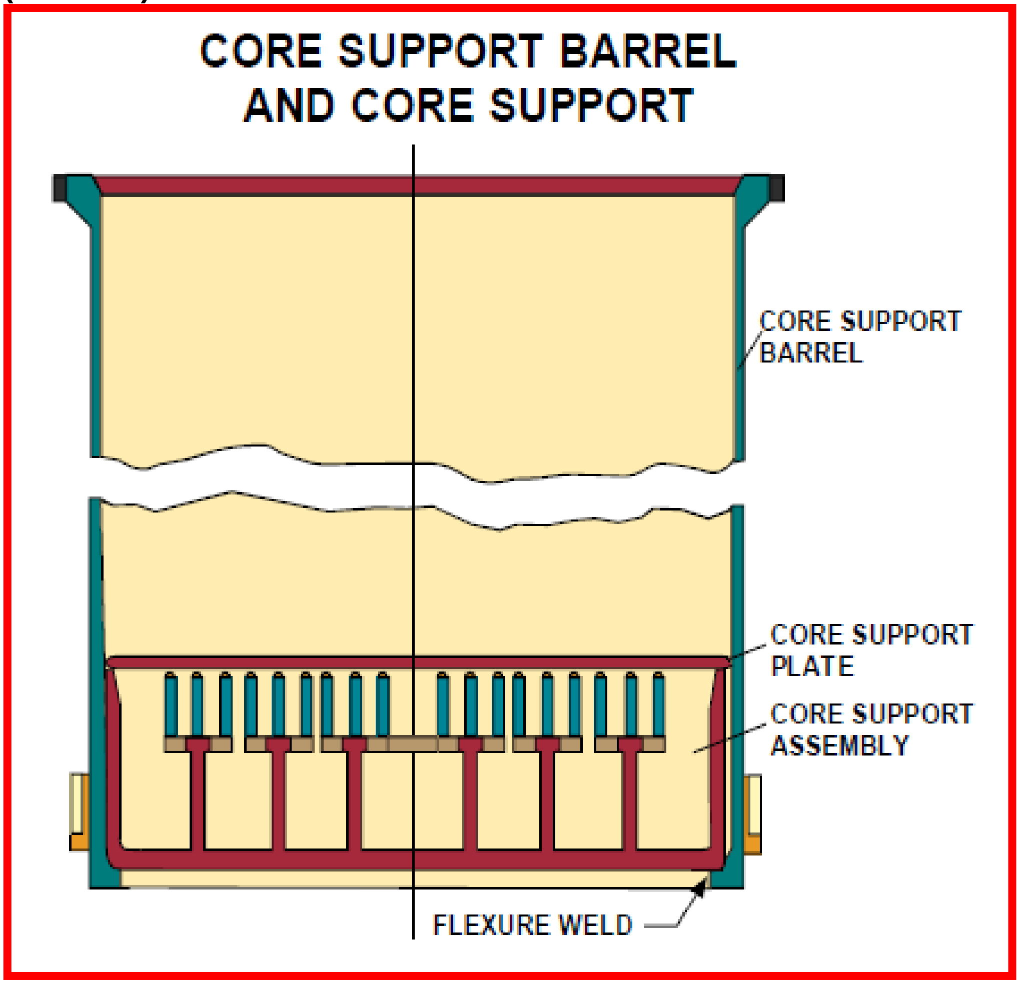
Core Support Plate and Support Columns

The support plate helps align the fuel assemblies, directs coolant flow into the fuel assemblies and transmits the core weight to the core support barrel via the columns and support beams beneath it.

The core support plate is 120” in diameter, 2” thick Type 304 stainless steel.

Flow distribution holes are machined into the plate to provide the necessary flow to the fuel assemblies. Fuel assembly locating holes (five for each assembly, only four are used) are also machined into the plate.

(Slide #67)



EO 4.4 **(Slide #92)**

Major Component Description

Reactor Core

Fuel Assembly

The lower end fitting is a cast fixture made of Type 304 stainless steel, machined to accept alignment pins which fit in corresponding holes in the core support plate.

Alignment pins (4) provide later alignment of the lower end of the fuel assembly.

The alignment pin length is such that the spacing between fuel assemblies will not be altered even during postulated accident conditions when a fuel assembly is lifted into contact with the Upper Guide Structure.

The alignment pins are held in place by stainless retaining pins that are crimped in place. (CR200501629, Loose part found in RV was an alignment pin.)

The lower end fitting contains two flow channels.

EO 4.4a **(Slide #109)**

Major Component Description

Upper Guide Structure

The UGS aligns and laterally supports the upper end of the fuel assemblies.

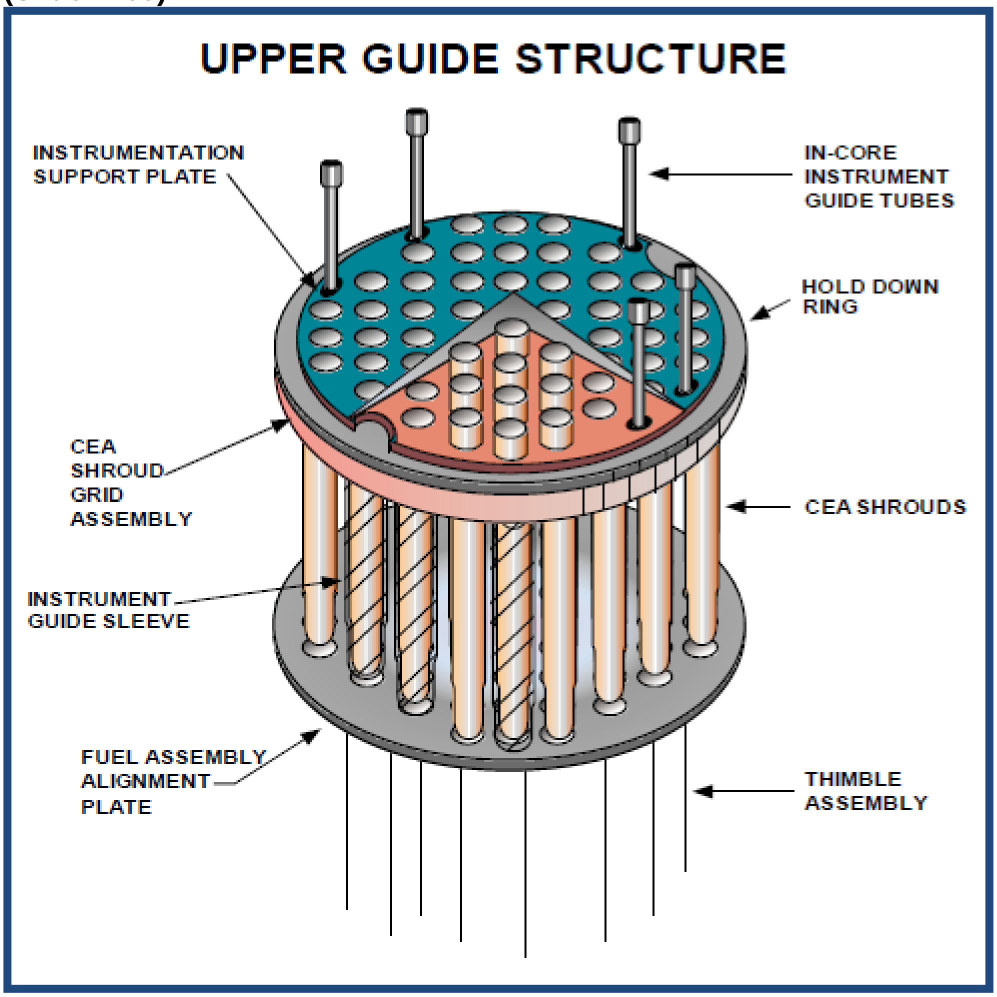
Maintains the CEA spacing.

Prevents fuel assemblies from being lifted out of position during pressure transients and severe accident conditions.

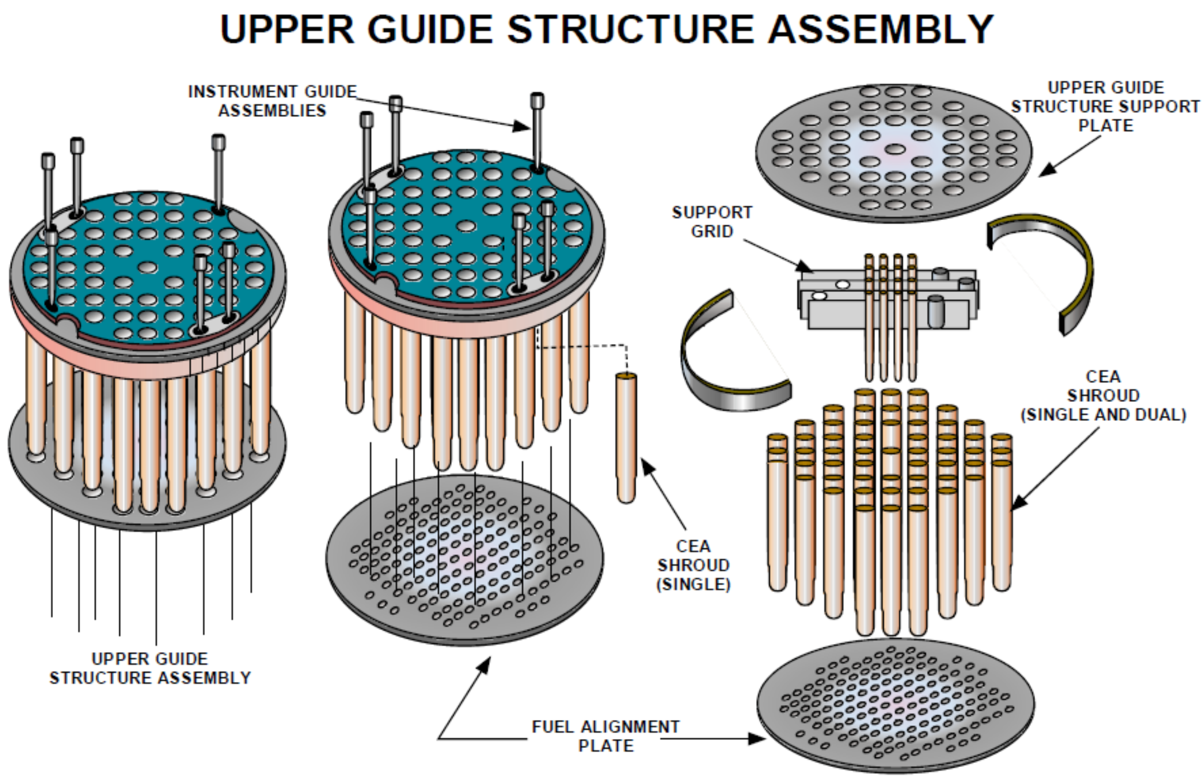
Provides a flow shroud for the CEAs which protects the CEAs from the effect of coolant cross flow in the upper plenum.

Supports the in-core instrumentation plate and guide tubes.

(Slide #109)



(Slide #110)


UPPER GUIDE STRUCTURE ASSEMBLY
INSTRUMENT GUIDE ASSEMBLIES
UPPER GUIDE STRUCTURE SUPPORT PLATE
SUPPORT GRID
CEA SHROUD (SINGLE AND DUAL)
CEA SHROUD (SINGLE)
UPPER GUIDE STRUCTURE ASSEMBLY
FUEL ALIGNMENT PLATE

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		2
Change: 1	Group/Category #		2
	K/A #	055 G 2.4.46	
Level of Difficulty: 4	Importance Rating		4.2

Condenser Air Removal System: Emergency Procedures/Plan: Ability to verify that the alarms are consistent with the plant conditions.

Question: 93

Given the following conditions:

- Condenser Off Gas is in its normal at-power alignment.
- ARP-AI-33C/A33C, Window 34 – RM-057 CONDENSER OFF GAS HIGH RADIATION, is in HI ALARM.
- Chemistry has been directed to sample the Reactor Coolant System (RCS) and reports RCS XE-133 concentration at 5 µCi/gram and primary to secondary leakage is 40 gpd.

As Shift Manager, which of the following would be procedurally required?

A. Commence an OP-4 Shutdown to reduce power to less than 50% power within 1 hour per AOP-21, Reactor Coolant System High Activity.

B. Direct Condenser Off Gas Effluent be aligned to the Auxiliary Building Ventilation Stack per OI-CE-1, Condenser Evacuation System Normal Operation.

C. Increase Steam Generator Blowdown flow to maximum per HR-21, Blowdown Operation.

D. Implement the Emergency Plan and declare a Notification of Unusual Event for RCS Leakage per EPIP-OSC-1, Emergency Classification.

Answer: B

K/A Match:

As the SRO, applicant must be able to establish correct plant conditions when Radiation Monitoring Systems go into alarm. This question requires assessment of conditions and specific knowledge of the procedure to select the appropriate procedure during reactor coolant high activity related to Condenser Evacuation System alignment. The high radiation alarm in conjunction with the chemistry sampling results indicates a steam generator tube leak is in progress. The correct answer is an action directed from station procedures associated with these conditions.

Explanation:

A. Incorrect. Plausible because this power reduction is required for steam generator tube leakage. Incorrect because this power level requirement is not required until leakage exceeds 75 gpd with an increase in leak rate greater than 30 gpm in an hour.

B. **Correct**. As required by OI-CE-1, the alternate operation flow path is through the Auxiliary Building Vent Stack which provides high range radiation monitoring capability for Reactor Coolant Activity $\geq$ 15 μCi/g Xe-133 and per AOP-22, RCS Leak, for steam generator tube leakage to minimize the spread of contamination.

C. Incorrect. Plausible because increasing steam generator blowdown flow provides increased cleanup capability and would reduce secondary plant contamination levels. Incorrect because blowdown flow is reduced per AOP-22 to minimize the spread of contamination due to the discharge of steam generator blowdown.

D. Incorrect. Plausible because the emergency plan is implemented for reactor coolant system leaks, and a NOUE is declared for leakage that exceeds technical specification limits in accordance with EPIP-OSC-1 (SU5). Incorrect because a NOUE is not declared for until boundary leakage exceeds 10 gpm.

Technical Reference:	OI-CE-1, Attachment 2, Step 1 NOTE, Rev. 29
(Attach if not previously provided including revision number)	Technical Specification LCO 2.1.4, Amendment #283
	ARP-AI-33C/A33C, Window 38, Rev. 32
	LP 7-12-3, Slides #262 to #264, Rev. 1
	OP-ST-SHIFT-0001, Page 16, Rev. 121
	AOP-21, Step 8, Rev. 9

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-12-3, Radiation Monitoring System-Licensed Operator
EO 4.0 - **EXPLAIN** the operations, actuations and applications of the individual radiation monitors.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ________

55.43 4

TECHNICAL SPECIFICATIONS

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.4 Reactor Coolant System Leakage Limits

Applicability

Applies to the leakage rates of the reactor coolant system whenever the reactor coolant temperature (T_{cold}) is greater than 210 °F.

Objective

To specify limiting conditions of the reactor coolant system leakage rates.

Specifications

To assure safe reactor operation, the following limiting conditions of the reactor coolant system leakage rates must be met:

(1) RCS operational LEAKAGE shall be limited to:

a. No Pressure Boundary LEAKAGE,
b. 1 gpm unidentified LEAKAGE,
c. 10 gpm identified LEAKAGE,
d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

(2) If RCS operational LEAKAGE limits of (1), above, are not met for reasons other than Pressure Boundary LEAKAGE or primary to secondary LEAKAGE, then reduce LEAKAGE to meet limits within 4 hours.

(3) If the Required Action and associated completion time of (2), above, is not met, OR Pressure Boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limits, then be in MODE 3, Hot Shutdown, within 6 hours AND be in MODE 4, Cold Shutdown, within 36 hours.

EO 4.0 **(Slide #262)**

RM-057

EC 37573 replaced the sampler and both RM-057 ratemeters in 2006.

The replacement ratemeters increased the detector range from 7 to 9 decades which allow for the detection of primary to secondary leakage down to 5 gallons per day.

The increased range allows the monitor to span up to the lowest detection range of the accident range main steam line monitor RM-064.

EO *1.4 **(Slide #263)** *RM-057* Condenser evacuation exhaust which is normally vented to atmosphere via a vent pipe through the Turbine Building room can be redirected to the Auxiliary Building Vent Stack for additional high range monitoring of RM-052 or RM-062/63. Standing order G-105 and the ARP for RM-057 give directions to make the realignment. Also, the realignment is required should RCS Xenon-133 concentration exceed 15.0 µCi/gram per OI-CE-1 regardless of any evidence of primary to secondary leakage

EO *4.1 **(Slide #264)** *RM-057* When the Control Room ratemeter reaches its HIGH setpoint it will cause RCV-978 to automatically close, terminating Steam Turbine 6^{th} stage extraction supply to the Aux. Steam System.

Attachment 2 - Condenser Evacuation Discharge Flow Paths

PREREQUISITES

5. Procedure Revision Verification:

Revision No. _______ Date:________

2.0 Radiation Monitor RM-057 is in operation per OI-RM-1 or grab samples are being taken per the ODCM.

3.0 RM-052 and/or RM-062 is in service in the Auxiliary Building Step 2 per OI-RM-1.

PROCEDURE

NOTES
1. The Shift Manager will determine which of the following flow paths will be used.
2. Alternate operation flow path is through the Auxiliary Building vent stack which provides high range radiation monitoring capability for Reactor Coolant Activity ≥ 15µCi/g Xe-133.

1. IF aligning for normal flow path,
THEN perform the following:

   a. Open VD-423, Condensers FW-1A&B Evacuation Pumps FW-8A-8C Discharge To Atmosphere Isol Valve (Turb Bldg 1036).

Panel: **AI-33C**	Annunciator: **A33C**	Window: **34**

RM-057 CONDENSER OFF GAS HIGH RADIATION

Page 1 of 2

RM-057
CONDENSER OFF GAS
HIGH RADIATION

SAFETY RELATED

Tech Spec References: None

Initiating Device RM-057 Setpoint ALERT/HIGH (TDB-IV.7) Power AI-40C

OPERATOR ACTIONS

4. Verify Alert/High Radiation per TDB-IV.7on RM-057 ratemeter. (AI-33B)

5. IF RM-057 alarm is high, THEN perform the following:

   5.1 Direct the Shift RP Technician to perform dose assessment per EPIP-EOF-6.

   5.2 Ensure RCV-978, Extr 6 to Aux Stm Hdr Control Valve is closed.

   5.3 Close the following valves:

   - HCV-1387A, S/G RC-2B Blowdown Isolation Valve INBD
   - HCV-1387B, S/G RC-2B Blowdown Isolation Valve OUTBD
   - HCV-1388A, S/G RC-2A Blowdown Isolation Valve INBD
   - HCV-1388B, S/G RC-2A Blowdown Isolation Valve OUTBD

   5.4 Align Condenser Evacuation discharge to the Aux Building Stack per OI-CE-1.

   5.5 IF Primary-to-Secondary leakage exceeds 1 gallon per day, THEN refer to SO-G-105.

(continue)

PROBABLE CAUSES

- Steam Generator Tube Leak
- I&C calibration of RM-057
- Malfunction of RM-057

REFERENCES

11405-E-406 Sh 5 22662 CH-ODCM-0001 TDB-IV.7
EM-057 12182

Panel: **AI-33C**	Annunciator: **A33C**	Window: **34**
RM-057 CONDENSER OFF GAS HIGH RADIATION **SAFETY RELATED**		Page 2 of 2

OPERATOR ACTIONS (continued)

6. IF RM-057 alarm is in alert, THEN perform the following:

    6.1 Notify Radiation Protection.

    6.2 Monitor RM-054A and RM-054B ratemeters to confirm activity increase. (AI-33A)

    6.3 Direct Chemistry to take confirmatory grab samples, if required.

    6.4 IF primary to secondary leakage exceeds 1 gallon per day, THEN refer to SO-G-105.

7. IF RM-057 alarm is clear OR alarm is due to radiation monitor failure, THEN perform the following:

    7.1 Comply with the requirements of CH-ODCM-0001.

    7.2 Initiate notification to the Work Week Manager of the radiation monitor malfunction.

OP-ST-SHIFT-0001, Page 16 of 59

APPLICABLE MODES:

Modes 1, 2, 3, 4 and 5

PROCEDURE REFERENCE:

- TDB-IV.7
- OP-ST-RM-0002
- CH-ODCM-0001
- Standing Order SO-G-105

TECH. SPEC. REFERENCE:

- 2.1.4(5)
- 2.21, Table 2-10, Item 3
- 3.1, Table 3-3, Item 3.a
- 3.1, Table 3-3, Item 5.a

ACCEPTANCE CRITERIA:

- ALERT SP per TDB-IV.7
- RM-057 counts have not doubled from previous day

REMARKS: If counts on RM-057 have doubled, contact Shift Chemist for primary-secondary sample and implement Standing Order SO-G-105.

AOP-21 Step 8

1. **IF** Xe-133 is greater than 10.0 μCi/gm, **THEN** contact the Reactor Engineer for further guidance.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		3
Change: 1	Group/Category #		1
	K/A #	2.1.35	
Level of Difficulty: 3	Importance Rating		3.9

<u>Conduct of Operations</u>: Knowledge of the fuel-handling responsibilities of SROs.

Question: 94

Given the following conditions:

- You assumed the shift as the Fuel Handling Supervisor at 1800.

During Shift Turnover, the off going Fuel Handling Coordinator informed you that:

- Shutdown Cooling (SDC) was lost for approximately one hour when Low Pressure Safety Injection (LPSI) Pump SI-1B tripped at 1500.
- LPSI Pump SI-1A is currently providing SDC while troubleshooting is continuing on LPSI Pump SI-1B.

The time is now 2000.

- An irradiated Fuel Assembly has been removed from the Upender.
- The refueling crew has just changed personnel running FH-1.
- They have re-verified prerequisites for OI-FH-1 and OP-12, and are awaiting your approval to index the refueling bridge to the required core location.
- LPSI Pump SI-1A has tripped and CANNOT be restarted.
- A Containment Spray Pump is in the process of being aligned to provide SDC.

Based on the above information, which of the following actions will you direct to be taken?

A. Insert the current fuel assembly in the required core location and then suspend further fuel movement until Shutdown Cooling can be restored.

B. Do NOT move the fuel assembly from its current location and suspend further fuel movement until Shutdown Cooling can be restored.

C. Insert the current fuel assembly in the Upender and then suspend further fuel movement until Shutdown Cooling can be restored.

D. Complete insertion of current fuel assembly and continue fuel loading provided that Shutdown Cooling can be restored to operation by 2100.

Answer: C

K/A Match:
As the SRO, applicant must be knowledgeable of the Required Action for moving irradiated fuel assemblies when Technical Specification LCOs are not met.

Explanation:

A. Incorrect. Plausible because fuel movement must be stopped until Shutdown Cooling (SDC) is restored. Incorrect because the irradiated Fuel Assembly should not be moved to the core and inserted instead of inserting the assembly into the upender.

B. Incorrect. Plausible because fuel movement must be stopped until SDC is restored. Incorrect because per the Technical Specification Definitions for REFUELING OPERATIONS and CORE ALTERATIONS, suspension of these shall not prevent completions of movement of a component to a safe, conservative position. Therefore, the Fuel Assembly should not be left its current position.

C. **Correct**. Per Technical Specification LCO 2.8.1(3), Note 1: SDC Loop can be secured for ≤ 1 hour per 8 hour period. SDC was lost for an hour at 1500, so the 8 hour period would go until 2300. SDC was again lost at 2000, which is within the 8 hour period of time. Therefore, this Note does NOT apply. The Required Action is to immediately suspend loading of irradiated fuel assemblies into the reactor core. This is an irradiated assembly, so they need to suspend loading of this Fuel Assembly and place it in a safe condition (inserted into the upender).

D. Incorrect. Plausible because the Fuel Assembly was be returned to the Upender. Incorrect because Note 1 applies to any 8 hour window and that just because the shift changed, the Note 1 clock is not reset.

Technical Reference: Technical Specification LCO 2.8.1(3), Amendment #283
(Attach if not previously provided including revision number) Technical Specification Definitions, Amendment #283

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator
EO 3.0 – **STATE** what plant equipment is covered by the Limiting Conditions for Operations.

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

Question History: Last NRC Exam 2012 NRC Exam, Question #94

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

10 CFR Part 55 Content: 55.41 ______
55.43 2

OP-12, Precautions 4

PRECAUTIONS

1. The Shift Manager shall control all Refueling Operations.

2. At no time shall Fuel Assemblies, CEAs, or Sources be moved without the knowledge and approval of the Control Room.

3. If voice communications between the Control Room, FH-1, or FH-12 are lost, all Refueling Operations must stop until communications are restored.

4. If a Fuel Assembly is in transit to the Core, the Fuel Handling Coordinator in Containment may direct the assembly to be returned to the Upender Area.

2.0 **LIMITING CONDITIONS FOR OPERATION**
2.8 Refueling
2.8.1 Refueling Shutdown

2.8.1(3) Shutdown Cooling System - High Water Level

Applicability
Applies to shutdown cooling requirements in MODE 5 with fuel in the reactor and with one or more reactor vessel head closure bolts less than fully tensioned, and the refueling cavity water level ≥ 23 ft. above the top of the core.

Objective
To minimize the possibility of a loss of shutdown cooling accident occurring inside containment that could affect public health and safety.

Specification
One OPERABLE Shutdown Cooling loop shall be IN OPERATION except as noted below:

1. The required Shutdown Cooling loop may be removed from operation for ≤ one hour per 8 hour period, provided no operations are permitted that would cause dilution of the RCS boron concentration.

2. The required Shutdown Cooling loop may be inoperable for up to eight hours provided (1) no operations are permitted that would cause dilution of the RCS boron concentration, (2) no CORE ALTERATIONS or REFUELING OPERATIONS are taking place, (3) all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere are closed within 4 hours, and (4) at least one loop is available under administrative controls.

Required Actions

(1) With no Shutdown Cooling loop IN OPERATION (except as allowed by notes 1 or 2 above),

a. Suspend operations involving a reduction in reactor coolant boron concentration immediately, and

b. Suspend loading of irradiated fuel assemblies into the reactor core immediately, and

TECHNICAL SPECIFICATION

DEFINITIONS

REACTOR OPERATING CONDITIONS (Continued)

Cold Shutdown Condition (Operating Mode 4)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ SHUTDOWN BORON CONCENTRATION but < REFUELING BORON CONCENTRATION.

Refueling Shutdown Condition (Operating Mode 5)

The reactor coolant T_{cold} is less than 210°F and the reactor coolant is ≥ REFUELING BORON CONCENTRATION.

Refueling Operation

Any operation involving the shuffling, removal, or replacement of irradiated fuel outside of the reactor pressure vessel. The suspension of any REFUELING OPERATION shall not preclude completion of movement of a component to a safe, conservative position.

TECHNICAL SPECIFICATION

DEFINITIONS

Core Alteration

The movement or manipulation of fuel, sources, reactivity control components, or other components affecting reactivity within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe, conservative position.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 11/17/15	Tier #		3
Change: 2	Group/Category #		1
	K/A #	2.1.6	
Level of Difficulty: 3	Importance Rating		4.8

Conduct of Operations: Ability to manage the control room crew during plant transients.

Question: 95

How does the Control Room Supervisor manage procedure use during a transient per OPD-4-09, EOP/AOP Users Guidelines?

A. Decides if step-by-step instruction of immediate action steps is required before operators take action.

B. Allows operators to perform a procedure-directed critical action without crew notification to expedite plant recovery.

C. Decides whether or not an EOP/AOP Floating Step is Continuously Applicable.

D. Allows an AOP to take precedence over an EOP to expedite plant recovery.

Answer: D

K/A Match:
As the SRO, applicant must manage concurrent use of procedures per EOP/AOP Users Guidelines.

Explanation:

A. Incorrect. Plausible because the CRS controls the execution of the EOPs. Incorrect because immediate action steps are performed by operators without procedure-in-hand and without step-by-step instruction by the CRS.
B. Incorrect. Plausible if thought that this did not affect the CRS command-and-control function. Incorrect because any action that impacts the ability of the CRS to maintain control over the transient or the ability of the crew to maintain control of the plant is not allowed.
C. Incorrect. By definition, EOP/AOP Floating Steps are considered Continuously Applicable steps, once the optimal recovery procedure is entered. Therefore, this is not a decision that the CRS makes, which makes this choice incorrect.
D. **Correct**. Per OPD-4-09, the CRS can allow an AOP to take precedence over in EOP when a procedure conflict arises. This option is performed on a case-by-case basis.

Technical Reference: OPD-4-09, Step 4.5.1, Rev. 20
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-67-5, OPD Manual-Licensed Operator
EO 2.0 - **DESCRIBE** the Performance Standards listed in the OPD Manual.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ______
55.43 5

4.5 Transient Response Procedures

4.5.1 The EOP's and AOP's have usage characteristics that set them apart from standard Continuous Use procedures. Those characteristics include the allowance for taking certain pre-approved steps from memory, the use of continuously applicable actions which can be taken regardless of step sequence if the conditions for the step are met and the concurrent use of step(s) and/or procedures, the ability to repeat steps as needed during the conduct of the procedure, and the ability to perform more than one step at a time.

A. Actions taken from memory:

1. The operator(s) shall perform immediate action steps from memory and memory aids when plant conditions requiring the actions are present. The operator shall be able to perform immediate action steps without the procedure in hand and without step by step instruction from the Control Room Supervisor.

2. When the steps are complete, the operator shall inform the CRS. The operator shall also inform the CRS of any contingency actions required to accomplish the immediate action steps. The CRS shall ensure the correct execution of the steps.

3. Performance of steps other than immediate actions from memory is allowed provided that the action is simple enough to be performed reliably from memory and training. The actions are necessary for the effective and efficient implementation of the mitigation strategy. Timely performance of the action is necessary to stabilize the plant that would or could lead to a plant trip.

B. Continuously Applicable Steps:

1. Continuously Applicable or Non-sequential steps are any step or action that is continuously applicable throughout the performance of a procedure or during a clearly delineated portion of the procedure.

2. The EOP/AOP Floating Steps are considered Continuously Applicable steps.

4.5.1 (continued)

C. Task Completion

1. The Control Room Supervisor may proceed to the next step in the procedure when the present step has been initiated, the action is in progress, and the CRS judges there to be no adverse consequence to the parallel performance of following steps. This action shall not be construed as conflicting with the normal navigation and progression through procedures written in two column format.

2. Branching to an applicable Attachment also is considered when determining the progress of the step. The attachment contains more detail associated with the action. The CRS will also determine what actions are necessary to progress to a follow-on step. The use of the attachment does not require completion prior to progressing in the body of the EOP or AOP.

D. Concurrent Use of Procedures

1. The Control Room Supervisor shall control the number of procedures being performed concurrently based upon the availability of resources and the ability to coordinate parallel activities.

2. During the performance of EOP immediate actions steps other plant operating procedures shall not be performed concurrently, except for those steps that must be performed immediately in response to actual plant conditions.

3. In no case will the concurrent performance of non-operating procedures be allowed to interfere with or impede the execution of EOP's and AOP's.

4. During the performance of concurrent procedures the operator shall inform the CRS and the crew prior to taking any critical action (i.e., any action that impacts the ability of the CRS to maintain command and control over the transient or the ability of the crew to maintain control of the plant).

5. If one or more procedure is being performed concurrently with an EOP and a procedure conflict arises the EOP should be accorded precedence in most cases. However, the CRS shall exercise authority in deciding each specific case.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/28/15	Tier #		3
Change: 0	Group/Category #		2
	K/A #	2.2.17	
Level of Difficulty: 2	Importance Rating		3.8

Equipment Control: Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

Question: 96

As stated in SO-G-87, Non-Routine Activities Requiring Formalized Plans, who approves Formalized Plans that fall into the "High Risk" category?

A. Only Plant Manager-FCS.

B. Manager-Operations and Shift Manager.

C. Plant Manager-FCS and Shift Manager.

D. System Engineer assigned to the activity.

Answer: B

K/A Match:
As the SRO, applicant must be familiar with Operations Management responsibilities necessary to control activities that require a heightened level of awareness.

Explanation:

A. Incorrect. Plausible if thought that only the Plant Manager approved Non-Routine Activities. Incorrect because independent of the risk category (low/medium/high/very high) the Shift Manager is always involved in approving formalized plans.
B. **Correct**. As stated in SO-G-87, Step 5.3.3, the Manager-Operations and Shift Manager's approval is required for activities that fall into the "High Risk" category.
C. Incorrect. Plausible because the Shift Manager's approval is required. Incorrect because the Manager-Operations is the other individual who approves formalized plans for High Risk activities.
D. Incorrect. Plausible because the System Engineer approves the formalized plan for low risk activities, and an engineering approval is required for High risk plans by the Manager of system Engineering.

Technical Reference: SO-G-87, Step 5.3, Rev. 16
(Attach if not previously provided including revision number)

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-1, Standing Orders-Licensed Operator
EO 1.0 - **STATE** the major sections of the Standing Orders.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:
55.41
55.43 5

SO-G-87
Non-Routine Activities Requiring Formalized Plans
Information Use
Page 10 of 17
Revision 16

5.3.3 High Risk

Operations	- Shift Manager (1) & Manager-Operations (1)
Engineering	- Manager-System Engineering
Other Departments	- Department Head

5.3.4 Very High Risk

Plant Review Committee (1)	
Operations	- Shift Manager (1) & Manager-Operations (1)
Engineering	- Manager-System Engineering
Other Departments	- Department Head

5.0 PROCEDURE

5.1 The Requestor of the activity shall determine whether the activity meets the definition of activities requiring a heightened level of awareness. (The Shift Manager may help with this determination and is ultimately responsible to ensure a formalized plan or approved procedure is used as appropriate.) If the activity does not require a heightened level of awareness, it may be performed following the guidance of SO-O-1. No formal plan is needed. However, if the activity does require a heightened level of awareness, a formalized plan is required, and then continue with this procedure. **[AR 12185]**

5.2 Develop a formalized plan for the activity that covers, as a minimum those items specified in Step 3.2. **[AR 12185]**

5.2.1 If time permits, develop alternative action plans to be performed based on the possible results obtained while performing the activity.

5.2.2 Perform walk-through or drills, where feasible, to validate plan assumptions (such as time-critical operations or maintenance activities).

5.3 Obtain written approval of the formalized plan by all affected departments. Even though these approvals are not required until actual start of the activity, it is recommended that these personnel be involved in the development stage of the activity. The approval level shall be based on the activity's potential impact on plant operations (Risk Assessment). Those individuals identified by (1) below shall approve all formalized plans at that risk level. If the activity will carry over to later shifts, the Shift Managers from those shifts should also approve the plan when possible. Attachment 1 may be used to document the risk level and approvals required.

i	**NOTE** Any of those responsible for approval of the plan may request an approved procedure or review by the Plant Review Committee.	i

5.3.1 Low Risk

Operations	- Shift Manager (1)
Engineering	- Engineer in System Engineering
Other Departments	- First Line Supervisor

5.3.2 Medium Risk

Operations	- Shift Manager (1)
Engineering	- Engineer in System Engineering
Other Departments	- First Line Supervisor

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		3
Change: 1	Group/Category #		2
	K/A #	2.2.38	
Level of Difficulty: 3	Importance Rating		4.5

Equipment Control: Knowledge of conditions and limitations in the facility license.

Question: 97

Given the following conditions:

- Plant is at 100% power.
- Outside (ambient) temperature is 101°F.
- The air conditioning unit for the East Switchgear Room has tripped on overload.
- Supplemental cooling using portable fans have failed to maintain temperature.
- East Switchgear Room temperature is 121°F and slowly rising.

Which of the following is required?

A. Maintain supplemental cooling per OI-VA-2, Auxiliary Building Ventilation System Normal Operation, and commence a shutdown only if Switchgear Room temperature reaches 125°F.

B. Declare both Safeguards Buses, 1A3 and 1A4, inoperable per Technical Specification LCO 2.7, and demonstrate operability of both Diesel Generators.

C. Commence a power reduction per AOP-05, Emergency Shutdown, and implement Technical Specification LCO 2.0.1 Required Actions.

D. Commence a power reduction per AOP-05, Emergency Shutdown, and implement Technical Specification LCO 2.7, Electrical Systems Required Actions.

Answer: C

K/A Match:

As the SRO, applicant must be familiar with conditions and limitations of the facility license.

Explanation:

A. Incorrect. Plausible because supplemental cooling must be maintained per OI-VA-2. Incorrect because switchgear and inverters are inoperable when room temperature reaches 120°F.
B. Incorrect. Plausible because both buses are inoperable. The action to verify operability of the diesel generators would be correct if only one bus was inoperable per TS 2.7.
C. **Correct**. Per OI-VA-2, Attachment 11, Temperature Limits Auxiliary Building Spaces, Technical Specification LCO 2.0.1 requires a plant shutdown because the switchgear and inverters are inoperable. Per LCO 2.0.1, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and less than 300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.
D. Incorrect. Plausible because Technical Specification LCO 2.7 addresses the equipment required for Electrical Systems. Incorrect because LCO 2.0.1 must be implemented.

Technical Reference: (Attach if not previously provided including revision number)
OI-VA-2, Attachment 11, Step 1 NOTE, Rev. 45
Technical Specification LCO 2.0.1, Amendment #283
ARP AI-187/A187, Window A-5

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-62-8, Technical Specifications-Licensed Operator
EO 10.0 - Given a copy of the Technical Specifications, **APPLY** the requirements to a given condition covered by the Administrative Controls.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content: 55.41 ___
55.43 1

OI-VA-2, Attachment 11, Step 1 NOTE

NOTE

1. The switchgear and inverters are inoperable if the Switchgear Room exceeds 120°F. Technical Specification 2.0.1 requires a Plant Shutdown if Switchgear or Electrical Penetration Room temperature exceeds 115°F.

   Worst case times for the Switchgear rooms to reach 120°F after a loss of all cooling at 100% power are (Ref FC06102)

   167 mins if the rooms are at 80°F when cooling is lost
   65 mins if the rooms are at 90°F when cooling is lost

   For Station Blackout, the time to reach 120°F is > 4 hours (Ref: FC06176)

2. The fans used for Section 1 are two "Heat Buster" Model HBD 3613 stored in the cage in Corridor 53. Use is supported by OpEval 14-015 associated with CR 2014-11223.

1. IF either of the following occur,
   THEN promptly perform SECTION 1.

   - Unplanned loss of all normal switchgear room air conditioning and ventilation with the Reactor above 300°F.
   - Switchgear room temperature cannot be maintained below 115°F

TECHNICAL SPECIFICATIONS

2.0 LIMITING CONDITIONS FOR OPERATION

2.0.1 General Requirements

Applicability

Applies to the operable status of all systems, subsystems, trains, components, or devices covered by the Limiting Conditions for Operation.

Objective

To specify corrective measures to be employed for system conditions not covered by or in excess of the Limiting Conditions for Operation.

Specification

(1) In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless corrective measures are completed that permit operation under the permissible action requirements for the specified time interval as measured from initial discovery or until the reactor is placed in an Operating Mode in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

Panel: **AI-187**	Annunciator: **A187**	Window: **A-4**

SWITCHGEAR ROOM A HIGH TEMPERATURE

Page 1 of 2

SAFETY RELATED

SWGR ROOM A HIGH TEMP

Tech Spec References: None

Initiating Device __TS-6605__ Setpoint _88ºF__ Power _MPP 1C3A-1_

OPERATOR ACTIONS

1.0 Check Switchgear Room "A" Air Conditioning Unit Equipment VA-87 (E Swgr) and VA-89 (outside south of TB) for proper operation.

1.1 IF VA-87, Switchgear Room “A” Air Handling Unit is not running, THEN attempt to restart VA-87 using handswitch OI/VA-87.

1.2 IF VA-87 restarts, THEN verify Switchgear temperature returns to normal.

2.0 IF VA-87 will not start, THEN perform the following to cross tie switchgear ventilation:

2.1 Verify VA-88, Switchgear Room “B” Air Handling Unit is in operation.

2.2 Verify Ventilation Dampers VA-91A, B and C are open.

2.2.1 IF VA-91A, B or C is not open, THEN notify the Control Room.

2.3 Check the breakers for VA-87 and VA-89 (Turbine Mezzanine):

- VA-87 MCC 3A4-B04
- VA-89 MCC 3A4-B05

2.4 IF either breaker is TRIPPED, THEN notify the Control Room of breaker trip.

(continue)

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 08/28/15	Tier #		3
Change: 0	Group/Category #		3
	K/A #	2.3.6	
Level of Difficulty: 3	Importance Rating		3.8

Radiation Control: Ability to approve release permits.

Question: 98

Given the following conditions:

- A Waste Gas Release is planned.
- FR-758, Stack Total Flowrate Recorder on AI-44, is NOT working.

Per the Offsite Dose Calculation Manual (ODCM), what additional actions, if any, are required to perform this release due to the FR-758, Stack Total Flowrate Recorder being inoperable?

A. Stack flow readings must be manually recorded on the gas discharge log at least every four hours.

B. The release is NOT allowed until FR-758 is repaired, has been recalibrated, and has passed a Functional Test.

C. NO additional actions are required as long as FR-532, Waste Gas Release Rate Recorder on AI-100, is OPERABLE.

D. Stack flow must be determined at least every four hours by multiplying the number of running Auxiliary Building Exhaust Fans by a value given in the ODCM.

Answer: A

K/A Match:

As the SRO, applicant must be familiar with the requirements of the Offsite Dose Calculation Manual.

Explanation:

A. **Correct**. Per the ODCM, a minimum of one operable channel is required for the Waste Gas Discharge Header (FR-532) and Auxiliary Building Stack (FR-758), therefore, this requirement is not met. Note that the question stem states that the flow recorder is not working but does not state that the flow measurement device is not working. The Required Action (Action 7) states that releases may continue provided the flow rate is estimated or recorded manually at least once per 4 hours during the release.

B. Incorrect. Plausible if thought that the flow recorder is required. Incorrect because the ODCM allows an exception to estimate the flow rate and manually record every 4 hours.

C. Incorrect. Plausible if thought that the recorder was sufficient to meet ODCM criteria. Incorrect because the flow rate measurement device needs to be OPERABLE. If not then sampling every 4 hours is required.

D. Incorrect. Plausible because a calculation must be made every 4 hours. Incorrect because the ODCM does not provide a value by which the number of exhaust fans can be multiplied to provide an estimated flow rate. ODCM, Table 3.2.1, Action 11, addresses Auxiliary Building Exhaust Fans in operation (page 18 of 130). Stack flow rate (3 Auxiliary Building Exhaust Fans) is also referenced on Page 74 of 130 for hi alarm setpoint calculation.

Technical Reference: ODCM, Section 3.2 & Table 3.2.1 , Rev. 24

(Attach if not previously provided including revision number)

Proposed references to be provided during examination: none

Lesson Plan / Learning Objective: Lesson Plan 19-50-04, Offsite Dose Calculation Manual-Licensed Operator EO 2.2 - **STATE** the action to be taken in the event liquid and gaseous Effluent Instrumentation is not operable.

Question Source: Bank # X

Modified Bank # ______ (Note changes or attach parent)

New ______

Question History: Last NRC Exam 2012 NRC Exam, Question #93

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 ______

55.43 4

3.2 Radioactive Gaseous Effluent Instrumentation

3.2.1 Limiting Condition for Operation

A. The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.2.1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Part II of the Off-Site Dose Calculation Manual.

APPLICABILITY: At all times

ACTION:

1. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the releases of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable.

2. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels operable, take the action shown in Table 3.2.1. Restore inoperable effluent monitoring instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radiological Effluent Release Report why this inoperability was not corrected in a timely manner. The reporting requirement is limited to the following instrumentation that monitors effluent streams: RM-057, RM-043, RM-062, RM-063, and RM-052.

Table 3.2.1 - Radioactive Gaseous Effluent Monitoring Instrumentation

Instrument	Minimum Channels Operable	Action
1. Auxiliary Bldg. Exhaust Stack (RM-052, RM-062)		
1.1 Noble Gas	1	1, 9, 11
1.2 Iodine and Particulate	1	2, 9, 11
2. Laboratory and Radwaste Processing Building Stack (RM-043)		
2.1 Noble Gas	1	3, 9
2.2 Iodine and Particulate	1	4, 9
3. Condenser Off Gas (RM-057)		
3.1 Noble Gas	1	5, 9
4. Containment Purge Line (RM-051, RM-052)		
4.1 Noble Gas	1	1, 6, 9, 11, 12
4.2 Iodine and Particulate	1	2, 9, 11, 12
5. Containment Pressure Relief Line (RM-051, RM-052)		
5.1 Noble Gas	1	1, 9, 11
5.2 Iodine and Particulate	1	2, 9, 11
6. Containment Penetrations M72 and M74 (Integrated Leak Rate Test Depressurization Vent Path)	N/A	10
7. Flow Rate Measurement Devices		
7.1 Waste Gas Discharge Header	1	7

Table 3.2.1 - Radioactive Gaseous Effluent Monitoring Instrumentation		
Instrument	**Minimum Channels Operable**	**Action**
7.2 Auxiliary Building Stack	1	7
7.3 Laboratory and Radwaste Processing Building Stack	1	7
7.4 Containment Purge Line	1	7
7.5 Containment Pressure Relief Line Annubar D/P	1	7
8. Radioactivity Chart Recorders		
8.1 Auxiliary Building Exhaust Stack	1	8

Table 3.2.1 Radioactive Gaseous Effluent Monitoring Instrumentation	
TABLE NOTATION	
ACTION 1	If the Auxiliary Building Exhaust Stack Noble Gas Monitor is inoperable, releases from the containment pressure relief line and the containment purge line are to be secured in the most expeditious manner. Ventilation of the auxiliary building via the Auxiliary Building Exhaust Stack may continue provided grab samples are taken once per 12 hours. (See Table 4.2)
ACTION 2	If the Auxiliary Building Exhaust Stack Iodine and Particulate Sampler is inoperable, ventilation of the Auxiliary Building and releases from the gaseous waste discharge header, containment pressure relief line or the containment purge line may continue through the Auxiliary Building Exhaust Stack provided sample collection in accordance with Table 4.2 using auxiliary sample collection equipment is initiated within 2 hours of the declaration of inoperability by the Shift Manager.
ACTION 3	If the Noble Gas Monitor is inoperable, ventilation of the LRWPB may continue via the LRWPB stack provided grab samples are taken at least once per 12 hours. (See Table 4.2)
ACTION 4	If the Iodine and Particulate Sampler is inoperable, ventilation of the LRWPB may continue via the LRWPB Stack provided sample collection using auxiliary sample collection equipment is initiated within 2 hours of the declaration of inoperability, by the Shift Manager, in accordance with Table 4.2.
ACTION 5	During power operation, when the condenser air ejector is in service, the condenser off gas discharge shall be monitored for gross radioactivity. If this monitor is inoperable, grab samples are taken at least once per 12 hours. (See Table 4.2)
ACTION 6	The release of airborne effluents from the Containment purge line will be secured if a noble gas monitor is unavailable to monitor the containment building atmosphere.
ACTION 7	With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided the flowrate is estimated or recorded manually at least once per four hours during the actual release.
ACTION 8	With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided the radioactivity level is recorded manually at least once per four hours during the actual release.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		3
Change: 1	Group/Category #		4
	K/A #	2.4.30	
Level of Difficulty: 3	Importance Rating		4.1

<u>Emergency Procedures/Plan</u>: Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

Question: 99

Given the following conditions:

- A loss of the 13.8 KV System has just occurred.
- ME-1, Weather Tower, has sustained a complete loss of power.

Which of the following notifications must the Shift Manager make within 30 minutes per SO-R-16, Weather Tower Problems?

Contact the __________ and generate a report per SO-R-16.

A. Nuclear Regulatory Commission (NRC)

B. Federal Aviation Administration (FAA)

C. Federal Communication Commission (FCC)

D. National Weather Service (NWS)

Answer: B

<u>K/A Match:</u>
As the SRO, applicant must be aware of required notifications to outside agencies upon equipment loss or failure.

Explanation:

A. Incorrect. Notification to a government agency (FAA) does not require the NRC to be notified pursuant to 10 CFR 50.72.b.2.xi. Plausible because this equipment failure is reportable pursuant to 10CFR50.72.b.3.xiii but it is an 8 hour report for loss of assessment capability.
B. **<u>Correct</u>**. Per OI-EG-4, Attachment 1, FCS is required to contact the FAA and file a report per SO-R-16, Weather Tower Problems.
C. Incorrect. Plausible because this would be correct if the tower had communication equipment.
D. Incorrect. Plausible because if an event is declared with the Weather Tower inoperable, the meteorological data for the Event Declaration Form (FC-1188) is obtained by calling the National Weather Service (directions are on Form FC-1188).

Technical Reference: (Attach if not previously provided including revision number)	OI-EG-4, Attachments 1 & 2, Step 1 NOTE & Step 2, Rev. 7
	FC-1188, Page 2 NOTE, Rev. 29
	SRO-R-16,Section 1, Rev. 11

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective:	Lesson Plan 7-12-16, Meteorological Systems-Licensed Operator EO 1.0 - **EXPLAIN** the principles of operation of the Meteorological System.
	Lesson Plan NRC Form 361, Emergency Event Notifications- LO EO 1.1 - **IDENTIFY** requirements for an Event Notification at Fort Calhoun Station.

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

Question History: Last NRC Exam None

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

10 CFR Part 55 Content:	55.41	
	55.43	5

NOTE

With a loss of 13.8KV, power will also be lost to Warehouse, TSC, Maintenance Shop, C&RP Building, Training Center, Administration Building, Weather Tower, temporary trailers, ISFSI Building, and ILRT equipment.

1. IF ERF alarm LOSS OF LIGHTS is received,
THEN contact the FAA AND generate a report per SO-R-16.

2. IF a complete loss of the ME-1, Weather Tower,
THEN complete the following:

a. Contact the FAA (402-294-3631) AND generate a report per SO-R-16.

b. Notify Shift Chemist to contact National Weather Service (1-800-425-9074, 402-359-4381) if data is needed.

FORT CALHOUN STATION
STANDING ORDER

SO-R-16
PAGE 2 OF 5

WEATHER TOWER PROBLEMS

1. **PURPOSE AND SCOPE**

1.1 Purpose

1.1.1 To provide guidance on how to report problems with the Weather Tower Lights.

1.1.2 Notify the Chemistry Department on the loss of meteorological data due to an unplanned loss of the Weather Tower.

1.2 Scope

1.2.1 This procedure will be used to report, within 30 minutes of light failure, to the FAA the failure or the repair of a light on the Weather Tower.

1.2.2 To provide guidance on contacting the Chemistry Department on the loss of meteorological data due to an unplanned loss of the Weather Tower.

2. **STATEMENT OF APPLICABILITY**

This Standing Order is applicable on any loss of Weather Tower lights or unplanned loss of meteorological data.

3. **DEFINITIONS**

None

4. **RESPONSIBILITIES**

The Shift Manager is responsible for ensuring the proper notifications are made.

FORT CALHOUN STATION
STANDING ORDER

SO-R-16
PAGE 4 OF 5

Attachment 7.1 - Loss of Weather Tower Lighting

NOTE: This notification to a government agency does not require the NRC to be notified pursuant to 10CFR50.72(b)(2)(xi).

1. Contact Lockheed Martin Outage Reporting and Notice to Airmen Line using the number provided in the Fort Calhoun Duty Assignment Call List.

2. You will then be given the choice to push “1” and talk to a Flight Service Specialist, which is what you want to do.

NOTE: Since the Weather Tower is not a communications tower, it does not have a FCC Registration Number.

FORT CALHOUN STATION
STANDING ORDER

SO-R-16
PAGE 5 OF 5

Attachment 7.2 - Unplanned Loss of Meteorological Data

1. Upon the discovery of the unplanned loss of meteorological data from the Weather Tower, inform the Shift Chemist of the need to derive synthetic data from the National Weather Service.

2. Upon restoration of meteorological data from the Weather Tower, inform the Shift Chemist to secure from deriving synthetic data.

FORT CALHOUN STATION
GENERAL FORM

FC-1188
R29
PAGE 2 OF 2

ENSURE that PAR information is given to the group within 15 minutes of event declaration, **DO NOT** delay relaying information if all members have not answered the COP.

ENSURE the following agencies are notified.

CONTACT any agency that has not answered the COP by using the number or alternate number listed in the Emergency Phone Book, after relaying the information contained on this form to the agencies that did answer the COP.

Notify the following agencies: (refer to Emergency Phone Book for alternate phone numbers)	✓	Name of contact (optional)
State of Iowa		
State of Nebraska		
Harrison County		
Pottawattamie County		
Washington County		

Record any comments, difficulties or observations you had while making this notification:

NOTE

If on-site meteorological data is not available, contact the National Weather Service (number in the Emergency Phone Book), and request wind speed and direction. For night time (sunset to sunrise), use a Δ T of +2.0 and a stability class F. For all other conditions, use a Δ T of -1.0 and a stability class D.

Examination Outline Cross-reference:	Level	RO	SRO
Rev. Date: 09/27/15	Tier #		3
Change: 1	Group/Category #		4
	K/A #	2.4.22	
Level of Difficulty: 2	Importance Rating		4.4

Emergency Procedures/Plan: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Question: 100

Given the following conditions:

- A Loss of Coolant Accident and Station Blackout are in progress.
- EOP-20, Functional Recovery, has been implemented.
- The Safety Function Status Checks must be performed.
- Resource Tree E, RCS and Core Heat Removal Safety Function, is being evaluated.

Which of the following describes the bases for how this Safety Function is evaluated?

The Safety Function can...

A. …be satisfied if any one of the Success Paths has ALL Steps satisfied.

B. …ONLY be satisfied if ALL Success Paths have ALL Steps satisfied.

C. …ONLY be satisfied if the required equipment for ALL Success Paths is available.

D. …be satisfied if the Shift Manager designates SELECTED steps in one or more of the Success Paths, and ALL of those selected steps are satisfied.

Answer: A

K/A Match:
As the SRO, applicant must be to explain the basis for how a Success Path is satisfied per the Emergency Operating Procedures. Knowledge of and navigation through the Resource Assessment Trees in the Functional Recovery procedure is SRO ONLY knowledge.

Explanation:

A. **Correct**. The requirement is for any of the Success Paths is to have all Steps satisfied.

B. Incorrect. Plausible if thought that the Safety Function Status Checks were designed this way, however, the requirement is for any of the Success Paths to have all Steps satisfied.

C. Incorrect. Plausible because each Success Paths identifies required equipment. Incorrect even though required equipment is available the Safety Function may not be satisfied.

D. Incorrect. Plausible if thought that the Shift Manager was allowed to make this determination.

Technical Reference: EOP-20, Resource Tree E, Rev. 28
(Attach if not previously provided including revision number)
OPD-4-09, Section 4.2, Rev. 20

Proposed references to be provided during examination: None

Lesson Plan / Learning Objective: Lesson Plan 7-18-18, Functional Recovery-Licensed Operator
EO 1.5 - Given the Resource Assessment Trees, basically **DESCRIBE** the Method, Path and Acceptance Criteria for each success path.
EO 1.6 - **EXPLAIN** how the Resource Assessment Trees are used in terms of Safety Function priority and success path priority within each tree.

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

Question History: Last NRC Exam None

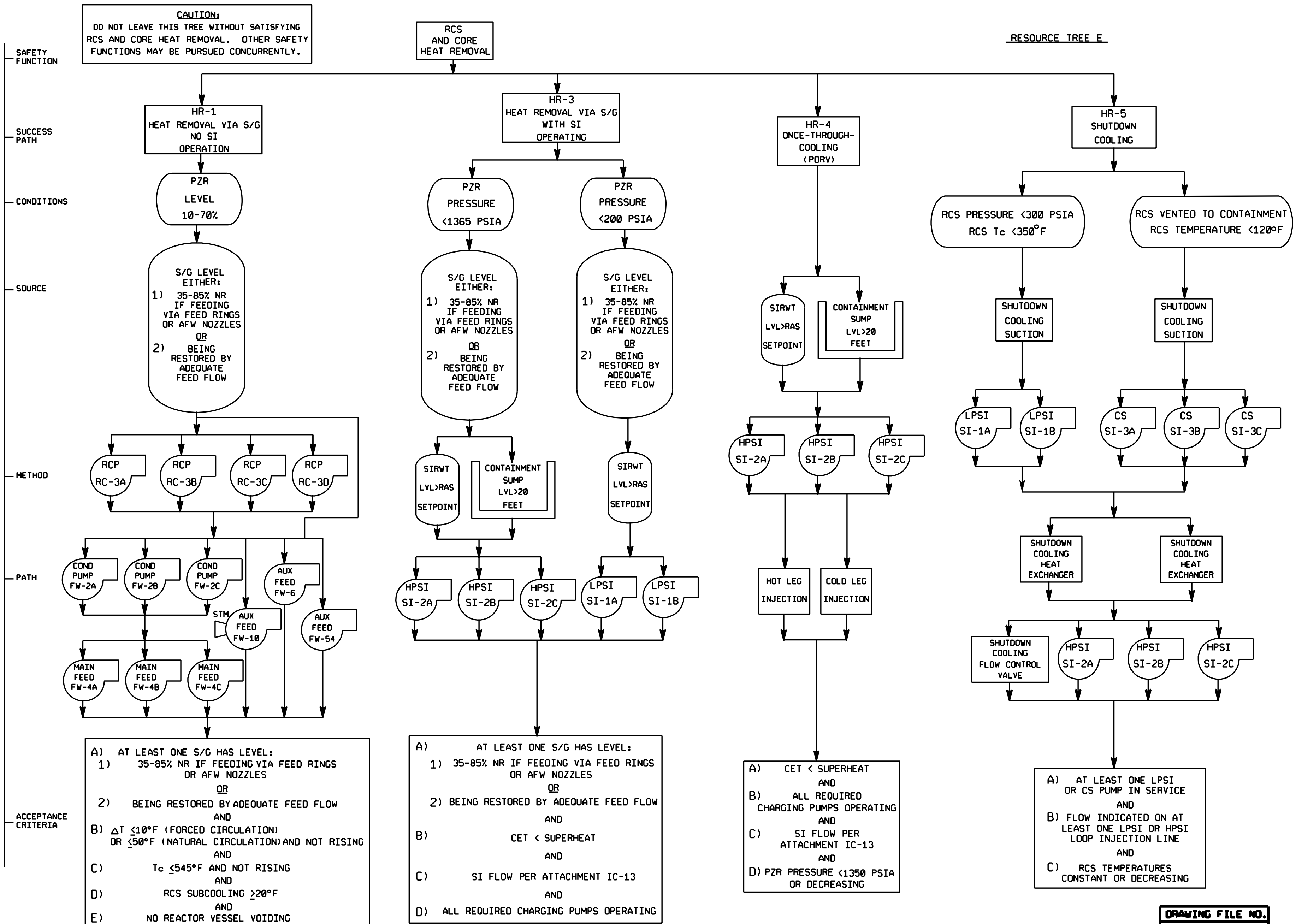
Question Cognitive Level: Memory or Fundamental Knowledge X
Comprehension or Analysis ____

10 CFR Part 55 Content: 55.41 ____
55.43 5

4.2 Safety Function

4.2.1 Because Safety Functions are a complete set of all the actions or conditions which will ensure public safety, they are used as the framework of all emergency guidance. In the ORPs, specific events such as LOCA or uncontrolled heat extraction are addressed. Because each event affects diverse parts of the plant, proper handling of these different events will emphasize different Safety Functions. For example, in a major LOCA, you would be most concerned about RCS pressure and inventory control in the short term. Therefore, the LOCA ORP actions are sequenced to achieve control of these two functions first, using equipment (success paths) designed for that purpose. Nonetheless, since all Safety Functions must be fulfilled to ensure public safety, each ORP must address all appropriate functions. In preparing EOPs the seven Safety functions were considered when developing the guidance to ensure that there were sufficient action steps to satisfy each affected Safety Function and that more than one success path for each affected function is mentioned in the guidance. Each ORP also includes a Safety Function status check chart which is used by the Control Room crew to continually determine whether the Safety Functions are being adequately maintained during the course of the event. The Safety Function status check is different for each ORP, and provides acceptance criteria which ensure that the Safety Function is being maintained by the success path most likely to be in use for that event.

4.2.2 The FRP is used by the operators when a diagnosis is not possible or when the ORP is not working (as determined by the Safety Function status check in each ORP). The FRP structure includes an expanded version of the Safety Function status check found in each ORP. This Safety Function status check provides acceptance criteria for all of the success paths which might fulfill the Safety Function. It is used by the operators to continually check the status of each Safety Function. For those Safety Functions which are found to be in jeopardy, a section of the FRP illustrates possible success paths for restoration of each Safety Function. Criteria which are used to judge successful Safety Function restoration are also provided for each success path. For the FRP, the Safety Functions actually form the main structure of the procedure. The procedure is divided into sections based on Safety Functions, and the sections are further divided to provide instructions on Safety Function restoration based on completion of individual success paths.



DRAWING FILE NO.
P-00142

FCS NRC Written Examination
Senior Reactor Operator
Answer Key

1.	C	26.	B	51.	A	76.	B
2.	D	27.	C	52.	A	77.	C
3.	A	28.	B	53.	C	78.	B
4.	C	29.	A	54.	D1*	79.	A
5.	B	30.	A	55.	B	80.	B
6.	B	31.	B	56.	B	81.	A
7.	D	32.	D	57.	D	82.	D
8.	A	33.	B	58.	A	83.	C *
9.	B	34.	B	59.	D	84.	C
10.	D	35.	C	60.	B	85.	D
11.	C	36.	A	61.	D	86.	D
12.	D	37.	C	62.	C	87.	A OR D
13.	C	38.	A	63.	D	88.	A
14.	B	39.	A	64.	C	89.	B
15.	A	40.	D	65.	A	90.	C
16.	C	41.	C	66.	D	91.	D
17.	C	42.	D	67.	B	92.	A
18.	B	43.	C	68.	C	93.	B
19.	A	44.	D	69.	A	94.	C
20.	D	45.	C	70.	C	95.	D
21.	A	46.	B	71.	D	96.	B
22.	C	47.	D	72.	B	97.	C
23.	D	48.	A	73.	D	98.	A
24.	D	49.	B	74.	B	99.	B
25.	D	50.	A	75.	A	100.	A

SRO Answer Breakdown:
A=25
B=25
C=24
D=26

* Question 83 deleted per Examination Report 2015-301

1* Question 54 deleted post examination on appeal

NRC RO/SRO Written Exam References

1. Generic Fundamentals Equation Sheet
2. Steam Tables

NRC RO/SRO Written Exam References

GENERIC FUNDAMENTALS EXAMINATION
EQUATIONS AND CONVERSIONS HANDOUT SHEET

EQUATIONS

$\dot{Q} = \dot{m}c_p\Delta T$

$\dot{Q} = \dot{m}\Delta h$

$\dot{Q} = UA\Delta T$

$\dot{Q} \propto \dot{m}^3_{Nat\ Circ}$

$\Delta T \propto \dot{m}^2_{Nat\ Circ}$

$K_{eff} = 1/(1 - \rho)$

$\rho = (K_{eff} - 1)/K_{eff}$

$SUR = 26.06/\tau$

$$\tau = \frac{\bar{\beta}_{eff} - \rho}{\lambda_{eff}\ \rho}$$

$$\rho = \frac{\ell^*}{\tau} + \frac{\bar{\beta}_{eff}}{1 + \lambda_{eff}\ \tau}$$

$\ell^* = 1 \times 10^{-4}$ sec

$\lambda_{eff} = 0.1\ sec^{-1}$ (for small positive ρ)

$DRW \propto \varphi^2_{tip}/\varphi^2_{avg}$

$P = P_o 10^{SUR(t)}$

$P = P_o e^{(t/\tau)}$

$A = A_o e^{-\lambda t}$

$CR_{S/D} = S/(1 - K_{eff})$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$1/M = CR_1/CR_X$

$A = \pi r^2$

$F = PA$

$\dot{m} = \rho A\vec{v}$

$\dot{W}_{Pump} = \dot{m}\Delta P\upsilon$

$E = IR$

Thermal Efficiency = Net Work Out/Energy In

$$\frac{g(z_2 - z_1)}{g_c} + \frac{(\vec{v}_2^{\,2} - \vec{v}_1^{\,2})}{2g_c} + \upsilon(P_2 - P_1) + (u_2 - u_1) + (q - w) = 0$$

$g_c = 32.2$ lbm-ft/lbf-sec^2

CONVERSIONS

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1 Mw	= 3.41×10^6 Btu/hr	1 Curie	= 3.7×10^{10} dps
1 hp	= 2.54×10^3 Btu/hr	1 kg	= 2.21 lbm
1 Btu	= 778 ft-lbf	1 gal_{water}	= 8.35 lbm
°C	= (5/9)(°F - 32)	1 ft^3_{water}	= 7.48 gal
°F	= (9/5)(°C) + 32		