DUKE ENERGY®

H.B. Robinson

ILC-13 NRC Licensing Exam
Written Exam
Book 2 of 2

ROBINSON

76. 008 AG2.1.7 SRO 001

Given the following plant conditions:

-Plant is at 100% RTP

-PCV-455C, PZR PORV, indicates open

-PT-444, CH I PRZR PRESS, is 2100 psig and lowering

-PT-445, CH II PRZR PRESS, is 2100 psig and lowering

-The OAC manually shuts PCV-455C

-I&C determines that the PCV-455C can't be controlled automatically

AOP-019, MALFUNCTION OF RCS PRESSURE CONTROL
AOP-025, RTGB INSTRUMENT FAILURE

Which ONE (1) of the following completes the statements below?

The crew has entered ___(1)___ . IAW ITS, PCV-455C ___(2)___ OPERABLE.

A. (1) AOP-019
(2) is **NOT**

B. (1) AOP-025
(2) is **NOT**

C.✓ (1) AOP-019
(2) is

D. (1) AOP-025
(2) is

The correct answer is C.

A) Incorrect. AOP-019 is the correct procedure because the valve failed open. PCV-455C is operable because the valve can still be manually operated. Plausible if the student thinks that the valve is inoperable because it had to be manually shut or because automatic control is gone. The valve can still be operable without automatic control.

B) Incorrect. AOP-019 is the correct procedure because the valve failed open. Plausible because if instrument that inputs into PCV-455C caused it to open, AOP-025 would be the correct procedure to enter. PCV-455C is operable because the valve can still me manually operated. Plausible if the student thinks that the valve is inoperable because it had to be manually shut or because automatic control is gone. The valve can still be operable without automatic control.

C) Correct. AOP-019 is the correct procedure because the valve failed open. PCV-455C is operable because the valve can still me manually opened and closed.

D) Incorrect, AOP-019 is the correct procedure because the valve failed open. Plausible because if instrument that inputs into PCV-455C caused it to open, AOP-025 would be the correct procedure to enter. PCV-455C is operable because the valve can still me manually opened and closed.

Question: 76
Tier/Group: 1/1
K/A Importance Rating: RO 4.4 SRO 4.7
K/A: 008 Pressurizer Vapor Space Accident
G2.1.7: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
References: Sim/Plant design, AOP-019, LCO 3.4.11
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 16 of PZR/PRT, Objective 1 AOP-019
Question Source: New
Question History: New
Question Cognitive Level: High
10 CFR part 55: (CFR 41.5 / 43.5 / 45.12 / 45.13)

Meets the K/A because the student needs to evaluate plant conditions and determine if the PORV is inoperable. Determining if the PORV is inoperable makes this an SRO question because the SRO must have knowledge of TS bases that is required to analyze TS required actions and terminology.

Correct

Purpose and Entry Conditions

(Page 1 of 1)

1. PURPOSE

This procedure provides instructions in the event RCS pressure is higher OR lower than required for current plant conditions.

This procedure is applicable in Modes 1, 2, and 3.

2. ENTRY CONDITIONS

This procedure may be entered when RCS pressure deviates from the desired control band due to a fault in pressure control components. (AOP-025 covers Instrument Failure)

- END -

Incorrect

Purpose & Entry Conditions

(Page 1 of 1)

1. **PURPOSE**

This procedure provides instructions for failure of process variable transmitters which provide input to RTGB controllers.

IF an applicable transmitter fails while the controller is operating in manual OR is being fed from an alternate channel, THEN entry to this procedure is NOT required.

This procedure is applicable in Modes 1, 2, 3, and 4.

2. **ENTRY CONDITIONS**

Failure of any process variable transmitter which affects automatic operation of RTGB controllers with the following exceptions:

- FT-605, RHR Flow
- LT-115, VCT Level
- LT-112, VCT Level
- PR NIS (NI-41, 42, 43, & 44)
- S/G Narrow Range Level

- END -

BASES

BACKGROUND (continued)

the PORVs minimize challenges to the pressurizer safety valves and also may be used for low temperature overpressure protection (LTOP). See LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

APPLICABLE SAFETY ANALYSES

The PORVs and their respective block valves are provided for plant operational flexibility and for limiting the number of challenges to the pressurizer safety valves. Operation of the PORVs is not explicitly considered to be a safety-related function for overpressure protection of the reactor coolant pressure boundary (RCPB) at normal operating temperature and pressure. Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. Operation of the PORVs in MODES 1, 2, and 3 is not classified as a safety-related function (i.e., one on which the results and conclusions of the safety analysis are based and that invokes the highest level of quality and construction). Also, an inadvertent opening of a PORV or a safety valve has been analyzed in the UFSAR (Ref. 1) as an anticipated operational occurrence (AOO) with acceptable consequences. For these reasons, the PORVs are not classified as safety related components.

Generic Letter 90-06 (Ref. 2) provided the NRC's resolution of PORV and block valve reliability concerns (Generic Issue 70), and set forth certain requirements to enhance safety. The pressurizer PORVs have no safety function and are not assumed to function during any UFSAR design basis accident or transient analysis. However, inclusion of the pressurizer PORVs is consistent with the guidance provided in Generic Letter 90-06. Therefore, they are being retained in Technical Specifications.

LCO

The LCO requires the PORVs and their associated block valves to be OPERABLE for manual operation.

An OPERABLE PORV is required to be capable of manually opening and closing, and not experiencing excessive seat leakage. Automatic control functions are not required for OPERABILITY of the PORVs.

(continued)

77. 054 AG2.4.35 SRO 001

Given the following plant conditions:

-The plant is at 98% RTP

- AOP-010 is entered due a leak in FW Heater 4B

-OPS Manager has chosen to repair the leak on line

AOP-010, MAIN FEEDWATER/CONDENSATE MALFUNCTION
OP-407, HEATER DRAINS AND VENTS

Which ONE (1) of the following completes the statements below?

The CRS will direct the OAO to use ___(1)___ for the specific steps to remove the required FW Heaters from service. The crew ___(2)___ have to reduce power for this evolution.

A. (1) OP-407
(2) will **NOT**

B. (1) AOP-010
(2) will **NOT**

C.✓ (1) OP-407
(2) will

D. (1) AOP-010
(2) will

The correct answer is C

A) Incorrect. OP-407 is correct. AOP-010 will direct you to OP-407 to remove the necessary string of heaters from service, the OAO will perform these actions. Will not have to reduce power is incorrect. Plausible because 98% power is a common power that we reduce to when performing evolutions dealing with AFW flow. The crew will have to reduce power to 659 MW.

B) Incorrect. AOP-010 is incorrect. There are several AOP's that direct performing specific actions either in the main body or in an attachment. This is not one of them. AOP-010 directs you to OP-407 to remove the heaters. Will not have to reduce power is incorrect. Plausible because 98% power is a common power that we reduce to when performing evolutions dealing with AFW flow. The crew will have to reduce power to 659 MW.

C) Correct. OP-407 is correct. There are several AOP's that direct performing specific actions either in the main body or in an attachment. This is not one of them. AOP-010 directs you to OP-407 to remove the heaters. The crew will have to reduce power to 659 MW.

D) Incorrect. AOP-010 is incorrect. There are several AOP's that direct performing specific actions either in the main body or in an attachment. This is not one of them. AOP-010 directs you to OP-407 to remove the heaters. The crew will have to reduce power to 659 MW is correct.

Question: 77
Tier/Group: 1/1
K/A Importance Rating: RO 3.8 SRO 4.0
K/A: 054 Loss of Main Feedwater
G 2.4.35: Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
Reference(s): AOP-010, OP-407
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 4 of AOP-010 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)
Comments:

This question meets the K/A because the candidate has to determine which feedwater heaters need to be removed (operational implication), and what to send the OAO to look at(Local AO tasks). It is SRO level because the candidate must determine which procedure to use to remove the feed water heaters.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 1 is immediate action step.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Feedwater Regulating Valves - OPERATING PROPERLY (MANUAL OR AUTO):** • FCV-478, FRV "A" • FCV-488, FRV "B" • FCV-498, FRV "C"	Perform the following: a. Verify FRV for affected S/G(s) in manual control. b. Attempt to stabilize S/G level using FRV and/or FRV Bypass Valves by matching steam flow with feed flow. c. Stop any load change in progress. d. Restore affected S/G level to between 39% and 52%. e. IF unable to control S/G level, THEN trip the Reactor AND Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION. f. Go To Step 37
* 2.	**Check Reactor Trip Setpoint - BEING APPROACHED**	IF a Reactor Trip Setpoint is approached, THEN trip the Reactor and Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION. Go To Step 4.
3.	**Trip The Reactor And Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION.**	
4.	**Make PA Announcement For Procedure Entry**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	Go To The Appropriate Step from The Table Below:	

EVENT	STEP
Main Feed Pump Trip	Step 6
Condensate AND Feed Pump Trip	Step 10
Condensate Pump Trip Without MFP Trip	Step 48
Heater Drain Pump Trip	Step 15
Pipe Break or Leak Feedwater Heater Leak	Step 21
HCV-1459 Failed Open	Step 35
Other	Step 24

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Check Reactor Power - LESS THAN 70%**	Trip the Reactor and Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION.
7.	**Check Reactor Power - GREATER THAN 60%**	Go To Step 13.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	NOTE Rapid power reductions may result in the axial flux difference exceeding the operating band values and require a power reduction to less than 50% to comply with ITS 3.2.3 Condition C.	
*21.	**Reduce Turbine Load At 1%/MIN To 5%/MIN To Match Feedwater Flow <u>AND</u> Steam Flows As Follows:**	
	a. Verify Rods in AUTOMATIC	a. Place ROD BANK SELECTOR to M (Manual) <u>AND</u> manually position control rods to maintain Tavg within -1.5 to +1.5°F of Tref
	b. Check Turbine Control Mode - AUTOMATIC 1) Depress the IMP IN Pushbutton 2) Set The Desired Load In The SETTER 3) Set The Desired Load Rate 4) Depress the GO Pushbutton or the HOLD Pushbutton as Necessary to Reduce Turbine Load	b. Momentarily depress the GV (down) button on the EH Control Panel as needed to lower Turbine Load
	c. Borate Per OP-301, RCS Boration Quick Checklist, as necessary to maintain AFD within the operating band	
	d. Check Feed Flow matched with Steam Flow	d. Continue with power reduction. <u>WHEN</u> flows are matched perform Step 21.e. Go To Step 22.
	e. Stop the power reduction	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

OP-407 provides instructions for removing Feed Water Heaters from service.

22. **Perform the following:**

a. Determine location of the leak

- Visual observation of an external leak
- Feed Water Heater level alarms
- Feed Water Heater Normal and Alternate drain valve positions
- Feed Water Heater #1 & #2 Emergency Dump valve positions

b. Isolate the leak.

RESPONSE NOT OBTAINED:

b. Determine appropriate strategy:

1) Consult with Operations Management to choose appropriate strategy

- Repair on line
- Shutdown to repair
- Trip Unit

2) IF the leak will be repaired on line, THEN observe the NOTE prior to Step 38 And Go To Step 38.

3) IF the Unit will be shutdown, THEN Go To GP-006-01, Normal Plant Shutdown From Power Operation To Hot Shutdown.

4) IF the Unit will be tripped, THEN trip the Reactor and Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION.

c. Notify the SM of leak location and method used to isolate leak.

23. **Observe The NOTE Prior To Step 38 And Go To Step 38**

78. 055 EA2.06 SRO 001

Given the following plant conditions:

-A Station Blackout has occurred from 100% RTP

-The crew is performing actions of EPP-1, LOSS OF ALL AC POWER

-Immediate actions are complete

-"A" Charging pump is running

-Offsite power is available to re-energize the SUT

-APP-009-B5, MAIN TRANSF PHASE TRIP, alarm is in

OP-603, ELECTRICAL DISTRIBUTION
OP-603-3, RESETTING HIGH IMPEDANCE FAULT TRIPS
86P, GENERATOR LOCKOUT RELAY, PRIMARY
86BU, GENERATOR LOCKOUT RELAY, BACKUP

Which ONE (1) of the following completes the statement below?

IAW EPP-1, the CRS will direct the use of OP-___(1)___ to restore power. APP-009-B5 is an indication that the ___(2)___ lockout needs to be reset.

A.✓ (1) 603
(2) 86P

B. (1) 603-3
(2) 86P

C. (1) 603
(2) 86 BU

D. (1) 603-3
(2) 86 BU

The correct answer is A

A) Correct. OP-603 will provide the guidance to restore power electrical power. 86P will need to be reset based of APP-009-B5. MAIN TRANSF PHASE TRIP feeds into the 86P which in turn, trips the turbine.

B) Incorrect. OP-603-3 is incorrect. Plausible since this procedure does give guidance on restoring power, however, it is restoring power to the DS Bus which has already happened. This is indicated by the "A" Charging pump running. 86P will need to be reset based of APP-009-B5. MAIN TRANSF PHASE TRIP feeds into the 86P which in turn, trips the turbine.

C) Incorrect. OP-603 will provide the guidance to restore power electrical power. 86BU is incorrect. Plausible since this is one of the two lockouts that need to be reset to restore power, however, this alarm is not associated with 86BU.

D) Incorrect. OP-603-3 is incorrect. Plausible since this procedure does give guidance on restoring power, however, it is restoring power to the DS Bus which has already happened. This is indicated by the "A" Charging pump running. 86BU is incorrect. Plausible since this is one of the two lockouts that need to be reset to restore power, however, this alarm is not associated with 86BU.

Question: 78
Tier/Group: 1/1
K/A Importance Rating: RO 3.7 SRO 4.1
K/A: 055 Loss of Offsite and Onsite Power (Station Blackout)
EA2: Ability to determine or interpret the following as they apply to a Station Blackout:
EA2.06: Faults and lockouts that must be cleared prior to re- energizing buses
Reference(s): Sim/Plant design, EPP-1, OP-603
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 4 of AOP-010 lesson plan
Question Source: RNP Bank
Question History: 2009 NRC Exam, Changed format and changed the DS powered pump that was running
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 43.5 / 45.13)
Comments:

This question meets the K/A becuase it requires the student to interpret which lockout needs to be reset in order to restore power from offsite. Since the student has to determine which procedure to use in order to restore power, this makes it an SRO question.

Correct

ALARM

MAIN TRANSF FAULT TRIP

AUTOMATIC ACTIONS

1. Generator Lockout (See Attachment 1) (2/3 logic)

CAUSE

1. Internal electrical fault in Main Transformer "A", "B", or "C" (Fault Pressure)
2. APP-045 Annunciator Panel TEST pushbutton.

OBSERVATIONS

1. Exciter Field Breaker Indication
2. Generator Output Breaker Position (52/8 and 52/9)

ACTIONS	CK (✓)
1. **IF** the reactor has tripped, **THEN REFER TO** EOP Network.	_____
2. **IF** the Turbine has tripped while below 40% power, **THEN REFER TO** AOP-007.	_____
3. **IF** a fault has occurred on one of the Main Transformers, **THEN CONTACT** the Load Dispatcher for repairs. (SOER 10-1)	_____

DEVICE/SETPOINTS

1. 63FP / Sensitivity Selection Position Zero (0)

POSSIBLE PLANT EFFECTS

1. Reactor Trip
2. Turbine Trip

REFERENCES

1. ITS SR 3.8.1.16
2. EOP Network
3. AOP-007, Turbine Trip Below P-8
4. CWD B-190628, Sheet 940, Cable P
5. SOER 10-1, Large Power Transformer Reliability, recommendation 6

ATTACHMENT 1
LOCKOUT AUTO ACTIONS

correct

Generator Lockout

1. 4KV Fast Bus Transfer
2. Trips Generator Output OCBs 52/8 and 52/9
3. Trips Exciter Field Breaker
4. Actuates Turbine Trip (20/AST and 20/ET)

Startup Transformer Lockout

1. Trips 4KV Breakers 52/12 and 52/17
2. Opens 115KV Motor Operated Disconnect
3. Trips UNIT NO. 2 START-UP TRANSF WEST BUS 115 KV OCB
4. Trips UNIT NO. 2 START-UP TRANSF EAST BUS 115 KV OCB

GENERATOR LOCKOUT RELAY INPUT SIGNALS

86P	86BU	BOTH
GEN GROUND TRIP (APP-009-B4)	GEN NEG SEQ/OCB BU TRIP (APP-009-C2)	OCB 52-8 FAILED TO OPEN (APP-009-D5)
MAIN TRANSF FAULT TRIP (APP-009-B5)	Exciter Field Breaker Tripped (Loss of field)	OCB 52-9 FAILED TO OPEN (APP-009-D6)
AUX TRANSF OVLD/PHASE Δ TRIP (APP-009-A6)	Unit Differential (87/GT) (APP-009-A5)	
AUX TRANSF FAULT TRIP (APP-009-B6)	Turbine Trip (Stop valves or Governor valves)	
Voltage Control Overcurrent Relay (51-27)		
Generator Differential (87/G)		
Turbine Trip (63AST)		

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*16.	**Perform The Following:**	
	a. Request assistance from Maintenance in restoring AC Power	
	b. Dispatch an Operator to complete EPP-22, ENERGIZING PLANT EQUIPMENT USING DEDICATED SHUTDOWN DIESEL GENERATOR	
	c. Contact the Load Dispatcher to determine when Offsite Power will become available	
	d. Check Off-Site Power (Grid) - AVAILABLE	d. WHEN Off-Site Power is available, THEN perform Step 16.e Observe the NOTE prior to Step 17 and Go To Step 17.
	e. Check Startup Transformer - AVAILABLE	e. IF the problem is associated with the Startup Transformer THEN perform the following: • IF personnel are available, THEN restore power using backfeed in EPP-25, ENERGIZING SUPPLEMENTAL PLANT EQUIPMENT USING THE DSDG. OR • IF ERO facilities are activated, THEN request a mission to restore power using backfeed in EPP-25, ENERGIZING SUPPLEMENTAL PLANT EQUIPMENT USING THE DSDG. Observe the NOTE prior to Step 17 and Go To Step 17.
	f. Restore normal power using OP-603	

Incorrect

TABLE OF CONTENTS

incorrect

1.0 PURPOSE

1. To provide instructions for safely restoring an electrical bus following a high impedance fault trip.

2.0 SCOPE

1. To provide instructions for inspecting busses following a fault trip, determining if it is safe to energize that bus, and remove nonessential loads prior to energizing. Loads required for safe shutdown are loaded after bus has been re-energized. The remaining loads will remain de-energized until full troubleshooting and repair are completed.

3.0 PRECAUTIONS AND LIMITATIONS

1. Resetting of the Generator Lockout Relays 86P and/or 86BU should be done only after all applicable Generator Lockout Signals have been reset and/or removed. Attempting to reset either 86P or 86BU with a Lockout signal present will reactivate the fast bus transfer sequence and may cause a loss of 4KV busses. [Section 7.1.1 Commitment 3] (CR 390095; OPS-NGGC-1000, Resetting Protective Devices)

2. Protection devices (breakers, fuses, bistables, overloads, lockouts, and so forth) which have tripped should only be reset, with CRS approval, under the following conditions:

    a. The cause of the trip has been identified and corrected.

    b. Where no evidence of abnormality is present, it is permissible to restore the protective device ONE TIME (see below on MOVs). Major control schemes (such as the 86 Generator Lockout scheme) SHALL **NOT** be reset without understanding the cause of the event **AND** required corrective action(s), unless it is determined that the action is required to support public health and safety concerns.

3. The SM may approve additional protective device resetting after consultation with engineering. Reenergizing busses without a proper investigation can lead to equipment damage, personnel injury or death.

79. 058 AG2.1.19 SRO 001

Given the following plant conditions:

-The reactor is at 100% RTP

-"B" Charging pump is running in Manual

-"C" Charging pump is running in AUTO

-A reactor trip occurs

-The following indications are seen:
-See next page for references

Which ONE (1) of the following completes the statements below?

Based off the indications above, "C" Charging pump is __(1)__ . The crew will transition from EOP-E-0 to __(2)__ to EPP-27.

A.✓ (1) running
(2) EOP-ES-0.1

B. (1) tripped
(2) EOP-ES-0.1

C. (1) running
(2) EPP-7

D. (1) tripped
(2) EPP-7

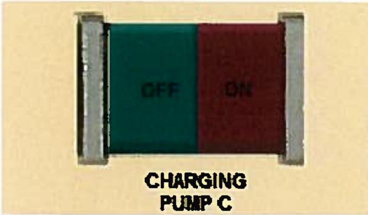
The correct answer is A

A) Correct. "C" Charging pump is still running, however, it has lost control power. The crew will transition from EOP-E-0 to EOP-ES-0.1 to EPP-27.

B) Incorrect. "C" Charging pump is still running, however, it has lost control power. Plausible since the indication for "C" Charging pump goes away during a loss of "B" DC. The crew will transition from EOP-E-0 to EOP-ES-0.1 to EPP-27.

C) Incorrect. "C" Charging pump is still running, however, it has lost control power. EPP-7 to EPP-27 is incorrect. Plausible since if the students believe that a loss of DC gives you an SI, you would go from EOP-E-0 to EPP-7 to EPP-27.

D) Incorrect. "C" Charging pump is still running, however, it has lost control power. Plausible since the indication for "C" Charging pump goes away during a loss of "B" DC. EPP-7 to EPP-27 is incorrect. Plausible since if the students believe that a loss of DC gives you an SI, you would go from EOP-E-0 to EPP-7 to EPP-27.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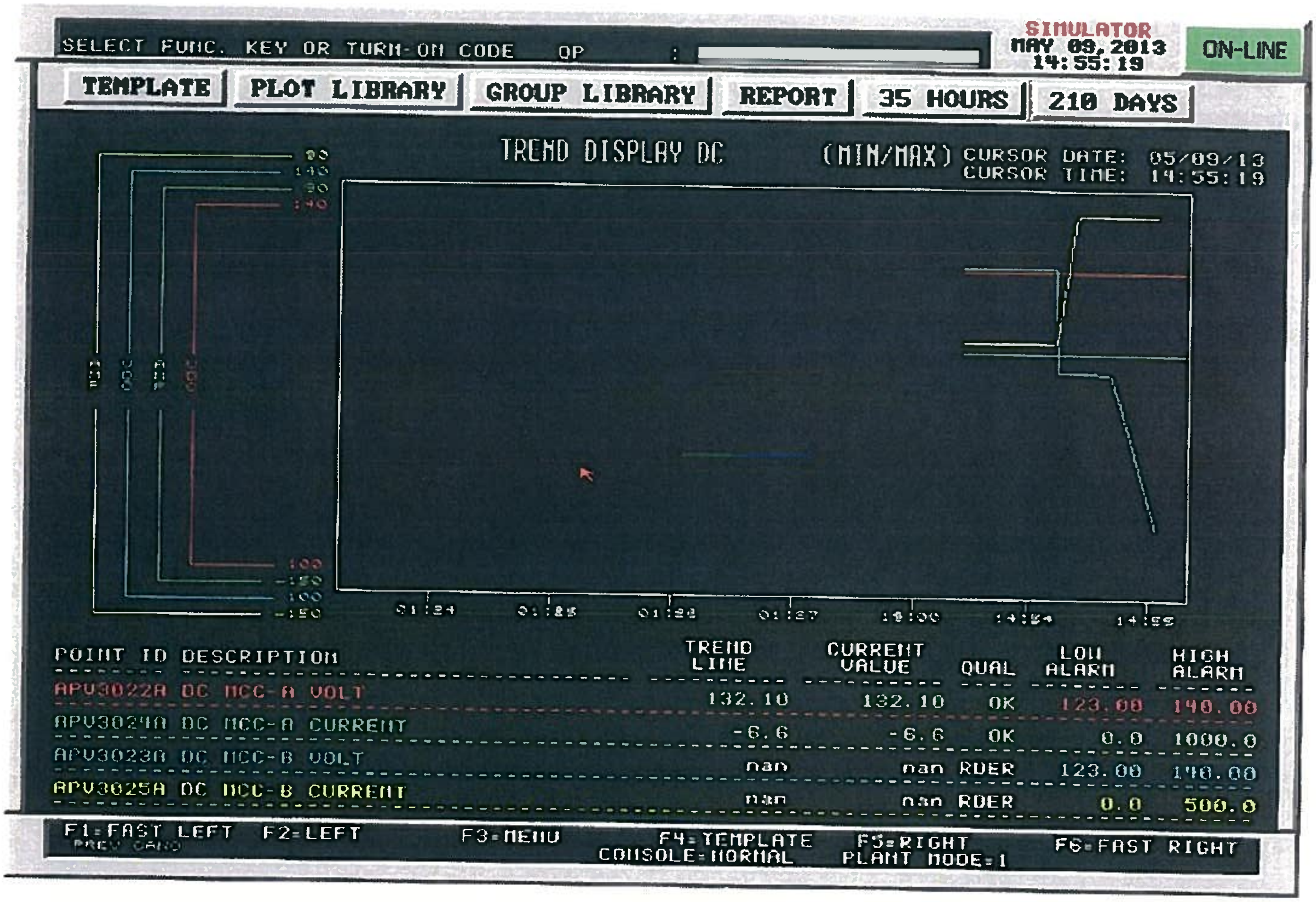
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SIMULATOR
MAY 09, 2013
14:55:19
ON-LINE
TEMPLATE
PLOT LIBRARY
GROUP LIBRARY
REPORT
35 HOURS
210 DAYS
TREND DISPLAY DC
(MIN/MAX) CURSOR DATE: 05/09/13
CURSOR TIME: 14:55:19
POINT ID DESCRIPTION
TREND LINE
CURRENT VALUE
QUAL
LOW ALARM
HIGH ALARM
APV3022A DC MCC-A VOLT 132.10 132.10 OK 123.00 140.00
APV3024A DC MCC-A CURRENT -6.6 -6.6 OK 0.0 1000.0
APV3023A DC MCC-B VOLT nan nan RDER 123.00 140.00
APV3025A DC MCC-B CURRENT nan nan RDER 0.0 500.0
F1=FAST LEFT
F2=LEFT
F3=MENU
F4=TEMPLATE
CONSOLE=NORMAL
F5=RIGHT
PLANT MODE=1
F6=FAST RIGHT

Question: 79
Tier/Group: 1/1
K/A Importance Rating: RO 3.9 SRO 3.8
K/A: 058 Loss of DC Power
AG2.1.19: Ability to use plant computers to evaluate system or component status.
Reference(s): Sim/Plant design, EPP-27
Proposed References to be provided to applicants during examination: Picture of ERFIS printout and "C" Charging pump status.
Learning Objective: Objective 4 of AOP-010 lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.10 / 45.12)
Comments:

This question meets the K/A because the student must use the ERFIS(plant computer) to determine they have lost DC power. From that, the student will determine the status of "C" Charging pump and then determine the procedure flow paths. Determining which procedures to use is what makes this question an SRO level question.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**Maintain Stable Plant Conditions:**	
	a. PZR pressure - AT 2235 PSIG	
	b. PZR level - AT 22%	
	c. S/G narrow range levels - BETWEEN 8% AND 50%	
	d. RCS temperature: • With any RCP running, RCS average temperature - STABLE AT OR TRENDING TO 547°F <u>OR</u> • With NO RCP running, RCS cold leg temperatures - STABLE AT OR TRENDING TO 547°F	
	e. Check DC busses A and B - ENERGIZED	e. Perform the following: • If DC bus A is <u>NOT</u> energized, <u>THEN</u> perform EPP-26, LOSS OF DC BUS "A", while continuing with this procedure. • If DC bus B is <u>NOT</u> energized, <u>THEN</u> perform EPP-27, LOSS OF DC BUS "B", while continuing with this procedure.
12.	**Perform Attachment 8, Aligning Balance Of Plant, As Time Permits While Continuing With This Procedure**	

Correct

INFORMATION USE

ATTACHMENT 1

MAJOR EFFECTS / LOAD LIST

(Page 2 of 4)

Major DC Loads Lost:

DC Control Power for 4KV Busses 3 & 4, 480V Busses 2B, 3, & 4 and 480V Emergency Bus E-2

"C" Charging Pumps Gets control Power From E2.

Inverter B

Reactor Protection and Safeguards Train B

Steam Dump System: Steam Dumps PRV-1324 A-2 & B-3 fail closed.

Steam Driven AFW Pump Control Power

125VDC MCCs B & B-A, Distribution Panels B & B1, and Power Panel No. 25

PCV-455C, PZR PORV

FIC-626, FCV-626 High Flow Closing Controls will NOT allow the valve to remain open while the valve is energized.

Aux Panel GC Fuse Panel:

Component	Function	Position	Ckt
FIC-626	FCV-626 High Flow Controls	Closed	01
CVC-303B	RCP B Seal Return	Open	19
CVC-307	RCP Seal Leakoff Bypass	Closed	21
RC-522A	PW to RCP A Standpipe	Closed	14
RC-519A	PW to PRT	Closed	24
RC-519B	PW to PRT	Closed	24
RC-519C	PW to PRT	Closed	20
RC-523	PRT to RCDT Pump	Closed	18
RC-550	PRT N_2 Supply Isolation	Closed	47
RC-544	RV Flange Leak Detection	Open	16
RC-568	RV Head Vent	Closed	52
RC-570	PZR Head Vent	Closed	52
RC-572	CV Atmos Solenoid Isolation	Closed	52

Correct

Since an SI does not occur on a loss of DC Bus B, entry to the procedure will be via EOP-ES-0.1, Reactor Trip. Certain actions will be completed while in EOP-ES-0.1, Attachment 7. These actions are:

a. Transfer of Instrument Bus 3 to MCC-8.

b. Shutdown of Emergency Diesel B.

STEP SPECIFIC DESCRIPTION AND RNP DIFFERENCES

The following pages will provide the RNP step number and the STEP basis for each step where applicable. This is a Robinson specific EOP, therefore there is no corresponding ERG series of steps. This procedure covers an event that is not covered by the ERGs (Loss of DC). The entire procedure may be categorized as an SSD 10. The steps within this procedure will not interfere with performance of the EOPs since this procedure does not consider any other event in progress other than a loss of DC Bus "B". The loss of a DC Bus at RNP is considered "beyond design basis" and is not analyzed in the UFSAR.

RNP STEP	BASIS
1	STEP BASIS

This step provides transitional direction for the subsequent step. If the Loss of DC occurred from an at power condition, the main generator output Circuit Breakers, 52/8 and 52/9, will be closed and action will be necessary to trip them.

If the event started from a low power or shutdown condition the subsequent step will not be necessary.

2 STEP BASIS

On a loss of DC Bus B the North and South Generator Circuit Breakers, 52/8 and 52/9, will receive a lockout signal. Due to the loss of DC these breakers will not open. This in turn causes backup relaying to open other breakers to isolate the generator. In order to accomplish actions later in the procedure and to allow reclosing the backup Circuit Breakers these breakers must be opened.

There are no local controls that will open the breakers without control power. There is, however, a maintenance control (for testing) at each phase of the Circuit Breakers. This feature will trip the Circuit Breakers one phase at a time. Since this function was not intended to be performed by Site Personnel the Load Dispatcher will be notified to request assistance in opening the breakers.

3 STEP BASIS

This continuous action step is provided to initiate efforts to repair the faulted DC Bus. It is placed early in the procedure so that efforts can be made to contact Maintenance personnel.

The high level step provides direction to diagnose the cause and provides transitional guidance. There are three possible failure mechanisms that are the most likely causes:

- Fault on B Battery
- Fault on B Battery Bus
- Fault on MCC-6

The failure, or tripping, of the in-service Battery Charger, is not a likely cause of the loss of DC since warning would be provided via an annunciator with ample time for Operator action to transfer the Chargers.

N4 STEP BASIS

The note reminds the Operator that AFW Pump B will not be available due to a loss of Control Power.

4 STEP BASIS

This step assures the maintenance of the secondary heat sink by maintaining S/G level at the standard range used throughout the EOP Network. In this case AFW Pump A and the SDAFW pump are specified since AFW Pump B is lost.

DISCUSSION

Incorrect

This Basis Document provides the step justification for a plant specific EPP. There is no ERG background for this procedure and no Safety Significant Deviation identification number is assigned for the steps since there are no corresponding ERG instructions.

The purpose of this procedure is to provide directions for combating conditions that arise from a loss of DC Bus A. The procedure is intended to handle situations arising from conditions in which the EPPs are applicable (> 350°F). This procedure assumes that no other casualty is in progress. Adjustment of the steps may be necessary if other events are in progress. A loss of DC is not an analyzed event at RNP and is not considered a credible event since a passive failure would be required to cause this event.

If the Reactor Trip Breakers are closed, there will always be a Rx Trip from a loss of power to the 52/RTA UV coil. EDG A will always auto-start, (loss of power to air start solenoids), but without control power, it cannot flash its field or close its output breaker. Since DC Bus A supplies Inverter A, Instrument Bus 2 and 7 will always be lost. However, Loss of DC Bus A has vastly different consequences depending on the initial AC electrical lineup. Initially, if at power, following the reactor and turbine trip, 4 KV and 480V Buses will remain energized as the turbine coasts down. Bus voltage will decrease as the turbine speed decreases. DC Bus A supplies Control Power to Breakers on 4KV Busses 1 & 2, 480V Busses 1 & 2A, and 480V Emergency Bus E-1. (480V Bus 2B is normally supplied from 4KV Bus 1, so it will follow the effects of Bus 1. However, its DC Control Power is supplied by Bus B so it will not lose protective relaying)

- If these busses were initially on the Startup Transformer (SUT) they will remain energized. However, all Busses except for 480V Bus 2B will lose DC Control Power.
- If the busses were initially on the Unit Aux Transformer (UAT), the resultant Rx/Turbine/Generator trip will attempt an auto-transfer, but without DC Control Power, this will not occur. The UAT will be deenergized along with all the busses and components it was supplying. The Loss of E-1 results in a loss of Instrument Bus 1. Since Instrument Bus 2 was lost from the Loss of DC Bus A, all bistables in both of these channels will fail and initiate an SI. (This will be a one-train SI since half the plant AC power is lost and the A train Sequencer is failed)

If the Reactor Trip Breakers were closed but all busses were still on the SUT, a Reactor Trip without SI will occur. If the Unit was at power and busses were on UAT, a Reactor Trip with SI will occur. Either way the EOP network will be entered. This EPP will be entered via PATH-1 and EPP-7 or EOP-ES-0.1. Certain actions necessary for a Loss of DC Bus A will be completed in EOP-E-0 or EOP-ES-0.1 attachments. These are actions that are performed to enable completion of certain steps in PATH-1 and steps needed to combat the loss of DC. These steps are:

- Alignment of makeup to the Charging Pump suction by bypassing LCV-115B (CVC-358 is opened). LCV-115B fails closed and since Letdown is isolated, a path of water must be aligned to the Charging Pumps.
- Instrument Bus 2 is transferred to MCC-8 in order to regain instrumentation to aid in diagnostics of PATH-1.
- The exciter field breaker is tripped locally to prevent further damage to the Generator and Exciter. As the Generator coasts down, the exciter will attempt to maintain voltage by increasing its output. Normal protection is not available because the control power to trip the Exciter Field Breaker is via DC Bus A.
- A EDG Fuel Racks are tripped to stop the engine. This is the fastest method of stopping the damage to the air start distributor. The EDG is running but can not be loaded because of the lack of control power to the Voltage Regulator.
- Instrument Air is isolated to the EDG to prevent start attempts and conserve air in the starting receiver.
- If MCC-5 is deenergized it is transferred to the DS Bus. If the loss of DC Bus A occurred from an at power condition MCC-5 will be lost. Transferring the Bus to the DS bus will regain Instrument Bus 1 and safety related loads, such as valve operators powered from MCC-5.

This procedure and EPP-27, for DC Bus "B" have been credited in the evaluation of INPO SOER 81-15, PARTIAL LOSS OF DC POWER, recommendation 2C. No specific steps or sections were identified in the evaluation.

80. 077 AA2.07 SRO 001

Given the following plant conditions:

-Plant is at 40% RTP

-The Load Dispatcher reports that grid voltage is degrading

-12:00:00 on 1/1/13, 480V Bus E2 voltage was reduced to 400V

-12:00:20 on 1/1/13, there were no changes to the electrical lineup

-12:30:00 on 1/1/13, grid voltage was restored

Which ONE (1) of the following completes the statements below?

The "B" EDG is designed to start under this condition when ___(1)___ degraded voltage relays sense their setpoint. If "B" EDG cannot be restored to service, the latest time the plant can be in MODE 3 is ___(2)___ IAW applicable LCO.

(REFERENCE PROVIDED)

A. (1) 1/2
(2) 18:00:20 on 1/7/13

B. (1) 1/2
(2) 18:00:20 on 1/8/13

C. (1) 2/3
(2) 18:00:20 on 1/7/13

D✓ (1) 2/3
(2) 18:00:20 on 1/8/13

The correct answer is D

A) Incorrect. 1/2 is incorrect. Plausible because this is the relay coincidence required for undervoltage, this is a degraded voltage condition. A degraded grid condition is 430V for 10 seconds. The undervoltage is 328V instantly. Place the plant in MODE 3 by 18:00:20 on 1/7/13 is incorrect. Plausible if they see the 7-day requirement and think that 1/7/13 would be the "seventh" day. This would cause them to believe that they have missed the time for condition B and would go to condition C which is be in MODE 3 in 6 hours.

B) Incorrect. 1/2 is incorrect. Plausible because this is the relay coincidence required for undervoltage, this is a degraded voltage condition. A degraded grid condition is 430V for 10 seconds. The undervoltage is 328V instantly. Place the plant in MODE 3 by 18:00:20 on 1/8/13 is correct. This would make the 7 days they had to restore the EDG plus the 6 hours they have to get the plant in MODE 3.

C) Incorrect. Degraded voltage requires 2/3 relays to start the EDG. A degraded grid condition is 430V for 10 seconds. The undervoltage is 328V instantly. Place the plant in MODE 3 by 18:00:20 on 1/7/13 is incorrect. Plausible if they see the 7-day requirement and think that 1/7/13 would be the "seventh" day. This would cause them to believe that they have missed the time for condition B and would go to condition C which is be in MODE 3 in 6 hours.

D) Correct. Degraded voltage requires 2/3 relays to start the EDG. A degraded grid condition is 430V for 10 seconds. The undervoltage is 328V instantly. Place the plant in MODE 3 by 18:00:20 on 1/8/13 is correct. This would make the 7 days they had to restore the EDG plus the 6 hours they have to get the plant in MODE 3.

Question: 80
Tier/Group: 1/1
K/A Importance Rating: RO 3.6 SRO 4.0
K/A: 077 Generator Voltage and Electric Grid Disturbances
AA2: Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:
AA2.07: Operational status of engineered safety features
Reference(s): Sim/Plant design, LCO 3.8.1
Proposed References to be provided to applicants during examination: LCO 3.8.1
Learning Objective: Objective 15 of EDG lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.5 and 43.5 / 45.5, 45.7, and 45.8)
Comments:

This question meets the K/A because based off of electrical grid disturbances, the candidate must know the required number of relays to activate the degraded grid voltage relays to start the EDG's. The candidate must also apply LCO 3.8.3 for the EDG being inoperable. Application of the LCO makes this an SRO level question.

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources• Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. The qualified circuit between the offsite transmission network and the onsite emergency AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) capable of supplying the onsite emergency power distribution subsystem(s).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. The qualified offsite circuit inoperable.	A.1	Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable.	12 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s).
	<u>AND</u>		
	A.2	Restore offsite circuit to OPERABLE status.	24 hours <u>AND</u> 8 days from discovery of failure to meet LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for the offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Perform SR 3.8.1.2 for OPERABLE DG	24 hours
	OR		
	B.3.2.1	Determine OPERABLE DG is not inoperable due to common cause failure. AND	24 hours
	B.3.2.2	Perform SR 3.8.1.2 for OPERABLE DG.	96 hours
	AND		
			(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	7 days AND 8 days from discovery of failure to meet LCO
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Two or more AC sources inoperable.	------------NOTE------------ Entry into this Required Action may be delayed for no greater than 2 hours during performance of Required Action B.3.1 and Required Action B.3.2.2. ----------------------------- D.1 Enter LCO 3.0.3.	Immediately

CORRECT

B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs on the emergency bus. There are two LOP start signals for each 480 V emergency bus.

Undervoltage relays with definite time characteristics are provided on each 480 V emergency bus for detecting a sustained degraded voltage condition or a loss of bus voltage. The Loss of Voltage Function is provided by two relays on each bus. These relays are arranged in a one-out-of-two logic, such that either relay will generate an LOP signal if the voltage is below approximately 68% for a short time (loss of bus voltage). The Degraded Voltage Function is provided by three relays on each bus, which are combined in a two-out-of-three logic to generate an LOP signal if the voltage is below approximately 90% for a long period of time (degraded voltage). The LOP start actuation is described in UFSAR, Section 8.3 (Ref. 1).

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the Degraded Grid Voltage Study (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

Trip Setpoints and tolerances are specified for each Function in the LCO. If the measured setpoint falls within the tolerance band, the relay is considered OPERABLE. Operation with a measured setpoint less conservative than the Trip Setpoint, but within the tolerance band, is acceptable provided that operation and testing is consistent with the assumptions of the setpoint calculation. Each Trip Setpoint specified is more conservative than the analytical values determined in Reference 2 in order to account for instrument uncertainties appropriate to the trip function.

(continued)

Incorrect

B 3.3 INSTRUMENTATION

B 3.3.5 Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation

BASES

BACKGROUND

The DGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate an LOP start if a loss of voltage or degraded voltage condition occurs on the emergency bus. There are two LOP start signals for each 480 V emergency bus.

Undervoltage relays with definite time characteristics are provided on each 480 V emergency bus for detecting a sustained degraded voltage condition or a loss of bus voltage. The Loss of Voltage Function is provided by two relays on each bus. These relays are arranged in a one-out-of-two logic, such that either relay will generate an LOP signal if the voltage is below approximately 68% for a short time (loss of bus voltage). The Degraded Voltage Function is provided by three relays on each bus, which are combined in a two-out-of-three logic to generate an LOP signal if the voltage is below approximately 90% for a long period of time (degraded voltage). The LOP start actuation is described in UFSAR, Section 8.3 (Ref. 1).

Trip Setpoints and Allowable Values

The Trip Setpoints used in the relays are based on the Degraded Grid Voltage Study (Ref. 2). The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account.

Trip Setpoints and tolerances are specified for each Function in the LCO. If the measured setpoint falls within the tolerance band, the relay is considered OPERABLE. Operation with a measured setpoint less conservative than the Trip Setpoint, but within the tolerance band, is acceptable provided that operation and testing is consistent with the assumptions of the setpoint calculation. Each Trip Setpoint specified is more conservative than the analytical values determined in Reference 2 in order to account for instrument uncertainties appropriate to the trip function.

(continued)

Provided reference

3.8 ELECTRICAL POWER SYSTEMS

3.8.1 AC Sources—Operating

LCO 3.8.1 The following AC electrical sources shall be OPERABLE:

a. The qualified circuit between the offsite transmission network and the onsite emergency AC Electrical Power Distribution System; and

b. Two diesel generators (DGs) capable of supplying the onsite emergency power distribution subsystem(s).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The qualified offsite circuit inoperable.	A.1 Declare required feature(s) with no offsite power available inoperable when its redundant required feature(s) is inoperable. AND A.2 Restore offsite circuit to OPERABLE status.	12 hours from discovery of no offsite power to one train concurrent with inoperability of redundant required feature(s). 24 hours AND 8 days from discovery of failure to meet LCO

(continued)

Provided Reference

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One DG inoperable.	B.1	Perform SR 3.8.1.1 for the offsite circuit.	1 hour AND Once per 12 hours thereafter
	AND		
	B.2	Declare required feature(s) supported by the inoperable DG inoperable when its required redundant feature(s) is inoperable.	4 hours from discovery of Condition B concurrent with inoperability of redundant required feature(s)
	AND		
	B.3.1	Perform SR 3.8.1.2 for OPERABLE DG	24 hours
	OR		
	B.3.2.1	Determine OPERABLE DG is not inoperable due to common cause failure.	24 hours
		AND	
	B.3.2.2	Perform SR 3.8.1.2 for OPERABLE DG.	96 hours
	AND		
			(continued)

Provided Reference

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Restore DG to OPERABLE status.	7 days AND 8 days from discovery of failure to meet LCO
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Two or more AC sources inoperable.	-----------NOTE----------- Entry into this Required Action may be delayed for no greater than 2 hours during performance of Required Action B.3.1 and Required Action B.3.2.2. -------------------------- D.1 Enter LCO 3.0.3.	 Immediately

81. W/E 05 EA2.2 SRO 001

Given the following plant conditions:

-The crew is implementing FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK

-CST is intact, level is 15% and lowering

OP-402, AUXILIARY FEEDWATER SYSTEM
FRP-H.1, ATTACHMENT 2, SERVICE WATER BACKUP TO MDAFW PUMPS

Which ONE (1) of the following completes the statements below?

The correct order in which the crew will attempt to restore Feed Flow to S/G's is AFW, __(1)__ . If the CST level reaches 10%, the CRS will direct the use of __(2)__ to align Service Water as a backup to the AFW pumps.

A.✓ (1) Main Feedwater, Condensate
(2) OP-402

B. (1) Main Feedwater, Condensate
(2) ATTACHMENT 2

C. (1) Condensate, Main Feedwater
(2) OP-402

D. (1) Condensate, Main Feedwater
(2) ATTACHMENT 2

The correct answer is A

A) Correct. The crew will attempt to restore feed using AFW, Main Feedwater, then Condensate in that order. Because the CST is not damaged, the CRS will direct the use of OP-402 to allign service water to the afw pumps as a backup source.

B) Incorrect. The crew will attempt to restore feed using AFW, Main Feedwater, then Condensate in that order. ATTACHMENT 2 is incorrect. Plausible because FRP-H.1 directs you to attachment 2 if CST level is lowering due to catastrophic failure.

C) Incorrect. AFW, Condensate, Main Feedwater is incorrect. Plausible because FRP-H.1 first has you check to see if the condensate system is available, if so, you first try to start a Main Feed Pump, if you can't, you move on to try and start a Condensate pump. Because the CST is not damaged, the CRS will direct the use of OP-402 to allign service water to the afw pumps as a backup source.

D) Incorrect. AFW, Condensate, Main Feedwater is incorrect. Plausible because FRP-H.1 first has you check to see if the condensate system is available, if so, you first try to start a Main Feed Pump, if you can't, you move on to try and start a Condensate pump. ATTACHMENT 2 is incorrect. Plausible because FRP-H.1 directs you to attachment 2 if CST level is lowering due to catastrophic failure.

Question: 81
Tier/Group: 2/1
K/A Importance Rating: SRO 4.3
K/A: W/E05 Loss of Secondary Heat Sink
EA2: Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink)
EA2.2: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
References: Sim/Plant design, FRP-H.1 and its basis document
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5/6 of FRP-H.1 lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR part 55: (CFR 41.10/43.5/45.13)

Meets the K/A because the candidate must determine the correct order for restoring feed flow within FRP-H.1. This is SRO level because the candidate must select a procedure from which to procede given the conditions above.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6.	**Check CST level - GREATER THAN 10%**	Align SW backup to the AFW Pumps using OP-402, Auxiliary Feedwater System, while continuing with this procedure. IF the CST is low due to catastrophic failure AND is inaccessable, THEN align SW backup to the MDAFW Pumps using Attachment 2, SW Backup To MDAFW Pumps. Go To Step 14.
7.	**Verify All S/G Blowdown AND Sample Isolation Valves - CLOSED**	
8.	**Check AFW Lines - INTACT**	Isolate break. IF the break is isolated, THEN Go To Step 9. IF the break can NOT be isolated, THEN Go To Step 14.

correct

incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

Feed flow is not re-established to any faulted S/G if an intact S/G is available.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Total Feed Flow - LESS THAN 300 GPM DUE TO OPERATOR ACTION**	Go To Step 3.
2.	**Reset SPDS And Return To Procedure And Step In Effect**	
3.	**Determine If Secondary Heat Sink Is Required As Follows:**	
	a. Check RCS pressure - GREATER THAN ANY NON-FAULTED S/G PRESSURE	a. Reset SPDS and return to procedure and step in effect.
	b. Check RCS temperature - GREATER THAN 350°F [310°F]	b. Perform the following: 1) Place RHR System in service using Supplement I. 2) WHEN adequate cooling with RHR is established, THEN reset SPDS and return to procedure and step in effect.
* 4.	**Check Any Two S/G Wide Range Levels - LESS THAN 10% [19%]**	IF any two S/G Wide Range Levels lower to less than 10% [19%], THEN Go To Step 5. Go To Step 6.
5.	**Perform The Following:**	
	a. Stop all RCPs	
	b. Observe CAUTION prior to Step 31 and Go To Step 31	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 6.	**Check CST level - GREATER THAN 10%**	Align SW backup to the AFW Pumps using OP-402, Auxiliary Feedwater System, while continuing with this procedure. <u>IF</u> the CST is low due to catastrophic failure <u>AND</u> is inaccessable, <u>THEN</u> align SW backup to the MDAFW Pumps using Attachment 2, SW Backup To MDAFW Pumps. Go To Step 14.
7.	**Verify All S/G Blowdown <u>AND</u> Sample Isolation Valves - CLOSED**	
8.	**Check AFW Lines - INTACT**	Isolate break. <u>IF</u> the break is isolated, <u>THEN</u> Go To Step 9. <u>IF</u> the break can <u>NOT</u> be isolated, <u>THEN</u> Go To Step 14.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
9.	**Try To Establish Motor Driven AFW Flow To At Least One S/G As Follows:**	
	a. Check AFW Pump Breakers - TRIPPED	a. Go To Step 9.c.
	b. Attempt to reclose any tripped breakers as follows:	
	1) Position the MDAFW Pump Control Switch to the STOP position	
	2) Reset SI	
	3) Position the MDAFW Pump Control Switch to the START position	
	4) Check MDAFW Pump - RUNNING	4) IF the tripped breaker will NOT reclose, THEN contact I&C to investigate. Go To Step 10.
	c. Verify AFW HDR DISCH Valves - OPEN: • V2-16A • V2-16B • V2-16C	
	d. Check AFW flow to S/Gs - GREATER THAN 300 GPM	d. Go To Step 10.
	e. Reset SPDS and return to procedure and step in effect	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
10.	**Attempt To Start SDAFW Pump As Follows:**	
	a. Verify STEAM DRIVEN AFW PUMP STM SHUTOFF Valves - OPEN • V1-8A • V1-8B • V1-8C	a. IF the steam supply valves can NOT be opened, THEN Go To Step 11.
	b. Verify STEAM DRIVEN AFW PUMP DISCH Valves - OPEN • V2-14A • V2-14B • V2-14C	
	c. Check AFW flow to S/Gs - GREATER THAN 300 GPM	c. Go To Step 11.
	d. Reset SPDS and return to procedure and step in effect	
11.	**Locally Investigate AND Attempt To Restore AFW Flow As Follows:**	
	a. Verify AFW Pump suction supply is available	
	b. Position the MDAFW Pump LOCAL/REMOTE Switch to LOCAL	
	c. Attempt to start a MDAFW Pump as follows:	
	1) Depress the MDAFW Pump local STOP Pushbutton	
	2) Depress the MDAFW Pump local START Pushbutton	
	3) Check MDAFW Pump - STARTED	3) Place the LOCAL/REMOTE Switch to REMOTE.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*12.	**Check AFW Flow To S/Gs - GREATER THAN 300 GPM**	IF feed flow to at least one S/G verified, THEN perform the following: a. Maintain flow to restore narrow range level to greater than 8% [18%]. b. WHEN narrow range level is greater than 8% [18%], THEN reset SPDS AND return to procedure and step in effect. Go To Step 14.
13.	**Reset SPDS And Return To Procedure And Step In Effect**	
14.	**Stop All RCPs**	
15.	**Check Condensate System - IN SERVICE** Showing That The Condensate System is checked in service Prior To Starting a main Feed Pump	Place the Condensate System in service as follows: a. IF the Condensate System is NOT available, THEN Go To Step 30. b. Open QCV-10426, COND POL SEC BYP. c. Close V5-3, COND PUMP DISCH. d. Momentarily place V5-3 to OPEN. e. Start one Condensate Pump. f. WHEN feedwater pressure is greater than 300 psig, THEN verify V5-3 full open. g. Open HCV-1459, LP HEATERS BYP. IF at least one Condensate Pump can NOT be started, THEN Go To Step 30.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

The subsequent step will defeat all FW Isolation signals which is necessary to allow starting of a Main Feedwater Pump. Manual Operator action will be required to initiate a FW Isolation.

16. **Place ALL The FEEDWATER ISOLATION Key Switches In The OVRD/RESET Position**

    - STM GEN A
    - STM GEN B
    - STM GEN C

NOTE

Local operation of the FRV and B/P valves below is via reverse acting handwheels.

17. **Attempt To Establish Feedwater Flow As Follows:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Verify the FW HDR SECTION Valves - CLOSED • V2-6A • V2-6B • V2-6C	
b. Start one Main FW Pump	b. Go To Step 20.
c. Open the FRV Bypass Valves: • FCV-479 • FCV-489 • FCV-499	c. Locally open the FRV Bypass Valve using the Manual Handwheel. (Requires small Locked Valve Key.)
d. Check FW Flow - ESTABLISHED	d. Go To Step 20.

82. 001 AA2.05 SRO 002

Given the following plant conditions:

-Unit 2 is raising power from 50% to 100% RTP

-Reactor power is currently at 55% RTP at EOL

-Control Bank D rods are at 171 steps

-30 minutes ago, a 200 gallon dilution to the RCS was performed

-The OAC pulls rods and releases the switch, Tavg and Reactor power steadily rise

-APP-005-E4, DELTA FLUX ALARM, is in

-VCT level has remained stable at 25% for the last 15 minutes

Which ONE (1) of the following completes the statements below?

Based on the above conditions, the accident the crew is dealing with is a ___(1)___ accident. The basis for the LCO actions to address the Delta Flux Alarm is to ___(2)___ .

A.✓ (1) uncontrolled rod withdraw
(2) limit the amount of axial power distribution

B. (1) dilution
(2) limit the amount of axial power distribution

C. (1) uncontrolled rod withdraw
(2) limit the gross radial power distribution

D. (1) dilution
(2) limit the gross radial power distribution

The correct answer is A

A) Correct. Tavg and Reactor power rising are indications of a uncontrolled rod withdraw. APP-005-E4 refers you to LCO 3.2.3, Axial Flux Difference, which the basis is to limit the axial power distribution.

B) Incorrect. Dilution is incorrect. Plausible if the student thinks that 200 gallon dilution is too much. A normal dilution at power for temperature control is roughly 20 gallons. However, 200 gallons would be an appropriate amount for raising power. The candidate should also realize that VCT level is constant, during a dilution, level would be rising. Limit the amount of axial power distribution is correct. APP-005-E4 refers you to LCO 3.2.3, Axial Flux Difference, which the basis is to limit the axial power distribution.

C) Incorrect. Uncontrolled rod withdraw is correct. Limit the gross radial power distribution is incorrect. Plausible if the student does not realize that APP-005-E4 sends you to LCO 3.2.3, not 3.2.4. 3.2.4 is QPTR and that LCO deals with limiting the gross radial power distribution. Both Deal with controlling power distribution, one is radially and the other is axially.

D) Incorrect. Dilution is incorrect. Plausible if the student thinks that 200 gallon dilution is too much. A normal dilution at power for temperature control is roughly 20 gallons. However, 200 gallons would be an appropriate amount for raising power. The candidate should also realize that VCT level is constant, during a dilution, level would be rising. Limit the gross radial power distribution is incorrect. Plausible if the student does not realize that APP-005-E4 sends you to LCO 3.2.3, not 3.2.4. 3.2.4 is QPTR and that LCO deals with limiting the gross radial power distribution. Both Deal with controlling power distribution, one is radially and the other is axially.

Question: 82
Tier/Group: 1/2
K/A Importance Rating: SRO 4.6
K/A: 001 Continuous Rod Withdrawal
AA2: Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal :
AA2.05: Uncontrolled rod withdrawal, from available indications
Reference(s): Simulator/plant design, APP-005-E4, LCO 3.2.3, 3.2.4 Basis Documents
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 43.5 / 45.13
Comments:

This question meets the K/A because from the given indications, the student must determine and interpret the casualty. To make it an SRO level question, the student needs to know the basis behind the Axial Flux Difference LCO.

Correct

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the values of the AFD in order to limit the axial power distribution skewing to either the top or bottom of the core. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses. Limiting power distribution skewing over time also minimizes the xenon distribution skewing, which is a significant factor in axial power distribution control.

The operating scheme used to control the axial power distribution, PDC-3, involves maintaining the AFD within a tolerance band around a burnup dependent target, known as the target flux difference, to minimize the variation of the axial peaking factor and axial xenon distribution during unit maneuvers.

The target flux difference is determined at equilibrium xenon conditions. The control banks must be positioned within the core in accordance with their insertion limits and Control Bank D should be inserted near its normal position (i.e., ≥ 210 steps withdrawn) for steady state operation at high power levels. The power level should be as near RTP as practical. The value of the target flux difference obtained under these conditions divided by the Fraction of RTP is the target flux difference at RTP for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RTP value by the appropriate fractional THERMAL POWER level.

Periodic updating of the target flux difference value is necessary to follow the change of the flux difference at steady state conditions with burnup.

The Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) and QPTR LCOs limit the radial component of the peaking factors.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

Incorrect

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," LCO 3.2.4, and LCO 3.1.6, "Control Bank Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 departure from nucleate boiling (DNB) criterion) that the hot fuel rod in the core does not experience a DNB condition (Ref. 2);

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 4).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot

(continued)

Δ FLUX ALARM

CAUSES:
1. Two or more operable Excore Channels outside their Target Bands with Reactor Power above 90% of rated power or 0.9 x APL.
2. Two or more operable Excore Channels outside the Operating Band with Reactor Power above 50% but less than or equal to 90% of rated power or 0.9 x APL.
3. Accumulation of greater than 60 Penalty Points with Reactor Power above 50% of rated power.

DEVICE:	SETPOINT:	LOCATION:
1. ERFIS (Refer to FMP-009)	Calculated by Core Axial Offset Calculation (CAOC)	MUX #4

OPERATOR ACTIONS

1. MONITOR any of the following parameters:
   - Reactor Power .... ______
   - Δ Flux .... ______
   - ERFIS Δ Flux Program printout .... ______
2. NOTIFY Reactor Engineering .... ______
3. REFER TO FMP-009, Power Distribution Control, to restore Δ Flux to acceptable values per Control Room Status Board. .... ______
4. REFER TO TS 3.2.3, Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology). .... ______

REFERENCES

1. B-190628 Sheet 441, Control Wiring Diagram (Cable AV)
2. HBR2-11098, Annunciator Window Engraving & Input Tabulation APP-005

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (Continued)		
f. Plant Vent flow rate	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that flow rate is estimated once per 4 hours.
2. Containment Vessel via Plant Vent		
a. Radionoble gas monitor (R-12) provides automatic termination of Containment Vessel releases upon exceeding alarm/trip Setpoint.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)
b. Radioparticulate Monitor (R-11) provides automatic termination of containment vessel releases exceeding alarm/trip setpoints.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)

* MCO - Minimum Channels Operable

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
6. Radwaste Building Exhaust (Continued)		
c. Sampler flow rate gauge	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided the flow rate is estimated once per 4 hours.
7. Deleted.	NA	NA

* MCO - Minimum Channels Operable

12/06

NOTES TO TABLE 3.10-1

Note 1 - No auxiliary sampling is required for periods when normal sampling is off ≤ 45 minutes.

Note 2 - This MCO is required during Modes 1, 2, 3, 4, and during the movement of recently irradiated fuel assemblies within the containment.

CHAPTER 6

SYSTEM OPERATION

Normal Operation

CONTAINMENT HVAC

Outdoor Air Makeup and Containment Purge (HVE-1A and HVE-1B)

Prior to activating these systems after shutdown, the containment particulate and gas monitors R-11 and R-12 are used to monitor the airborne activity levels inside the containment as a guide for routine release from the building. One of the following is required for monitoring during CV purge operation: 1) R-11 **AND** R-12, 2) R-14C (Refer to ODCM Table 3.10-1). In Modes 1,2,3,4 and when moving recently irradiated fuel in the CV, the CV Ventilation Isolation signal from R-11 **AND** R-12 is required to be OPERABLE (Refer to ITS 3.3.6). If R-11 **OR** R-12 become inoperable during CV Purge operation when ITS 3.3.6 is applicable, the CV Purge supply and exhaust valves shall be closed IMMEDIATELY. The Outdoor Air Makeup system has an air motor operated damper that opens when the containment purge fan HVE-1A or HVE-1B is started or when the Vacuum Relief system is energized. The air temperature leaving the heating coils is used to control steam to the coils. The Containment Purge system is manually energized from the RTGB. When the containment purge fan is selected, the outside air louver opens and the butterfly valves open, the fan then starts and the intake damper to that fan opens. Fans are interlocked so that if the running fan motor trips electrically the other fan motor will start. Stopping the fan de-energizes the controls, the dampers, intake louver, and the isolation valves shut.

IF R-11 OR R-12 alarms, <u>THEN</u> the running CV PURGE FAN, HVE-1A or HVE-1B will stop, and the following valves close:

- V12-6, CV PURGE INLET
- V12-7, CV PURGE INLET
- V12-8, CV PURGE OUTLET
- V12-9, CV PURGE OUTLET
- V12-10, CV PRESS RELIEF
- V12-11, CV PRESS RELIEF
- V12-12, CV VAC RELIEF

Provided Reference

3.10 Radioactive Gaseous Effluent Monitoring Instrumentation

Applicability

Applies to the radioactive gaseous effluent instrumentation system.

Objective

To define the operating requirements for the radioactive gaseous effluent instrumentation system.

Specification

CONTROLS

3.10.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.10-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of ODCM Specification 3.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

ACTIONS

3.10.2 With a radioactive effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.10.3 With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels operable take the action shown in Table 3.10-1.

3.10.4 The provisions of ODCM Specification 8.1 are not applicable.

BASES

Radioactive Gaseous Effluent Instrumentation

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20, Appendix B, Table 2, Column 1. The operability and use of this instrumentation are consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

83. 060 AG2.2.40 SRO 001

Given the following plant conditions:

-The plant is in MODE 1

-A CV Purge is in progress

-R-14C, PLANT STACK, NOBLE GAS, is OOS

-R-12, CV AIR OR PLANT STACK, NOBLE GAS, fails high

Which ONE (1) of the following completes the statements below?

The running CV PURGE EXHAUST UNIT, HVE-1A or HVE-1B will ___(1)___ . IAW the ODCM, OFF-SITE DOSE CALCULATION MANUAL, effluent releases ___(2)___ continue.
(REFERENCES PROVIDED)

A. (1) trip
(2) can

B.✓ (1) trip
(2) can **NOT**

C. (1) **NOT** trip
(2) can

D. (1) **NOT** trip
(2) can **NOT**

The correct answer is B.

A) Incorrect. HVE-1A or B, depending on which one is running, will trip from R-12 failing high. R-12 failing high causes a containment isolation signal which in turn, trips the running HVE-1A or B. The effluent release cannot continue. Plausible if the student only goes to the section for R-14C being out of service.

B) Correct. HVE-1A or B, depending on which one is running, will trip from R-12 failing high. R-12 failing high causes a containment isolation signal which in turn, trips the running HVE-1A or B. The effluent release cannot continue on because both R-12 and R-14C are now OOS.

C) Incorrect. HVE-1A or B, depending on which one is running, will trip from R-12 failing high. Plausible if the student does not know what happens on a Containment Isolation signal or thinks that both R-11 AND R-12 must alarm to stop the purge. The effluent release cannot continue. Plausible if the student only goes to the section for R-14C being out of service.

D) Incorrect. HVE-1A or B, depending on which one is running, will trip from R-12 failing high. Plausible if the student does not know what happens on a Containment Isolation signal. The effluent release cannot continue on because both R-12 and R-14C are now OOS.

Question: 83
Tier/Group: 1/2
K/A Importance Rating: SRO 4.7
K/A: 060 Accidental Gaseous Radwaste Release
AG 2.2.40: Ability to apply Technical Specifications for a system.
Reference(s): Sim/Plant design, ODCM
Proposed References to be provided to applicants during examination: Pg's 130, 131, 133, 139 of the ODCM
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR 41.10/43.2/43.5/45.3)
Comments:

Meets the K/A because the student must apply the ODCM(Tech Specs) to the given scenario. Application of the ODCM(Tech Specs) makes this an SRO level question.

3.10 Radioactive Gaseous Effluent Monitoring Instrumentation

Applicability

Applies to the radioactive gaseous effluent instrumentation system.

Objective

To define the operating requirements for the radioactive gaseous effluent instrumentation system.

Specification

CONTROLS

3.10.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.10-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of ODCM Specification 3.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the ODCM.

ACTIONS

3.10.2 With a radioactive effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, without delay suspend the release of radioactive gaseous effluents, change the setpoint so it is acceptably conservative, or declare the channel not operable.

3.10.3 With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels operable take the action shown in Table 3.10-1.

3.10.4 The provisions of ODCM Specification 8.1 are not applicable.

BASES

Radioactive Gaseous Effluent Instrumentation

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20, Appendix B, Table 2, Column 1. The operability and use of this instrumentation are consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

TABLE 3.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (R-14)		
a. Radionoble gas monitor (R14C) provides automatic termination of Waste Gas Decay Tank releases upon exceeding alarm/trip setpoint.	1	With the number of channels operable less than the MCO requirements: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that prior to initiating a waste gas decay tank release: 1. Two independent samples are analyzed in accordance with the Surveillance Requirements of ODCM Specification 3.2.1 and; 2. Two members of the facility staff independently verify the release rate calculations and the discharge line valving.
b. Radionoble gas monitor (R14C) monitors all effluents from Auxiliary Building Ventilation System without providing automatic termination of release upon exceeding their respective alarm setpoints.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that grab samples are collected once per 12 hours and are analyzed for radionoble gases within 24 hours.

* MCO - Minimum Channels Operable

Provided Reference

TABLE 3.10-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (R-14)		
a. Radionoble gas monitor (R14C) provides automatic termination of Waste Gas Decay Tank releases upon exceeding alarm/trip setpoint.	1	With the number of channels operable less than the MCO requirements: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that prior to initiating a waste gas decay tank release: 1. Two independent samples are analyzed in accordance with the Surveillance Requirements of ODCM Specification 3.2.1 and; 2. Two members of the facility staff independently verify the release rate calculations and the discharge line valving.
b. Radionoble gas monitor (R14C) monitors all effluents from Auxiliary Building Ventilation System without providing automatic termination of release upon exceeding their respective alarm setpoints.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that grab samples are collected once per 12 hours and are analyzed for radionoble gases within 24 hours.

* MCO - Minimum Channels Operable

Provided Reference

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
1. Plant Vent (Continued)		
f. Plant Vent flow rate	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that flow rate is estimated once per 4 hours.
2. Containment Vessel via Plant Vent		
a. Radionoble gas monitor (R-12) provides automatic termination of Containment Vessel releases upon exceeding alarm/trip Setpoint.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)
b. Radioparticulate Monitor (R-11) provides automatic termination of containment vessel releases exceeding alarm/trip setpoints.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the Plant Vent Radionoble Gas Monitor (R14C) is operable; otherwise, suspend all releases via this pathway. (note 2)

* MCO - Minimum Channels Operable

Provided reference

TABLE 3.10-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
6. Radwaste Building Exhaust (Continued) c. Sampler flow rate gauge	 1	 With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided the flow rate is estimated once per 4 hours.
7. Deleted.	NA	NA

* MCO - Minimum Channels Operable

12/06

NOTES TO TABLE 3.10-1

Note 1 - No auxiliary sampling is required for periods when normal sampling is off ≤ 45 minutes.

Note 2 - This MCO is required during Modes 1, 2, 3, 4, and during the movement of recently irradiated fuel assemblies within the containment.

84. 061 AA2.02 SRO 001

Given the following plant conditions:

-The reactor is at 100% RTP

-The BOP reports R-9 is trending up and currently reads 0.6 R/hr

OP-920, RADIATION MONITORING SYSTEM
OMM-014, RADIATION MONITOR SETPOINTS

Which ONE(1) of the following completes the statements below?

The BOP will verify R-9 setpoint IAW ___(1)___ . There ___(2)___ an EAL declaration that needs to be entered.

(REFERENCE PROVIDED)

A. (1) OMM-014
(2) is **NOT**

B.✓ (1) OMM-014
(2) is

C. (1) OP-920
(2) is **NOT**

D. (1) OP-920
(2) is

The correct answer is B

A) Incorrect. OMM-014 will be used to verify R-9 setpoint. There is not an EAL declaration is incorrect. Plausible if the student does not convert R/hr to mr/hr.

B) Correct. OMM-014 will be used to verify R-9 setpoint. There is an EAL declaration that needs to be made. R-9 is currently reading 600 mr/hr. SU5.1 for R-9 reading >500mr/hr.

C) Incorrect. OP-920 is incorrect. Plausible since this is the procedure that is used to adjust the alarm setpoint for R-9. There is not an EAL declaration is incorrect. Plausible if the student does not convert R/hr to mr/hr.

D) Incorrect. OP-920 is incorrect. Plausible since this is the procedure that is used to adjust the alarm setpoint for R-9. There is an EAL declaration that needs to be made. R-9 is currently reading 600 mr/hr. SU5.1 for R-9 reading >500mr/hr.

Question: 84
Tier/Group: 1/2
K/A Importance Rating: SRO 3.2
K/A: 061 Area Radiation Monitoring (ARM) System Alarms
AA2: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:
AA2.02: Normal radiation intensity for each ARM system channel .
Reference(s): Sim/Plant design, OMM-014, Hot Conditions EAL
Proposed References to be provided to applicants during examination: Hot Conditions EAL
Learning Objective: Self Study procedure
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 43.5 / 45.13)
Comments:

This question meets the K/A because the candidate must determine if this is a normal reading IAW OMM-014. From this, they must determine if an EAL Declaration is necessary. The EAL decleration is what makes this an SRO level question.

Correct

1.0 **PURPOSE**

1.1 Provide a list of setpoints for the Area, Process and Accident Radiation Monitors.

1.2 Provide a discussion of the basis used for calculation of the setpoints.

1.3 Provide a method of documenting current setpoints and setpoint changes for the listed Area Radiation Monitors, Process Radiation Monitors, Accident Radiation Monitors, and the Radiation Monitoring System (RMS) Recorder.

2.0 **REFERENCES**

2.1 Improved Technical Specification LCO 3.3.6 and 3.9.3

2.2 UFSAR - Sections 11 and 12

2.3 10CFR 20, Appendix B

2.4 AOP-005, Radiation Monitoring System

2.5 OP-002, Nuclear Instrumentation System

2.6 OP-920, Radiation Monitoring System

2.7 RST-001, Radiation Monitor Source Checks

2.8 Station Curve Book, Section 6

2.9 EMP-013, Operation of R-14 and F-14

2.10 EMP-020, Operation of R-22 and R-38

2.11 EMP-022, Gaseous Waste Release Permits

2.12 EMP-023, Liquid Waste Release and Sampling

2.13 EMP-024, ODCM Surveillance

2.14 EMP-027, Calibration and Operation of GA Monitors R-37 and R-19A, B, and C

2.15 EMP-028, Process Monitor Setpoint Determination

2.16 EMP-034, Operation of R-24 A, B, and C

2.17 CP-014, Primary to Secondary Leak Rate Calculation

incorrect

1.0 PURPOSE

1.1 Provide instructions for Placing In-Service, Normal Operation, Removing from Service, and Infrequent Operation of the Area Monitors, Process Monitors, the Westronics Series 3000 Recorder, and the Yokogawa VR204 View Recorder.

2.0 REFERENCES

2.1 Improved Technical Specification LCO 3.3.6 and LCO 3.4.15

2.2 ODCM 3.10 and 3.11

2.3 SD-019, Radiation Monitoring System

2.4 OMM-001-12, Minimum Equipment List and Shift Relief

2.5 OMM-014, Radiation Monitor Setpoints

2.6 OP-001, Reactor Control and Protection System

2.7 OP-406, Steam Generator Blowdown/Wet Layup System

2.8 OP-509-1, Condensate Polishing System

2.9 OP-603, Electrical Distribution

2.10 OP-903, Service Water System

2.11 OP-917, Secondary Sampling System

2.12 OST-021, Daily Surveillances

2.13 OST-924-1, Area Radiation Monitoring System

2.14 OST-924-2, Process Radiation Monitoring System

2.15 RST-001, Radiation Monitor Source Checks

2.16 RST-008, Calibration of Radiation Monitor System, Monitors R-1 through R-8

2.17 RST-009, Calibration of Radiation Monitor System, Monitors R-9, R-30, R-31A, B, C and R-33

2.18 RST-010, Calibration of Radiation Monitoring System, Monitor R-11

85. W/E14 EG2.2.38 SRO 001

Given the following plant conditions:

-A Large Break LOCA has occurred

-Neither CV Spray pump started

-CV Pressure has reached 40.5 psig

FRP-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK
FRP-J.1, RESPONSE TO HIGH CONTAINMENT PRESSURE

Which ONE (1) of the following completes the statements below?

The crew will use __(1)__ to mitigate this event. Containment design pressure __(2)__ been exceeded.

A. (1) FRP-P.1
(2) has **NOT**

B.✔ (1) FRP-J.1
(2) has **NOT**

C. (1) FRP-P.1
(2) has

D. (1) FRP-J.1
(2) has

The correct answer is B

A) Incorrect. FRP-P.1 does not mitigate this casualty since a LBLOCA has occurred. Plausible because you meet the entry conditions for FRP-P.1, however, since it was a LBLOCA, FRP-P.1 will not mitigate this. Containment design pressure has not been exceeded. Design pressure is 42 psig, this pressure is 40.5 psig.

B) Correct. FRP-J.1 will be used to mitigate the high CV pressure casualty since containment pressure is >10psig and there are no CV Spray Pumps running. Containment design pressure has not been exceeded. Design pressure is 42 psig, this pressure is 40.5 psig.

C) Incorrect. FRP-P.1 does not mitigate this casualty since a LBLOCA has occurred. Plausible because you meet the entry conditions for FRP-P.1, however, since it was a LBLOCA, FRP-P.1 will not mitigate this. Containment design pressure has not been exceeded. Plausible because RNP's old containment pressure value was 40.5 psig.

D) Incorrect. FRP-J.1 will be used to mitigate the high CV pressure casualty since containment pressure is >10psig and there are no CV Spray Pumps running. Containment design pressure has not been exceeded. Plausible because RNP's old containment pressure value was 40.5 psig.

Question: 85
Tier/Group: 1/2
K/A Importance Rating: SRO 4.5
K/A: W/E14 High Containment Pressure
EG2.2.38: Knowledge of conditions and limitations in the facility license.
Reference(s): Sim/Plant design, FRP-J.1, LCO 3.6.4 Basis Document
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.7 / 41.10 / 43.1 / 45.13)
Comments:

This question meets the K/A because the student must have knowledge of the basis for LCO 3.6.4, CONTAINMENT PRESSURE, in order to answer this question. This is also what makes this question an SRO level question.

Correct

Purpose and Entry Conditions

(Page 1 of 1)

1. **PURPOSE**

   This procedure provides actions to respond to high containment pressure.

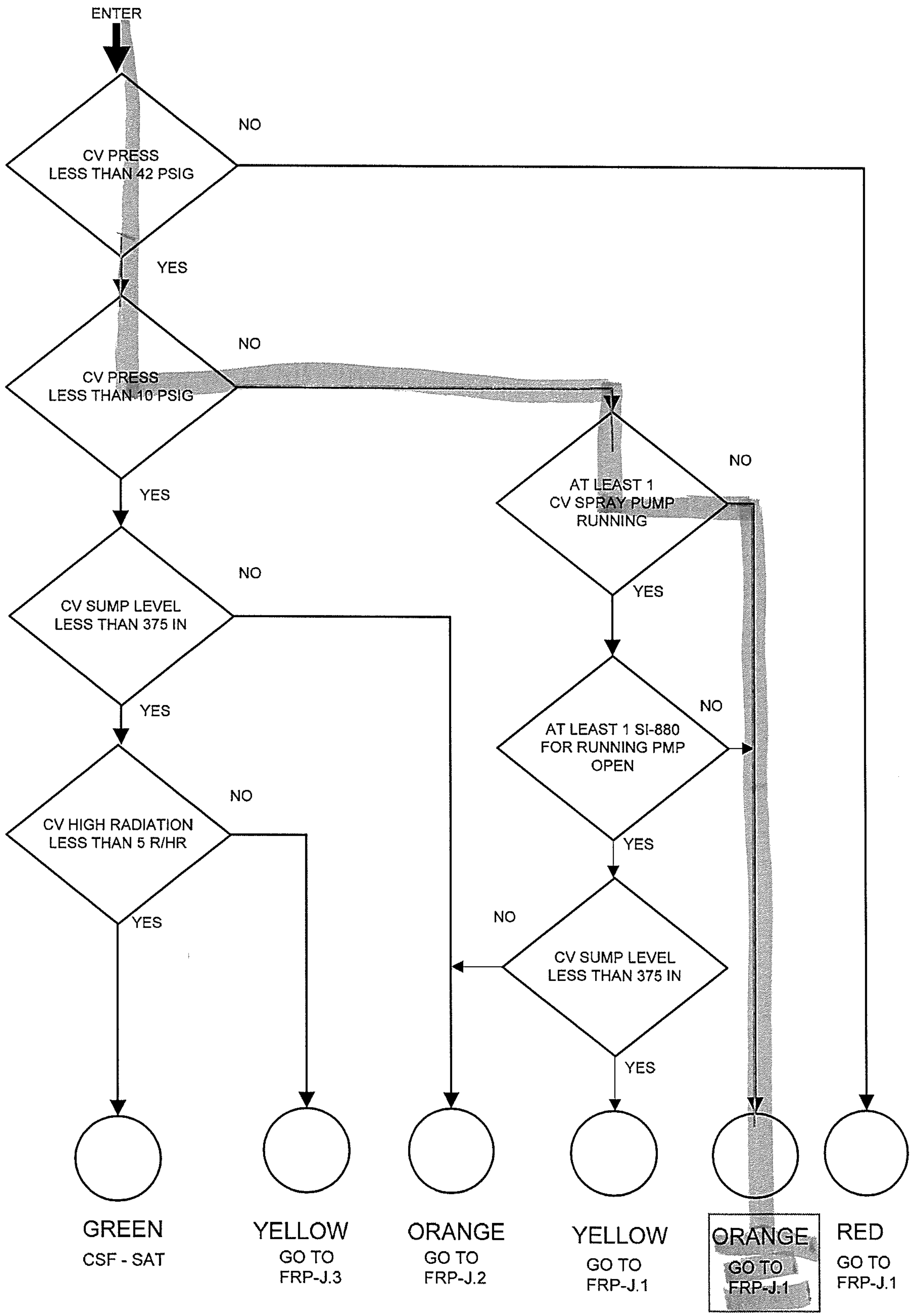
2. **ENTRY CONDITIONS**

   CSF-5, Containment Critical Safety Function Status Tree on a RED, ORANGE or YELLOW condition.

- END -

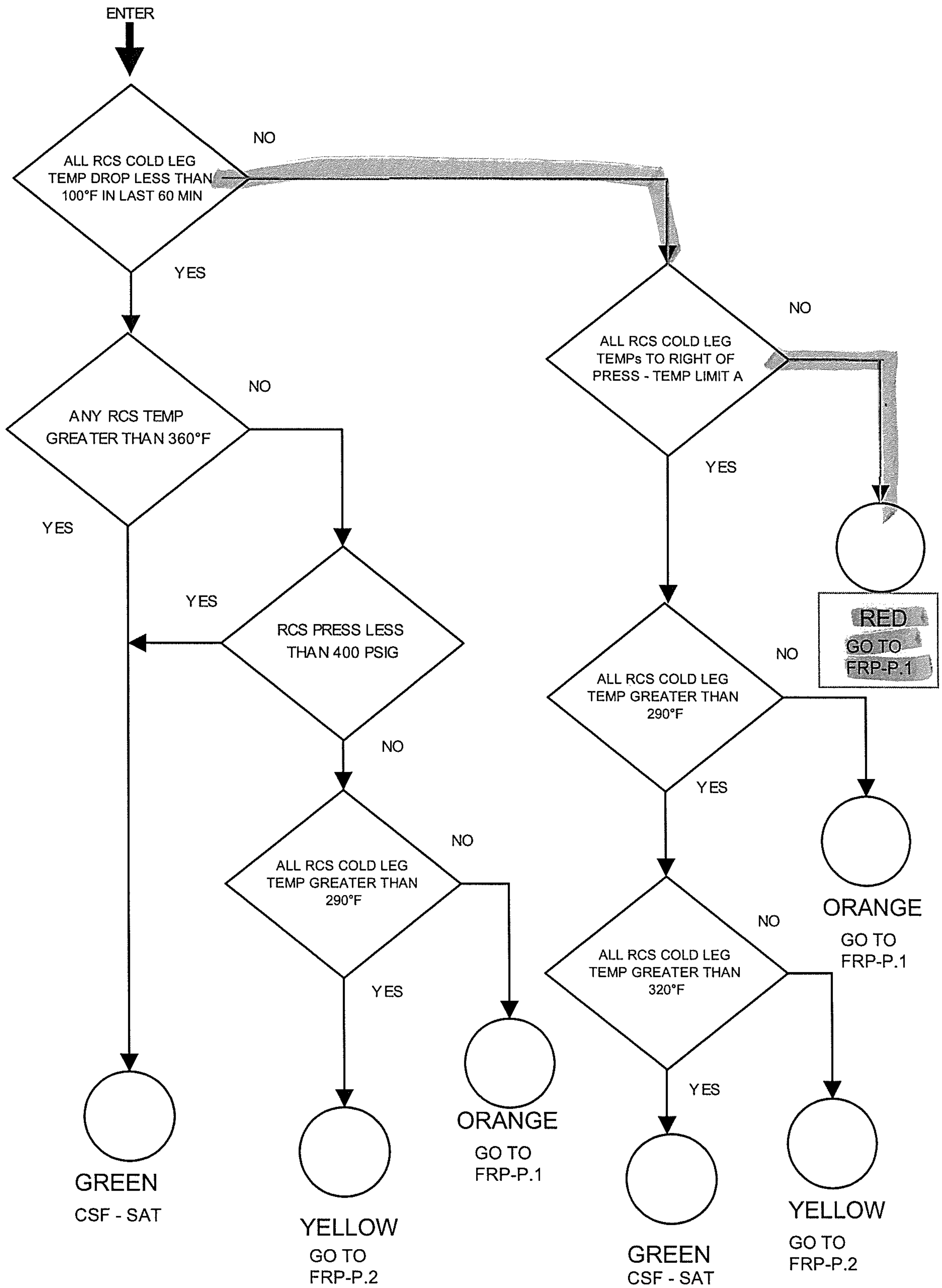
Correct

CSF-5, CONTAINMENT



incorrect

CSF-4, RCS INTEGRITY



incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.	**Check CST Level - LESS THAN 10%**	IF CST Level lowers to less than 10%, THEN perform Step 2. Go To Step 3.
2.	**Align Service Water to the suction of the AFW Pumps using OP-402, Auxiliary Feedwater System.**	IF Service Water is unavailable, THEN align Fire Water to the CST using EPP-1, Attachment 3, CST Emergency Fill From The Fire System.
* 3.	**Determine If RCS Cooldown Is Due To A Large Break LOCA As Follows:**	
	a. Check both of the following conditions exist: • RCS pressure - LESS THAN 275 PSIG [350 PSIG] AND • RHR flow on FI-605 - GREATER THAN 1200 GPM	a. Go To Step 4.
	b. Reset SPDS AND return to procedure and step in effect	
4.	**Check RCS Cold Leg Temperature - LOWERING**	Go To Step 11.
5.	**Attempt To Stop RCS Cooldown As Follows:**	
	a. Verify STEAM LINE PORVs - CLOSED	
	b. Verify COND DUMPs - CLOSED	
	c. Check RHR System - ALIGNED FOR CORE COOLING	c. Go To Step 6.
	d. Stop cooldown from RHR System	

86. 012 G2.1.20 SRO 001

Given the following plant conditions:

Initial Conditions:

-APP-006-D4, S/G A STM LINE HI FLOW, alarm is in

-APP-006-E4, S/G B STM LINE HI FLOW, alarm is in

-APP-006-F4, S/G C STM LINE HI FLOW, alarm is in

-Steam line pressure is 600 psig

Subsequently:

-The crew has transitioned to EPP-16

-AFW flow has been throttled to each S/G

-S/G conditions are now as follows:

A-40% WR level, Pressure is 355 PSIG and rising
B-35% WR level, Pressure is 300 PSIG and lowering
C-35% WR level, Pressure is 290 PSIG and lowering

EOP-E-2, FAULTED STEAM GENERATOR ISOLATION
EPP-16, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS

Which ONE (1) of the following completes the statements below?

Based off the initial conditions above, the plant ___(1)___ received an automatic SI signal. The crew will ___(2)___ .

A. (1) has **NOT**
(2) remain in EPP-16

B. (1) has
(2) remain on in EPP-16

C.✓ (1) has
(2) transition to EOP-E-2

D. (1) has **NOT**
(2) transition to EOP-E-2

The correct answer is C

A) Incorrect. An automatic SI signal has occurred. Plausible if the student does not know the setpoint for the high steam flow with low steamline pressure SI setpoint. Since the High Steam Line Flow alarms are already in, the only other thing they need is the low steam line pressure < 614 psig, which they have at 600 psig. The crew will transition to EOP-E-2, not continue on with EPP-16. Plausible if the student is unaware that Foldout D is in effect as soon as you enter EPP-16. Since one S/G pressure is rising, the foldout tells you to go to EOP-E-2.

B) Incorrect. An automatic SI signal has occurred. The crew will transition to EOP-E-2, not continue on with EPP-16. Plausible if the student is unaware that Foldout D is in effect as soon as you enter EPP-16. Since one S/G pressure is rising, the foldout tells you to go to EOP-E-2.

C) Correct. An automatic SI signal has occurred. The crew will transition to EOP-E-2. EPP-16 directs you to Foldout D as soon as you enter the procedure. This foldout has criteria that if S/G pressure is rising, transition to EOP-E-2.

D) Incorrect. An automatic SI signal has occurred. Plausible if the student does not know the setpoint for the high steam flow with low steamline pressure SI setpoint. Since the High Steam Line Flow alarms are already in, the only other thing they need is the low steam line pressure < 614 psig, which they have at 600 psig. The crew will transition to EOP-E-2. EPP-16 directs you to Foldout D as soon as you enter the procedure. This foldout has criteria that if S/G pressure is rising, transition to EOP-E-2.

Question: 86
Tier/Group: 2/1
K/A Importance Rating: SRO 4.6
K/A: 012 Reactor Protection System
G 2.1.20: Ability to interpret and execute procedure steps.
Reference(s): Sim/Plant design, EPP-16, Foldout D, APP-004-E1
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective lesson plan
Question Source: BANK
Question History: Has not been used on an NRC exam
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.12)
Comments:

This question meets the K/A because the student must determine from the information given if there has been an automatic SI signal (RPS). Then the student must execute procedure steps by determining what the correct transition is based off of Foldout Criteria. Determining which procedure to go with is what makes this an SRO level question.

HI STM FLO
LO TAVG/
LO SLP
SFGRD/TRIP

CAUSES:
1. Steam Line Break downstream of MSIVs and Check Valves.
2. CV High Pressure (HI-HI).

DEVICE:	SETPOINT:	LOCATION:
1. High Steam Flow:		
a. FC-474, FC-475	37.25% - 109% (Ramped from 20% to 100% Turbine PWR)	Hagan Rack #16, #24
b. FC-484, FC-485	37.25% - 109% (Ramped from 20% to 100% Turbine PWR)	Hagan Rack #16, #25
c. FC-494, FC-495	37.25% - 109% (Ramped from 20% to 100% Turbine PWR)	Hagan Rack #16, #25
2. Low T_{AVG}:		
a. TC-412E, TC-422E, TC-432E	543°F	Hagan Rack #1, #11, #14
3. Low Steam Line Pressure:		
a. PC-474A, PC-485A, PC-496A	614 psig	Hagan Rack #13, #17, #24
4. CV HI-HI pressure:		
a. PC-951A, PC-953A, PC-955A	10 psig	Hagan Rack #3, #12, #15
b. PC-950, PC-952, PC-954	10 psig	Hagan Rack #3, #12, #15

NOTE:
Any one of the following conditions will cause the associated function(s) to occur:
- High Steam Flow 1/2 flows on 2/3 lines in conjunction with either Low Steam Line Pressure on 2/3 steam lines OR Low T_{AVG} 2/3 channels will cause Steam Line Isolation and Safety Injection)
- CV HI-HI pressure (2/3 channels on 2/2 matrixes will cause a Safety Injection, a Steam Line Isolation and CV Spray Actuation)
- A Safety Injection Signal will cause a Reactor Trip

OPERATOR ACTIONS

1. **IF** Reactor has tripped, **THEN** GO TO EOP Network. .... _______
2. **IF** Reactor has NOT tripped **AND** either of the following conditions are present:
   - Transient operations in progress
   - Bistable status panel indicates reactor trip logic has been met

   THEN:

   a. TRIP Reactor. .... _______

   b. GO TO EOP Network. .... _______

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Open Foldout D**	
2.	**Perform The Following:** a. Reset SPDS b. Initiate monitoring of Critical Safety Function Status Trees	

CONTINUOUS USE

FOLDOUT D

(Page 1 of 1)

1. SI REINITIATION CRITERIA

   IF EITHER condition below occurs, THEN start both SI Pumps:

   - RCS Subcooling - LESS THAN 35°F [55°F]
   - PZR Level - CAN NOT BE MAINTAINED GREATER THAN 14% [31%]

2. SECONDARY INTEGRITY CRITERIA

   IF any S/G pressure rises at any time, THEN Go To EOP-E-2, Faulted Steam Generator Isolation, Step 1.

3. COLD LEG RECIRCULATION SWITCHOVER CRITERIA

   IF RWST level lowers to less than 27%, THEN Go To EPP-9, Transfer To Cold Leg Recirculation.

4. AFW SUPPLY SWITCHOVER CRITERIA

   IF CST level lowers to less than 10%, THEN switch to backup water supply using OP-402, Auxiliary Feedwater System.

5. RHR PUMP PIT ISOLATION CRITERIA

   IF ANY condition below occurs, THEN Go To EPP-24, Isolation Of Leakage In The RHR Pump Pit:

   - APP-001-D4, RHR PIT A HI-HI LEVEL - ILLUMINATED
   - APP-001-D5, RHR PIT B HI-HI LEVEL - ILLUMINATED
   - EITHER RTGB RHR Pit indication - GREATER THAN 24 INCHES

- END -

87. 061 G2.4.20 SRO 001

Given the following plant conditions:

-The plant is in MODE 3

-The crew is currently in EPP-28, LOSS OF ULTIMATE HEAT SINK

-CST Level is 8%

-Deepwell Pump D is the only available water supply to MDAFW Pump A

OP-402, AUXILIARY FEEDWATER SYSTEM
ATTACHMENT 6, DEEPWELL COOLING

Which ONE(1) of the following completes the statements below?

IAW EPP-28, total AFW flow to the S/G's is limited to ___(1)___ gpm. The crew will use ___(2)___ to align Deepwell Pump D.

A. (1) 195
(2) ATTACHMENT 6 of EPP-28

B. (1) 140
(2) ATTACHMENT 6 of EPP-28

C.✓ (1) 195
(2) OP-402

D. (1) 140
(2) OP-402

The correct answer is C

A) Incorrect. 195 gpm is correct per the NOTE in EPP-28. Attachment 6 is incorrect. Plausible because this attachment does allign deepwell cooling, however, it is to the EDG and not to the MDAFW pumps.

B) Incorrect. 140 gpm is incorrect. Plausible because this is the flow limit for one MDAFW pump alligned to one deepwell pump per OP-402. Attachment 6 is incorrect. Plausible because this attachment does allign deepwell cooling, however, it is to the EDG and not to the MDAFW pumps.

C) Correct. 195 gpm is correct per the NOTE in EPP-28. EPP-28 tells you to use OP-402 to allign an AFW suction source.

D) Incorrect. 140 gpm is incorrect. Plausible because this is the flow limit for one MDAFW pump alligned to one deepwell pump per OP-402. EPP-28 tells you to use OP-402 to allign an AFW suction source.

Question: 87
Tier/Group: 2/1
K/A Importance Rating: SRO 4.3
K/A: 061 Auxiliary / Emergency Feedwater (AFW) System
G 2.4.20: Knowledge of the operational implications of EOP warnings, cautions, and notes.
Reference(s): Sim/Plant design, EPP-28 and its basis document, OP-402
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 4 of EPP-28 lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)
Comments:

This question meets the K/A because the candidate must know that the operational implication of the NOTE in EPP-28 is that while using Deepwell Pump D, total AFW flow is limited to 195 gpm. The student must determine which procedure to use to align the Deepwell pump and this is what makes this an SRO level question.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

Subsequent steps may require access to Non-vital Areas. Access to these areas prior to nullification of the threat could cause loss of personnel.

****

NOTE

- FRPs, EPP Foldouts, EPP-1, AOP-014, AOP-020, AND AOP-022 are NOT applicable for this event. Transition OR use of these procedures should NOT be made unless otherwise directed in this procedure.
- Any time MCC-6 is deenergized, coordination of field activities will require the use of portable radios OR cell phones. It is recommended that the DS Radios OR Fire Protection radios be used for this communication.

1. Check Reason For Entry - LOSS OF INTAKE STRUCTURE

Go to the Section for the current Plant Mode:

MODE	SECTION
3	Section E
4	Section F
5	Section G
6	Section H
Defueled	Section I

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

2. Go To The Step OR Section For The Current Plant Mode:

MODE	STEP/SECTION
3	3
4	Section A
5	Section B
6	Section C
Defueled	Section D

3. Verify The MSIVs AND MSIV BYPs Valves - CLOSED

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

The calculated maximum times the following components may operate without cooling to preclude adverse system effects is as follows:

- EDGs - 40 minutes
- CCW Heat Exchanger - Less than 60 minutes

****

NOTE

SW pressure at the CCW Heat Exchanger outlet must be maintained greater than 18 psig.

* 4. Check Status Of Off-Site Power - LOST

<u>IF</u> Off-Site Power is lost, <u>THEN</u> Go To Step 5.

Perform the following:

a. Dispatch an Operator to perform Attachment 6, Deepwell Cooling for one of the available EDGs.

b. <u>WHEN</u> Attachment 6 is complete, <u>THEN</u> dispatch an Operator to perform Attachment 7, Establishing CCW Cooling, while continuing with this procedure.

c. Verify V6-16C, SW ISOLATION TO TURBINE BUILDING - CLOSED

d. <u>WHEN</u> Attachment 7 is complete, <u>THEN</u> Monitor CCW Temperature <u>AND</u> maintain Below 125°F By Throttling SW-739 <u>AND</u> SW-740 In Equal Increments, while continuing with this procedure.

e. Observe the <u>NOTE</u> prior to Step 21 and Go To Step 21.

incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
46.	**Transfer AFW To The MDAFW Pump As Follows:** a. Verify ALL V1-8 Valves - CLOSED • MS-V1-8A, SG "A" STM SUPPLY TO STM DRIVEN AFW PUMP • MS-V1-8B, SG "B" STM SUPPLY TO STM DRIVEN AFW PUMP • MS-V1-8C, SG "C" STM SUPPLY TO STM DRIVEN AFW PUMP b. Verify ALL V2-14 Valves - CLOSED • AFW-V2-14A, STEAM DRIVEN FWP FDWTR DSCHG TO SG "A" • AFW-V2-14B, STEAM DRIVEN FWP FDWTR DSCHG TO SG "B" • AFW-V2-14C, STEAM DRIVEN FWP FDWTR DSCHG TO SG "C" c. Establish AFW flow using the available MDAFW Pump d. Maintain S/G levels between 39% <u>AND</u> 50% using the MDAFW Pump	
*47.	Check CST Level - LESS THAN 10%	<u>WHEN</u> CST Level is less than 10% <u>THEN</u> observe the <u>NOTE</u> prior to Step 48 and perform Step 48. Observe the <u>NOTE</u> prior to Step 49 and Go To Step 49.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

When the MDAFW Pumps are being supplied via Deepwell Pump "D" the total AFW flow to the Steam Generators is limited to 195 gpm.

correct

48. Align SW To The AFW Pump Suction Using OP-402, Auxiliary Feedwater System

NOTE

Attachment 14 should be performed in 4 hours or less.

49. Contact Plant Operations Staff To Establish A Mission to Perform Attachment 14, WCCU Cooling

50. Go To Attachment 13, RCS Cooldown To 300°F

51. Contact Plant Operations Staff To Initiate A TSC Repair Mission To Restore At Least 7,500 GPM SW Flow Capability

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*52.	Check Ultimate Heat Sink - RESTORED	When the Ultimate Heat Sink is restored, <u>THEN</u> Go To Step 53. Go To Step 51.
53.	Return To Procedure And Step In Effect	

\- END -

INCORRECT

8.4.2.2 (Continued)

INIT VERI

NOTE: When Deepwell Water is being supplied to the AFW Pumps, the MDAFW Pumps should have cooling water to the oil coolers and sealing water to the seals supplied from the Firemain VIA direction from the EOP Network. If this is **NOT** done, lube oil temperature should **NOT** exceed 140• F (IAW Tech Manual) **AND** no damage to the seals will occur.

CAUTION

Exceeding the Maximum flow rates below could cause the AFW pumps to trip on low discharge pressure.

Number of running deepwell pumps	1 (200 gpm)	2 (400 gpm)	3 (600 gpm)
SDAFW Pump (ONLY) (145 gpm Recirc + 105.2 gpm leakoff)	N/A	145 gpm	345 gpm
2 MDAFW Pumps (120 gpm Recirc)	80 gpm	280 gpm	480 gpm NOTE 1
1 MDAFW Pump (60 gpm Recirc)	140 gpm	325 gpm NOTE 1	325 gpm NOTE 1

NOTE 1: Flow no more than 325 gpm per MDAFW pump to prevent trip on overcurrent.

d. Based on the above limitations, **START** AFW Pumps as follows:

1) **IF** SDAFW Pump is to be used, **THEN PERFORM** the following:

.a) **REMOVE** cap **AND OPEN** AFW-7, SDAFW PUMP SUCTION VENT. ____

.b) **WHEN** a solid stream of water issues, **THEN CLOSE** AFW-7 **AND INSTALL** the cap. ____ ____

88. 063 A2.02 SRO 001

Given the following plant conditions:

-The plant is in MODE 4

-Battery Charger A is in service

-APP-036-F10, BATT RM A/B HI/LO TEMP, alarms

-It is determined that Battery A's representative cells have an average electrolyte temperature of 65°F

3.8.4, DC SOURCES-OPERATING
3.8.5, DC SOURCES-SHUTDOWN
3.8.6, BATTERY CELL PARAMETERS

Which ONE (1) of the following completes the statements below?

Per APP-036-F10, the SRO will direct the OAO to verify the ___(1)___ is operating.
The SRO is required to enter LCO(s) ___(2)___ .

A. (1) alternate heater
(2) 3.8.5 only

B. (1) air conditioning
(2)3.8.5 only

C✓ (1) alternate heater
(2)3.8.4 and 3.8.6

D. (1) air conditioning
(2)3.8.4 and 3.8.6

The correct answer is C

A) Incorrect. Per APP-036-F10, you do verify the heater is operating because temperature is below 69ºF. LCO 3.8.5 only is incorrect. Plausible if the student is unsure of the MODE requirements for LCO 3.8.5.

B) Incorrect. Verifying the air conditioning is operating is incorrect. Plausible because APP-036-F10 does have you check the air conditioning is operating, however, its only if temperature is high, above 86ºF. LCO 3.8.5 only is incorrect. Plausible if the student is unsure of the MODE requirements for LCO 3.8.5.

C) Correct. Per APP-036-F10, you do verify the heater is operating because temperature is below 69ºF. You are initially in LCO 3.8.6 for the battery cell parameter being <67ºF. The battery should be declared inoperable. Based off this, the candidate must determine if they should enter LCO 3.8.4 or 3.8.5.

D) Incorrect. Verifying the air conditioning is operating is incorrect. Plausible because APP-036-F10 does have you check the air conditioning is operating, however, its only if temperature is high, above 86ºF. You are initially in LCO 3.8.6 for the battery cell parameter being <67ºF. The battery should be declared inoperable. Based off this, the candidate must determine if they should enter LCO 3.8.4 or 3.8.5.

Question: 88
Tier/Group: 2/1
K/A Importance Rating: SRO 3.1
K/A: 063 DC Electrical Distribution System
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.02: Battery capacity as it is affected by discharge rate
Reference(s): Sim/Plant design, LCO 3.8.4, 3.8.6
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 15 of DC lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Comments:

This question meets the K/A because the student must determine/predict the status(operable/inoperable)of the batteries due to it's parameters. Based off of declaring the batteries inoperable, they will decide which Tech Specs to use to control the consequences of declaring the batteries inoperable.

Correct

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is ≥ 125.7 V on float charge.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 67°F. OR One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

Correct

ALARM

BATT RM A/B HI/LO TEMP

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Loss of either Air Conditioning or Heat in the Battery A&B Room

OBSERVATIONS

1. None Applicable

ACTIONS

CK (✓)

1. **DISPATCH** an operator to the Battery A&B Room to check the temperature of the Temperature Indicating Switch TIS-4361. _____
2. **IF** the temperature is below 69°F, **THEN PERFORM** the following:
   1) **CHECK** to see which heater is selected at the HVAC Panel. _____
   2) **VERIFY** the selected heater is operating. _____
   3) **IF** the selected heater will **NOT** operate, **THEN SWITCH** to the alternate heater. _____
   4) **VERIFY** the alternate heater operates. _____
   5) **CHECK CLOSED** breaker MCC-2 (7M), BATTERY ROOM HVAC/HEATER CONTROL PANEL. _____
   6) **CHECK CLOSED** individual heater breakers (Breaker A and Breaker B located inside the lower control panel). _____
   7) **IF** neither heater operates, **THEN OBTAIN** a temporary source of heat. _____
   8) **IF** the temperature drops to 68°F or lower, **THEN INITIATE** action to trouble shoot and repair the heaters **AND NOTIFY** Engineering. _____
   9) **IF** Station Battery A **OR** B pilot cell temperature drops below 67°F, **THEN DECLARE** the affected battery inoperable. _____
3. **IF** the temperature is above 86°F, **THEN VERIFY** the air conditioning is operating. _____
4. For High **OR** Low temperature, **CHECK** the temperature in the Battery Room at least every 4 hours **AND RECORD** the temperature in the Control Operator's Log until the alarm resets. (ACR 94-00022) _____

Correct

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for Train A and Train B batteries shall be within the limits of Table 3.8.6-1 and average electrolyte temperature of representative cells shall be within limit.

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Category A or B limits.	A.1 Verify pilot cell electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	<u>AND</u>	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours <u>AND</u> Once per 7 days thereafter
	<u>AND</u>	
	A.3 Restore battery cell parameters to Category A and B limits of Table 3.8.6-1.	31 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 67°F. OR One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 24 hours after a battery discharge < 110 V AND Once within 24 hours after a battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 67°F.	92 days

Table 1.1-1 (page 1 of 1)
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥ 0.99	≤ 5	NA
3	Hot Standby	< 0.99	NA	≥ 350
4	Hot Shutdown[b]	< 0.99	NA	$350 > T_{avg} > 200$
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources-Shutdown

LCO 3.8.5 DC electrical power subsystem shall be OPERABLE to support the DC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems-Shutdown."

APPLICABILITY: MODES 5 and 6, and
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND	Immediately
	A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.5, and SR 3.8.4.6. -- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.3 SR 3.8.4.5 SR 3.8.4.2 SR 3.8.4.4 SR 3.8.4.6	In accordance with applicable SRs

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration. AND A.2.4 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.4, SR 3.8.4.5, and SR 3.8.4.6. --- For DC sources required to be OPERABLE, the following SRs are applicable: SR 3.8.4.1 SR 3.8.4.3 SR 3.8.4.5 SR 3.8.4.2 SR 3.8.4.4 SR 3.8.4.6	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems – Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6, and
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC instrument bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.4 Initiate actions to restore required AC, DC, and AC instrument bus electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.5 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.10.1	------------------NOTE------------------ Actual voltage measurement is not required for the AC vital buses supplied from constant voltage transformers. -- Verify correct breaker alignments and voltage to required AC, DC, and AC instrument bus electrical power distribution subsystems.	7 days

89. 076 A2.01 SRO 001

Given the following plant conditions:

-A plane crashed into the Intake Structure

-The plant was manually tripped

-The SUT tripped

-The CRS has entered EPP-28, EPP-28, LOSS OF ULTIMATE HEAT SINK

ATTACHMENT 6, DEEPWELL COOLING
ATTACHMENT 7, ESTABLISHING CCW COOLING

Which ONE (1) of the following completes the statements below?

The maximum time the crew has to restore cooling to an EDG to preclude adverse effects is ___(1)___ minutes. In order to provide cooling to at least one EDG, the CRS will direct ATTACHMENT ___(2)___ to be performed.

A. (1) 60
(2) 6

B. (1) 60
(2) 7

C.✓ (1) 40
(2) 6

D. (1) 40
(2) 7

The correct answer is C

A) Incorrect. 60 minutes is incorrect. Plausible because within the same caution, the caution states that the max time CCW heat exchangers can operate without cooling is 60 minutes. Attachement 6 is correct. This will align deep well cooling to the running EDG.

B) Incorrect. 60 minutes is incorrect. Plausible because within the same caution, the caution states that the max time CCW heat exchangers can operate without cooling is 60 minutes. Attachement 7 is incorrect. Plausible because you will utilize attachement 7 in epp-28, however, it is to establish cooling to the CCW heat exchangers and not the EDG. While performing attachment 7, there are multiple valves that are aligned in the EDG room.

C) Correct. 40 minutes is the maximum time the EDG may operate without cooling to preclude adverse system effects. The CRS will direct attachment 6 of epp-28 to restore cooling to the EDG using the "D" deepwell pump.

D) Incorrect. 40 minutes is the maximum time the EDG may operate without cooling to preclude adverse system effects. Attachement 7 is incorrect. Plausible because you will utilize attachement 7 in epp-28, however, it is to establish cooling to the CCW heat exchangers and not the EDG. While performing attachment 7, there are multiple valves that are aligned in the EDG room.

Question: 89
Tier/Group: 2/1
K/A Importance Rating: SRO 3.7
K/A: 076 Service Water System (SWS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.01: Loss of SWS
Reference(s): Sim/Plant design, EPP-28
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 5 of EPP-28 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45/3 / 45/13)
Comments:

This question meets the K/A because the student must predict the impact a total loss of service water has on the EDG. The student must then determine which attachment to use to restore cooling to the EDG. Determining which attachment to use is what makes this an SRO level question.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

correct

The calculated maximum times the following components may operate without cooling to preclude adverse system effects is as follows:

- EDGs - 40 minutes
- CCW Heat Exchanger - Less than 60 minutes incorrect

****

NOTE

SW pressure at the CCW Heat Exchanger outlet must be maintained greater than 18 psig.

* 4. **Check Status Of Off-Site Power - LOST**

IF Off-Site Power is lost, THEN Go To Step 5.

Perform the following:

a. Dispatch an Operator to perform Attachment 6, Deepwell Cooling for one of the available EDGs.

b. WHEN Attachment 6 is complete, THEN dispatch an Operator to perform Attachment 7, Establishing CCW Cooling, while continuing with this procedure.

c. Verify V6-16C, SW ISOLATION TO TURBINE BUILDING - CLOSED

d. WHEN Attachment 7 is complete, THEN Monitor CCW Temperature AND maintain Below 125°F By Throttling SW-739 AND SW-740 In Equal Increments, while continuing with this procedure.

e. Observe the NOTE prior to Step 21 and Go To Step 21.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**Establish Cooling To ONE EDG Using Deepwell Pump "D" As Follows:**	
	a. Check EDG "B" - AVAILABLE	a. IF "A" EDG is available, THEN dispatch an Operator to perform Attachment 6, Deepwell Cooling, for "A" EDG AND Go To Step 6 IF neither EDG is available, THEN Go To Section J, SBO With No Service Water.
	b. Dispatch an Operator to perform Attachment 6, Deepwell Cooling, for "B" EDG	Correct
6.	**On the EDG To Remain Running Verify The Following Components:**	
	a. 1 CCW Pump - RUNNING	
	b. 1 Charging Pump - RUNNING	
	c. MDAFW Pump - STOPPED	
	d. BOTH SW Pumps - STOPPED	
	e. SWBP - STOPPED	
	f. BOTH HVH Units - STOPPED	

NOTE

If a valve was open when power was lost on the bus it may be assumed to remain open. There are no spurious valve operations for this event.

7. **Verify The Following CCW Valves - OPEN:**

- CC-716A, CCW TO RCP
- CC-716B, CCW TO RCP
- CC-735, THERM BAR OUT ISO
- FCV-626, THERM BAR FLOW CONT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	Check Attachment 6 - COMPLETE	<u>WHEN</u> Attachment 6 is complete, <u>THEN</u> Go To Step 9.
9.	Check MCC-5 - ENERGIZED	Go To Step 11.
10.	Locally Verify The Following Breakers At MCC-5 - CLOSED: • BATTERY CHARGER A-1 (CMPT-5B) • BATTERY CHARGER A (CMPT-11CR)	
11.	Check MCC-6 - ENERGIZED	Go To Step 13.
12.	Locally verify the following breakers at MCC-6 - CLOSED: • BATTERY CHARGER B-1 (CMPT-14M) • BATTERY CHARGER B (CMPT-15FR)	
13.	Dispatch An Operator To Perform Attachment 7, Establishing CCW Cooling	Incorrect
14.	Verify V6-16C, SW ISOLATION TO TURBINE BUILDING - CLOSED	
*15.	Check Attachment 7 - COMPLETED	<u>WHEN</u> Attachment 7 has been completed, <u>THEN</u> observe the <u>NOTE</u> prior to Step 16 and perform Step 16. Observe the <u>NOTE</u> prior to Step 17 and Go To Step 17.

<u>NOTE</u>

SW pressure at the CCW Heat Exchanger outlet must be maintained greater than 18 psig.

16.	Monitor CCW Temperature <u>AND</u> Maintain Below 125°F By Throttling SW-739 <u>AND</u> SW-740 In Equal Increments	

90. 103 A2.03 SRO 002

Given the following plant conditions:

-A main steam line break inside CV results in a Safety Injection and CV Spray actuation

-EOP-E-0, REACTOR TRIP OR SAFETY INJECTION, has been implemented

-CV pressure is currently 14 psig

-CC-735, THERM BAR OUT ISO, is open

SUPPLEMENT A, SAFETY INJECTION COMPONENT ALIGNMENT
SUPPLEMENT B, PHASE B AND CV SPRAY COMPONENT ALIGNMENT

Which ONE (1) of the following completes the statement below?

Based on current plant conditions, CC-735 __(1)__ in the correct position and the CRS may direct the use of __(2)__ to verify it's position.

A. (1) is
(2) SUPPLEMENT A

B. (1) is **NOT**
(2) SUPPLEMENT A

C. (1) is
(2) SUPPLEMENT B

D✓ (1) is **NOT**
(2) SUPPLEMENT B

The correct answer is D

A) Incorrect. CC-735 is in the correct position is incorrect. Plausible if the student does not know that CC-735 is shut on a Phase B signal. Also plausible if they student does not know that at this CV pressure, a Phase B signal is automatically generated. Supplement A is incorrect. Plausible if the student believes that CC-735 gets shut by a phase A signal.

B) Incorrect. CC-735 is not in its correct position is correct. It should be shut since the plant received a Phase B signal based of CV pressure being > 10 psig. Supplement A is incorrect. Plausible if the student believes that CC-735 gets shut by a phase A signal.

C) Incorrect. CC-735 is in the correct position is incorrect. Plausible if the student does not know that CC-735 is shut on a Phase B signal. Also plausible if they student does not know that at this CV pressure, a Phase B signal is automatically generated. Supplement B is correct. This supplement gives you specific valve positions for all Phase B valves.

D) Correct. CC-735 is not in its correct position is correct. It should be shut since the plant received a Phase B signal based of CV pressure being > 10 psig. Supplement B is correct. This supplement gives you specific valve positions for all Phase B valves.

Question: 90
Tier/Group: 2/1
K/A Importance Rating: SRO 3.8
K/A: 103 Containment System
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations
A2.03: Phase A and B isolation
Reference(s): Sim/Plant design, EOP-E-0, Supplement B
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 2
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.5 / 43.5 / 45.3 / 45.13)
Comments:

This question meets the K/A because the student must determine which isolation signal (phase A or B) did not function properly based off the valve position in the stem. Then, must determine which procedure/portion of a procedure with which to use to verify the specific phase B valves are shut. Determining which procedure or section of a procedure is what makes this an SRO level question.

CONTINUOUS USE

Supplement B

Phase B And CV Spray Component Alignment

(Page 1 of 1)

1. To establish Phase B Containment Isolation, verify the following valves - CLOSED

   a. RCP Cooling

      - CVC-381, SEAL WTR RTRN ISO
      - FCV-626, THERM BAR FLOW CONT
      - CC-735, THERM BAR OUT ISO
      - CC-716A, CCW TO RCP ISO
      - CC-716B, CCW TO RCP ISO
      - CC-730, BRG OUTLET ISO

   b. MSIV AND MSIV BYPs - CLOSED

2. To establish CV Spray, perform the following:

   a. Verify valves positioned as follows:

      - SI-844A, PUMP A INLET - OPEN
      - SI-844B, PUMP B INLET - OPEN
      - SI-845A, SAT DISCH - OPEN
      - SI-845B, SAT DISCH - OPEN
      - SI-845C, SAT THROTTLING - THROTTLED TO APPROXIMATELY 12 GPM
      - SI-880A, PUMP A DISCH - OPEN
      - SI-880B, PUMP A DISCH - OPEN
      - SI-880C, PUMP B DISCH - OPEN
      - SI-880D, PUMP B DISCH - OPEN

   b. Return to procedure and step in effect.

- END -

Incorrect

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

(Page 1 of 14)

1. Go to the appropriate step from the table below:

FUNCTION	STEP
To Establish Cold Leg Injection	Step 2
To Establish Hot Leg Injection	Step 4
To Establish Phase A Containment Isolation	Step 6
To Establish Containment Ventilation Isolation	Step 8
To Establish Motor Driven AFW Pump alignment	Step 10
To Establish Feedwater Isolation	Step 12
Shift Control Room Ventilation to Emergency Pressurization Mode	Step 14

Incorrect

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

NOTE

Fuse pullers are located with AOP/EOP/DSP Tool Kits.

6. To establish Phase A Containment Isolation, perform the following verification:

   a. Verify the following letdown valves - CLOSED

      - CVC-200A, LTDN ORIFICE
      - CVC-200B, LTDN ORIFICE
      - CVC-200C, LTDN ORIFICE
      - CVC-204A, LTDN LINE ISO
      - CVC-204B, LTDN LINE ISO

   b. Check IA PCV-1716, INSTRUMENT AIR ISO TO CV - CLOSED

   c. IF PCV-1716 does NOT indicate closed, THEN obtain fuse pullers AND fail PCV-1716 closed by removing the following fuses on Auxliary Panel GC:

      - Aux. Panel GC, circuit 32, fuse 67
      - Aux. Panel GC, circuit 32, fuse 68

(CONTINUED NEXT PAGE)

Incorrect

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

(Page 8 of 14)

6. (CONTINUED)

d. Verify the following Sample Line valves - CLOSED

- PS-956A, PZR STEAM SPACE SAMPLE
- PS-956B, PZR STEAM SPACE SAMPLE
- PS-956C, PZR LIQUID SPACE SAMPLE
- PS-956D, PZR LIQUID SPACE SAMPLE
- PS-956E, RCS HOT LEG SAMPLE
- PS-956F, RCS HOT LEG SAMPLE
- PS-956G, ACCUM SAMPLE LINE AIR OPERATED ISOLATION
- PS-956H, ACCUM SAMPLE LINE AIR OPERATED ISOLATION

e. Verify the following PRT valves - CLOSED

- RC-516, PRT TO GAS ANALYZER
- RC-553, PRT TO GAS ANALYZER
- RC-550, PRT NITROGEN SUPPLY
- RC-519A AND B, PRIMARY WATER TO PRESSURIZER RELIEF TANK

f. Verify CC-739, EXCESS LTDN HX OUTLET - CLOSED

NOTE

Local operation of the valve below is via a reverse acting handwheel.

g. Verify SI-855, ACCUMULATOR NITROGEN SUPPLY - CLOSED

(CONTINUED NEXT PAGE)

Incorrect

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

6. (CONTINUED)

h. Verify the following Fire Protection System valves - CLOSED

- FP-248, ELECT. PENETRATION CV ISOLATION
- FP-249, ELECT. PENETRATION CV ISOLATION
- FP-256, RCP SPRINKLER ISOLATION VALVE
- FP-258, RCP SPRINKLER ISOLATION VALVE

i. Verify the following Waste Disposal System valves - CLOSED

- WD-1721, RCDT PUMP DISCHARGE LINE AUTO ISOLATION
- WD-1722, RCDT PUMP DISCHARGE LINE AUTO ISOLATION
- WD-1723, CONTAINMENT SUMP PUMP DISCHARGE AUTO ISOLATION
- WD-1728, CONTAINMENT SUMP PUMP DISCHARGE AUTO ISOLATION
- WD-1786, RCDT VENT
- WD-1787, RCDT VENT
- WD-1789, RCDT SAMPLE LINE TO GAS ANALYZER
- WD-1794, RCDT SAMPLE LINE TO GAS ANALYZER

(CONTINUED NEXT PAGE)

INCORRECT

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

6. (CONTINUED)

j. Check the following S/G blowdown and sample valves - CLOSED

- SGB-FCV-1930A, STEAM GENERATOR A BLOWDOWN LINE
- SGB-FCV-1930B, STEAM GENERATOR A BLOWDOWN LINE
- SGB-FCV-1931A, STEAM GENERATOR B BLOWDOWN LINE
- SGB-FCV-1931B, STEAM GENERATOR B BLOWDOWN LINE
- SGB-FCV-1932A, STEAM GENERATOR C BLOWDOWN LINE
- SGB-FCV-1932B, STEAM GENERATOR C BLOWDOWN LINE
- SGB-FCV-1933A, STEAM GENERATOR A SAMPLE LINE
- SGB-FCV-1933B, STEAM GENERATOR A SAMPLE LINE
- SGB-FCV-1934A, STEAM GENERATOR B SAMPLE LINE
- SGB-FCV-1934B, STEAM GENERATOR B SAMPLE LINE
- SGB-FCV-1935A, STEAM GENERATOR C SAMPLE LINE
- SGB-FCV-1935B, STEAM GENERATOR C SAMPLE LINE

k. IF any S/G blowdown and sample valves do NOT indicate closed, THEN de-energize the R-19 skid (A/B/C) for the affected valves.

l. Verify the following Radiation Monitoring System valves CLOSED

- RMS-1, CONTAINMENT OUTLET TO R-11 AND R-12
- RMS-2, R-11 AND R-12 INLET ISOLATION
- RMS-3, CONTAINMENT INLET FROM R-11 AND R-12
- RMS-4, R-11 AND R-12 OUTLET ISOLATION

(CONTINUED NEXT PAGE)

Incorrect

CONTINUOUS USE

Supplement A

Safety Injection Component Alignment

(Page 11 of 14)

6. (CONTINUED)

   m. Check the IVSW Sytem automatic header isolation valves OPEN.

      - PCV-1922A, AUTOMATIC HEADER PRESSURE CONTROL VALVE
      - PCV-1922b, AUTOMATIC HEADER PRESSURE CONTROL VALVE

   n. IF PCV-1922A OR PCV-1922B is closed, THEN fail air to the affected valve(s) by performing the following:

      1) Isolate the affected valve(s) air supply isolation valve.

      2) Open the affected valve(s) air regulator petcock to bleed air from the valve.

7. Return to procedure and step in effect.

8. To establish Containment Ventilation Isolation, verify the following valves - CLOSED

   - V12-6, CONT PURGE VALVES
   - V12-7, CONT PURGE VALVES
   - V12-8, CONT PURGE VALVES
   - V12-9, CONT PURGE VALVES
   - V12-10, CONTAINMENT PRESSURE RELIEF
   - V12-11, CONTAINMENT PRESSURE RELIEF
   - V12-12, CONTAINMENT VACUUM RELIEF
   - V12-13, CONTAINMENT VACUUM RELIEF

9. Return to procedure and step in effect.

91. 001 A2.13 SRO 001

Given the following plant conditions:

-APP-004-E4, CV HI PRESS SFGRD/TRIP, flashes and is confirmed valid

-The reactor is at 100% RTP

-CET's are 1225°F and rising

FRP-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS
FRP-C.1, RESPONSE TO INADEQUATE CORE COOLING
SACRM-1, SEVERE ACCIDENT CONTROL ROOM MANAGEMENT INITIAL RESPONSE

Which ONE (1) of the following completes the statements below?

IAW FRP-S.1, the crew unsuccessfully tried tripping the reactor by opening the ___(1)___ . The CRS is required to transition to ___(2)___ .

A.✓ (1) Generator A & B Circuit Breakers
(2) SACRM-1

B. (1) Generator A & B Circuit Breakers
(2) FRP-C.1

C. (1) feeder breaker to 480V busses 2B and 3
(2) SACRM-1

D. (1) feeder breaker to 480V busses 2B and 3
(2) FRP-C.1

The correct answer is A

A) Correct. The immediate actions of FRP-S.1 has you trip the Generator A & B Circuit Breakers. This is an effort to de-energize the control rods and let them fall into the core. The CRS will transition to SACRM-1 because CET's are >1200°F.

B) Incorrect. The immediate actions of FRP-S.1 has you trip the Generator A & B Circuit Breakers. This is an effort to de-energize the control rods and let them fall into the core. FRP-C.1 is incorrect. Plausible because you would have a red on CSFST for Core Cooling because CET's are >1200°F which sends you to FRP-C.1. You would not go there because Criticality is a higher priority red ball on CSFST's.

C) Incorrect. Opening the feeder breaker to 480V busses 2B and 3 is incorrect. Plausible because this would de-energize the Rod Drive Motor Generators. However, FRP-S.1 has you trip the feeder breaker FROM 480V 2b and 3 to the Rod Drive Motor Generators. The CRS will transition to SACRM-1 because CET's are >1200°F.

D) Incorrect. Opening the feeder breaker to 480V busses 2B and 3 is incorrect. Plausible because this would de-energize the Rod Drive Motor Generators. However, FRP-S.1 has you trip the feeder breaker FROM 480V 2b and 3 to the Rod Drive Motor Generators. FRP-C.1 is incorrect. Plausible because you would have a red on CSFST for Core Cooling because CET's are >1200°F which sends you to FRP-C.1. You would not go there because Criticality is a higher priority red ball on CSFST's.

Question: 91
Tier/Group: 2/2
K/A Importance Rating: SRO 4.6
K/A: 001 Control Rod Drive System
A2: Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.13: ATWS
Reference(s): Simulator/plant design, FRP-S.1
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 5 of FRP-S.1 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 41.5/43.5/45.3/45.13
Comments:

This question meets the K/A because the stduent must know that the crew will try to de-energize the control rods because of the ATWS event in progress. Because they couldn't trip the rods, the CRS must know the procedure that the crew will use to mitigate this event.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

RCPs should NOT be tripped with Reactor Power GREATER THAN 5%.

****

> NOTE
>
> Steps 1 AND 2 are Immediate Action steps.

1. **Check REACTOR TRIP As Follows:**

   - REACTOR TRIP MAIN AND BYP BKRs - OPEN
   - Rod Position indication - ZERO
   - Rod Bottom lights - ILLUMINATED
   - Neutron Flux - LOWERING

Perform the following:

a. Depress both Reactor Trip Pushbuttons.

b. IF Reactor Trip Breakers will NOT open, THEN perform the following:

   1) Insert Control Rods.

   2) Dispatch an operator to the MG SET Room to trip the following breakers:

      - REACTOR TRIP BREAKER A
      - REACTOR TRIP BREAKER B
      - GENERATOR A CIRCUIT BREAKER
      - GENERATOR B CIRCUIT BREAKER

   3) Dispatch an operator to 480V Busses 2B and 3 to trip the following breakers:

      - ROD DRIVE MOTOR GENERATOR SET A
      - ROD DRIVE MOTOR GENERATOR SET B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

<u>CAUTION</u>

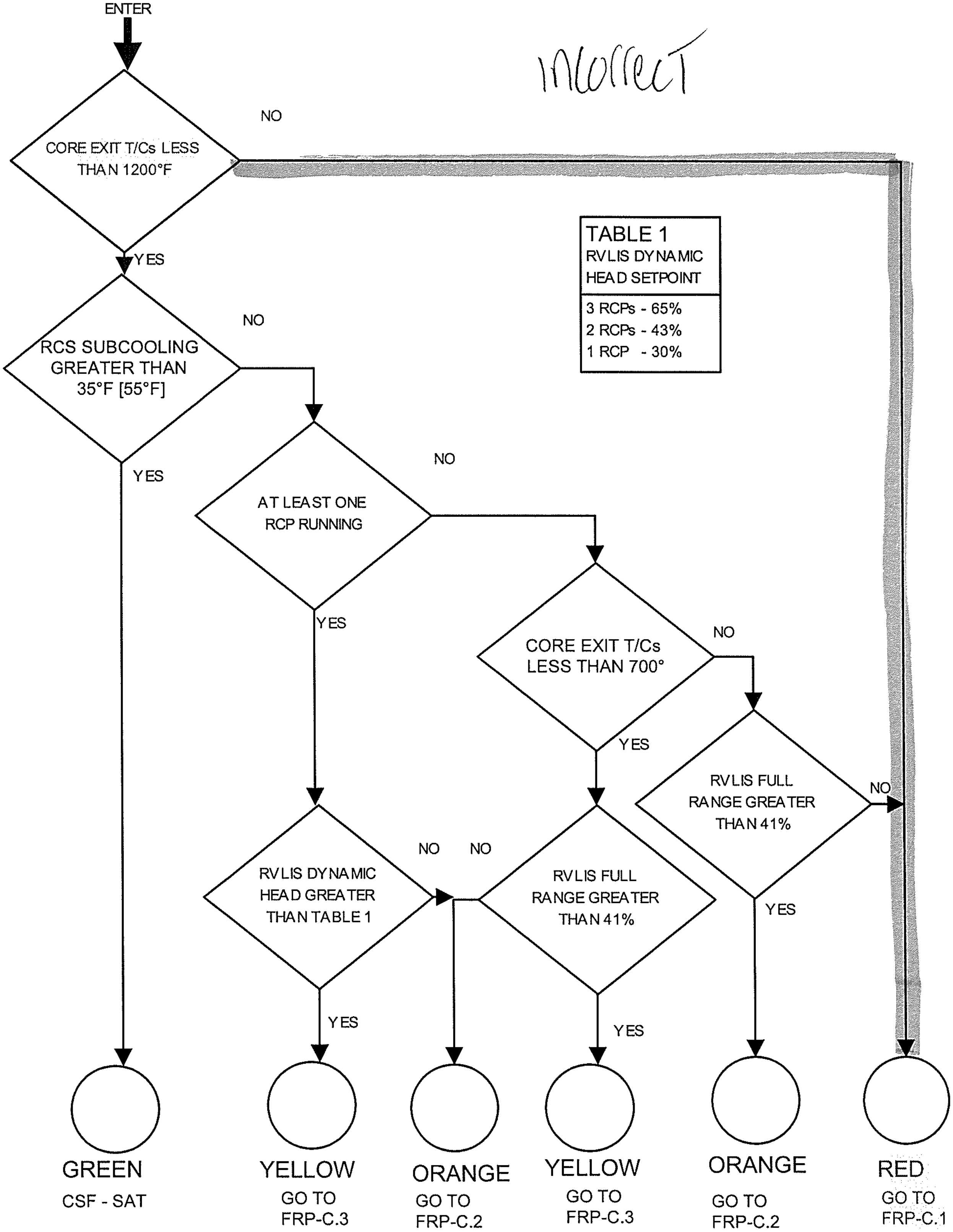
A loss of DC power may occur if the DC busses are at maximum load and the battery chargers are not restarted within 60 minutes of a loss of all AC power.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
20.	**Check Battery Chargers - IN SERVICE:**	
	• APP-036-D1, BATT CHARGER A/A1 TROUBLE - EXTINGUISHED	• Locally Check In Service Battery Charger A <u>OR</u> A-1 - OPERATING <u>IF</u> In Service Battery Charger is <u>NOT</u> operating, <u>THEN</u> restart tripped BATTERY CHARGER A <u>OR</u> A-1, using OP-601 DC Supply System, while continuing with this procedure.
	<u>AND</u>	
	• APP-036-D2, BATT CHARGER B/B1 TROUBLE - EXTINGUISHED	• Locally Check In Service Battery Charger B <u>OR</u> B-1 - OPERATING. <u>IF</u> In Service Battery Charger is <u>NOT</u> operating, <u>THEN</u> restart tripped BATTERY CHARGER B <u>OR</u> B-1, using OP-601 DC Supply System, while continuing with this procedure.
21.	**Check Core Exit T/Cs - LESS THAN 1200°F**	<u>IF</u> Core Exit T/Cs are greater than 1200°F <u>AND</u> rising, <u>THEN</u> Go To SACRM-1, Severe Accident Control Room Management - Initial Response. <u>IF</u> Core Exit T/Cs are greater than 1200°F <u>AND</u> lowering, <u>THEN</u> Go To Step 22.

CSF-2, CORE COOLING

incorrect



92. 068 A2.04 SRO 001

Given the following plant conditions:

-An approved radioactive liquid waste release is in progress

-Subsequently, R-18, LIQUID EFFLUENT WASTE DISPOSAL, is reading above its alarm setpoint

-The release is still in progress

Which ONE(1) of the following completes the statements below?

R-18 reading above the alarm setpoint ___(1)___ have terminated the release. Per the ODCM, the release may continue provided ___(2)___ .
(REFERENCES PROVIDED)

A.✓ (1) should
(2) two independent samples are analyzed and two facility staff independently verify the release rate calculations and discharge line valving

B. (1) should **NOT**
(2) two independent samples are analyzed and two facility staff independently verify the release rate calculations and discharge line valving

C. (1) should
(2) the flow rate is estimated at lease once per 4 hours during actual releases

D. (1) should **NOT**
(2) the flow rate is estimated at lease once per 4 hours during actual releases

The correct answer is A.

A. Correct. Since the monitor reading has exceeded its alarm setpoint, RCV-018 should have shut automatically. Per ODCM Table 2.6-1, two independent samples are analyzed and two facility staff independently verify the release rate calculations and discharge line valving.

B. Incorrect. R-18 should NOT have terminated the release is incorrect. Plausible if the student does not know that R-18 alarming automatically shuts RCV-018. Per ODCM Table 2.6-1, two independent samples are analyzed and two facility staff independently verify the release rate calculations and discharge line valving.

C. Incorrect. Since the monitor reading has exceeded its alarm setpoint, RCV-018 should have shut automatically. The flow rate being estimated every 4 hours is incorrect. Plausible if the student uses the wrong portion of the table from Table 2.6-1. This part is if the flow meter is not in service.

D. Incorrect. R-18 should NOT have terminated the release is incorrect. Plausible if the student does not know that R-18 alarming automatically shuts RCV-018. The flow rate being estimated every 4 hours is incorrect. Plausible if the student uses the wrong portion of the table from Table 2.6-1. This part is if the flow meter is not in service.

Question: 92
Tier/Group: 2/2
K/A Importance Rating: SRO 3.4
K/A: 068 Liquid Radwaste System (LRS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.04: Effluent release
Reference(s): SIM/Plant Design, ODCM Table 2.6-1
Proposed References to be provided to applicants during examination: ODCM Table 2.6-1
Learning Objective: Objective 9 from RMS lesson plan
Question Source: RNP Bank
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 41.5 / 43.5 / 45.3 / 45.13
Comments:

This question meets the K/A because the candidate must predict what should have happened to the release based off of R-18 being above the alarm setpoint, and from that, use the ODCM to control the release.

TABLE 2.6-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway / Instrumentation	MCO*	Compensatory Measures
1. Liquid Radwaste Effluent Discharge Line a. Monitor (R-18)	1	With the number of channels operable less than the MCO requirements: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that prior to initiating a release: 1. Two independent samples are analyzed in accordance with the Surveillance Requirements of ODCM Specification 2.2.1 and; 2. Two members of the facility staff independently verify the release rate calculations and the discharge line valving.
b. Flow rate measurement device	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may be continued, provided that the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated "in situ" and tank volumes may be used to estimate flow.

correct

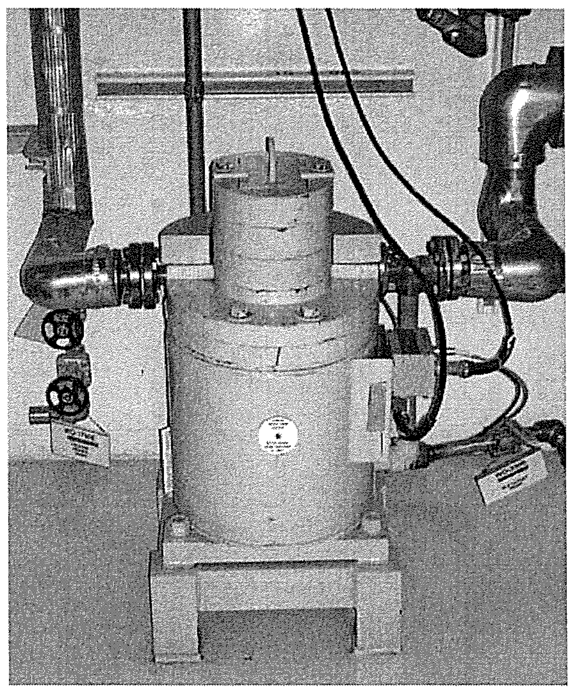
incorrect

*MCO - Minimum Channels Operable

R-18, Liquid Waste Disposal Effluent

1. Uses a MD-51 Gamma Scintillation Detector.
2. This channel continuously monitors all Waste Disposal System liquid releases from the plant.
3. The alarm setpoint is determined for each liquid waste release. Upon alarm, RCV-018 will automatically close.
4. The detector is mounted in a shielded assembly.

Figure 17 R-18; Liquid Waste Disposal Effluent Monitor



Provided Reference

TABLE 2.6-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway / Instrumentation	MCO*	Compensatory Measures
1. Liquid Radwaste Effluent Discharge Line	1	With the number of channels operable less than the MCO requirements: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that prior to initiating a release: 1. Two independent samples are analyzed in accordance with the Surveillance Requirements of ODCM Specification 2.2.1 and; 2. Two members of the facility staff independently verify the release rate calculations and the discharge line valving.
b. Flow rate measurement device	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may be continued, provided that the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated "in situ" and tank volumes may be used to estimate flow.

*MCO - Minimum Channels Operable

Provided reference

TABLE 2.6-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway / Instrumentation	MCO*	Compensatory Measures
2. Steam Generator Blowdown Effluent Line a. Monitor (R-19A,B, and C)	1 per S/G	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that grab samples are analyzed for gross radioactivity (beta or gamma) with a lower limit of detection of at least 1.0E-07μCi/ml or are analyzed for principle gamma emitters consistent with Table 2.8-1; 1. Once per 24 hours when the specific activity of the secondary coolant is ≤0.01 μCi/ml Dose Equivalent I-131, or; 2. Once per 12 hours when the specific activity of the secondary coolant is >0.01 μCi/ml dose Equivalent I-131.
b. Flow rate measurement devices - each Steam Generator has its own blowdown flow rate measuring device. These devices only measure flow directed through the heat recovery system, and will not measure flow which bypasses the heat recovery system.	1 per S/G	With the number of channels operable less than the MCO requirement due to inoperable equipment: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 **AND**, With the number of channels operable less than the MCO requirement due to inoperable equipment, **OR** if the steam generator blowdown system is aligned such that any flow bypasses the flow measurement device(s) (i.e. heat recovery is not in service): b. Effluent releases via this pathway may continue provided that the flow rate for the affected blowdown line(s) is estimated at least once per 24 hours.

*MCO - Minimum Channels Operable

Provided reference

TABLE 2.6-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
2. Steam Generator Blowdown Effluent Line (continued) c. R-19A, B and C flow measurement device – each monitor has its own flow rate measurement device	1 per S/G	With the number of channels operable less than the MCO requirement due to inoperable equipment: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that the flow rate for the affected monitor line(s) is estimated at least once per 24 hours.
3. Discharge Canal Flow	Note 1	With the number of channels operable less than the MCO requirement suspend effluent release via this pathway.
4. Tank Level Indicating Devices		With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Liquid additions to the affected tank(s) may continue provided that the liquid level for the affected tanks is estimated during all liquid additions to the affected tank(s).
a. Refueling Water Storage Tank	1	
b. Monitor Tanks		
Tank A	1	
Tank B	1	
c. Waste Condensate Tanks		
Tank C	1	
Tank D	1	
Tank E	1	
d. Outside Temporary Tanks (Note 2)	1 per Tank	

*MCO - Minimum Channels Operable

Provided reference

TABLE 2.6-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
5. Containment Fan Cooling Water Monitor (Service Water Effluent Line) a. Monitor (R-16) does not provide automatic termination of release upon exceeding alarm setpoint.	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that, once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) with a lower limit of detection of at least 1.0E-07 µCi/ml or are analyzed for principal gamma emitters consistent with Table 2.8-1.
6. Composite Sampler for Settling Ponds	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that, grab samples are collected and composited three times per week and analyzed in accordance with Table 2.8-1.

02/09

*MCO - Minimum Channels Operable

Provided reference

TABLE 2.6-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Release Pathway/Instrumentation	MCO*	Compensatory Measures
7. Condensate Polisher Liquid Waste Monitor a. Monitor (R-37) provides automatic termination of release upon exceeding alarm/trip setpoint	1	With the number of channels operable less than the MCO requirement: a. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner in accordance with Technical Specification 5.6.3 and, b. Effluent releases via this pathway may continue provided that, once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) with a lower limit of detection of at least 1.0E-07 µCi/ml or are analyzed for principal gamma emitters consistent with Table 2.8-1.

*MCO - Minimum Channels Operable

93. 071 G2.4.46 SRO 001

Given the following plant conditions:

-The crew has started a gaseous waste release from WGDT C

-ATTACHMENT 10.3, GASEOUS WASTE RELEASE PERMIT-WASTE GAS DECAY TANK, is complete **(SEE REFERENCE ON NEXT PAGE)**

-During the release, R-14C, PLANT STACK, NOBLE GAS, read 12K cpm

Which ONE(1) of the following completes the statements below?

The ___(1)___ had to approve the release before it could begin. Based off of ATTACHMENT 10.3, R-14C ___(2)___ alarm during the release.

A. (1) E&C Supervisor
(2) did **NOT**

B. (1) E&C Supervisor
(2) did

C. (1) SM
(2) did **NOT**

D.✓ (1) SM
(2) did

GASEOUS WASTE RELEASE PERMIT - WASTE GAS DECAY TANK

RELEASE NUMBER: 13-013G SSN: 12345 DATE: Today

This revision is the latest revision available as verified by:

Mike Anderson	[signature]	[signature]	Today
Name (Print)	Initial	Signature	Date

PART I: PRE-RELEASE INFORMATION (E&C)			
A / B / C / D Waste Gas Decay Tank (Circle Appropriate Letter)		Estimated Release Start Today (Date) 1000 (Time) Estimated Release Stop Today (Date) 1037 (Time)	
Monitor	Setpoint	Basis (Circle One)	CV Purge (Circle One) **(NCR 410785)**
R-14C	1.00 E +6 CPM	EC Activity	In Service Not In Service

Maximum WGDT Flow Rate: 100 CFM

PART II: RADIATION MONITOR INFORMATION (OPS and E&C)	
READING	R-14C[2]
PRIOR[1] (Channel Check)	BB CPM
SOURCE CHECK[2]	E&C INI. AJ
SETPOINT VERF. AT[3]	1.01 E+4 CPM
UPDATE STATUS BOARD[4]	OPS INI. BB
DURING RELEASE	12 K CPM
AFTER RELEASE	20.9 CPM
SETPOINT RETURNED TO[3,5]	N/A CPM
STATUS BOARD UPDATED[4]	OPS INI. BB

The correct answer is D

A) Incorrect. E&C Supervisor is incorrect. Plausible since they can approve certain releases, however, this is not one they can approve. R-14C did not alarm is incorrect. Plausible if the student misuses attachment 10.3 and thinks that the setpoint in PART 1 for R-14C gets reset to 1.00E+6 cpm. Since R-14C is already set at a more conservative value, 1.01E+4 cpm, R-14C's setpoint would not have changed for the release.

B) Incorrect. E&C Supervisor is incorrect. Plausible since they can approve certain releases, however, this is not one they can approve. R-14C did alarm during the release. R-14C's setpoint for the release was 1.01E+4. The highest value it reached was 12K cpm, which exceeds it's setpoint.

C) Incorrect. SM is the correct person who will approve this release. SM is the correct person who will approve this release. R-14C did not alarm is incorrect. Plausible if the student misuses attachment 10.3 and thinks that the setpoint in PART 1 for R-14C gets reset to 1.00E+6 cpm. Since R-14C is already set at a more conservative value, 1.01E+4 cpm, R-14C's setpoint would not have changed for the release.

D) Correct. SM is the correct person who will approve this release. R-14C did alarm during the release. R-14C's setpoint for the release was 1.01E+4. The highest value it reached was 12K cpm, which exceeds it's setpoint.
Print out Attachment 10.3 from Z:\ILC-13 NRC Written Exam\SRO References\071 G2.4.46 SRO, titled "release". They will need this sheet to answer the question.

Question: 93
Tier/Group: 2/2
K/A Importance Rating: SRO 4.2

K/A: 071 Waste Gas Disposal System (WGDS)
2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Reference(s): EMP-022
Proposed References to be provided to applicants during examination: Attachment 10.3 of EMP-022
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 41.10 / 43.5 / 45.3 / 45.12
Comments:

This question meets the K/A because the student must determine if R-14C alarmed during a release based off of setpoints that could have changed on Attachment 10.3. Asking who approves the release is an SRO level task and that is what makes this an SRO level question.

Correct

ATTACHMENT 10.3
Page 1 of 2
GASEOUS WASTE RELEASE PERMIT - WASTE GAS DECAY TANK

RELEASE NUMBER: ______________________ SSN: ___________________ DATE: ___________________

This revision is the latest revision available as verified by:

Name (Print)	Initial	Signature	Date

PART I: PRE-RELEASE INFORMATION (E&C)			
A / B / C / D Waste Gas Decay Tank (Circle Appropriate Letter)	Estimated Release Start ________ Date ________ Time Estimated Release Stop ________ Date ________ Time		
Monitor	Setpoint	Basis (Circle One)	CV Purge (Circle One) **(NCR 410785)**
R-14C	CPM	EC Activity	In Service Not In Service

Maximum WGDT Flow Rate:_____________CFM

This release can be made within the limits of 10CFR20 and 10CFR50 using the setpoints and restrictions stated herein.

Prepared By: ______________________________ Peer Review: ______________________________

* Verified By ____________________________
* Only required if performing a WGDT gaseous waste release with R-14C out of service.

E&C Supervisor: ____________________________

PART II: RADIATION MONITOR INFORMATION (OPS and E&C)	
READING	R-14C[2]
PRIOR[1] (Channel Check)	CPM
SOURCE CHECK[2]	E&C INI.
SETPOINT VERF. AT[3]	CPM
UPDATE STATUS BOARD[4]	OPS INI.
DURING RELEASE	CPM
AFTER RELEASE	CPM
SETPOINT RETURNED TO[3,5]	CPM
STATUS BOARD UPDATED[4]	OPS INI.

Approved for Release: ______________________________________ (CR 97-00059)
Shift Manager

94. G2.1.38 SRO 001

Which ONE (1) of the following completes the statement below?

The CRS is responsible for conducting a(n) _______ every 45 – 60 minutes during long lasting events.

A. Crew Update

B. Alignment Brief

C. Crew Shift Brief

D✓ Plant Status Brief

The correct answer is D.

A. Incorrect. Plausible because it is one of the listed types of briefs in OPS-NGGC-1314 and the title of the brief does not suggest when it is performed.

B. Incorrect. Per OPS-NGGC-1314, Alignment briefs should be performed when transitioning between event procedures.

C. Incorrect. Plausible because it is one of the listed types of briefs in OPS-NGGC-1314 and the title of the brief does not suggest when it is performed.

D. Correct. A Plant status brief is conducted IAW OPS-NGGC-1314 ever 45-60 minutes during long lasting events.

Question: 94
Tier/Group: 3 (SRO)
K/A Importance Rating: RO 3.7 SRO 3.8

K/A: 2.1 Conduct of Operations
2.1.38 Knowledge of the stations requirements for verbal communication when implementing procedures.

Reference(s): OPS-NGGC-1314
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: 43.5
Comments:

This question meets the K/A because the candidate needs to know the station expectations for communicating information during implementation of AOP and EOPs.

Incorrect

9.4.2 **Middle-of-Shift Brief**

1. A middle-of-shift brief of the CRS, BOP and AOs shall be conducted. The purpose of this meeting is to discuss remaining / ongoing activities for the remainder of the shift, identify roadblocks, and assure a successful error-free outcome. The middle-of-shift brief may be waived at the discretion of the CRS/SM.

9.4.3 **Crew Update**

1. A Crew Update is to be used when a time-critical, status-critical or a key piece of information would be beneficial for the crew to know.
2. When an update is used, it should contain the following elements:
   a. Announcement of "Crew Update"
   b. Ensure all affected personnel are attentive signified by each raising a hand.
   c. Concise statement of the critical information.
   d. Announcement "End of Update".
3. Any member of the control room staff may use a “Crew Update” to communicate significant changes or discoveries to the entire control room staff. For example: “Crew update. Maintenance has completed setting the head on the vessel. End of update.”
4. The following are examples of when a Crew Update should be considered:
   a. Announcement of Emergency Action Level classification
   b. Significant changes in plant status (adverse containment values in effect, off site power restored, etc.).
   c. Trip of a critical component.
   d. Significant milestones in an evolution

INCORRECT

9.4.5 Alignment Briefs

a. Alignment Briefs are shorter in duration and less formal than Plant Status Briefs. Alignment Briefs should last for less than 30 seconds when possible. The purpose of an Alignment Brief is to ensure that crew members are aligned when time does not permit the performance of a formal Plant Status Brief, or when a formal Plant Status Brief is not warranted.

b. Alignment Briefs are held at the discretion of the CRS/SM. Alignment Briefs can be requested by any crew member.

c. Alignment Briefs should be used at the following times:

1) When entering or transitioning between event procedures.
2) When any crew member has a question related to mitigation strategy.
3) After an event in which plant response is not understood.
4) Prior to taking actions that will have a significant impact on plant operations.

d. When an Alignment Brief is used, it should contain the following elements:

1) CRS states "Attention in the Control Room for an Alignment Brief"
2) Ensure all affected personnel are attentive, signified by each raising a hand.
3) Discuss information as required for alignment. The discussion may include input/questions from crew members, provided that the discussion remains brief. Crew member input is not required.
4) The CRS announces "End of brief".

EXAMPLE

Alignment Briefs

Example 1: "Attention in the Control Room for an Alignment Brief. We are entering AP-510, Rapid Power Reduction, to lower power to 70% due to elevated vibrations on Reactor Coolant Pump A. Ted- you are going to manually maintain level in the "A" Steam Generator between 40% and 60% due to the feed reg valve failure. Any questions or comments? End of brief."

Example 2: "Attention in the Control Room for an Alignment Brief. We will be tripping the Reactor due to a malfunctioning Main Feedwater Pump. OAC, you will trip the Reactor and perform immediate actions. BOP, you will initiate Emergency Feedwater and trip Main Feedwater Pumps. Any questions or comments? End of brief."

Example 3: "Attention in the Control Room for an Alignment Brief. Do we understand why Reactor Coolant pressure is lowering?" … *[Brief discussion with crew members]* … "End of brief."

Incorrect

9.4 Briefs

9.4.1 Crew Shift Brief

1. Standards

   a. A Shift Brief is conducted at the beginning of each operating shift following the watch station turnovers. This briefing is expected to be free of distractions in order to facilitate the effective communication of plant conditions.

   b. Personnel of the on-coming shift attend and participate in this briefing. It is the Shift Manager's responsibility to make sure that the meeting has a clearly defined beginning and end, and that all participate in the information exchange.

   c. Communication during the briefing focuses on recent or pending status changes.

   d. The shift briefing is led by the duty Shift Manager, who establishes an atmosphere of open communication that is free from unnecessary distractions (side conversations, phone calls, etc.); during the brief, Operations standards for formality and professionalism are maintained.

2. Expectations

   a. The Shift Brief is lead by the Shift Manager or designee and shall be normally conducted outside the Control Room.

   b. The CRS will ensure the telephone access is restricted in the Control Room prior to the brief and will be restored at the conclusion as required.

   c. The OAC will participate in the brief to the extent that it does not distract from the primary responsibility to monitor the reactor.

   d. All participants must speak loudly enough to be heard over any background noise.

   e. At the discretion of the SM/CRS, the crew briefing may be stopped if unusual conditions exist which demand the prompt attention of the operating crew. The SM/CRS resumes the brief when the situation is again conducive to an effective exchange of information.

   f. Attachment 1, Shift Brief Checklist should be used as a guide for the shift brief to ensure all pertinent information is discussed.

   g. SM should review the crew composition for the shift. This review considers the crew makeup for individuals who do not routinely work or train together as a crew. Based on the results of this assessment compensatory measures are taken as appropriate.

Correct

9.4.4 **Plant Status Briefs**

1. Plant Status Briefs are conducted for the purpose of bringing Control Room personnel to the same level of understanding of the present plant status, overall mitigation plan, and emergency plan status.
2. Plant Status Briefs are held at the discretion of the CRS and are normally held following plant stability during emergency procedures. Additionally, Plant Status Briefs should be held:
   a. After mitigation of a major symptom.
   b. At the request of Control Room personnel.
   c. When exiting event procedures.
   d. Prior to performing complex/significant evolutions. **[R4]**
   e. Every 45 – 60 minutes during long lasting events.
3. Plant Status Briefs shall not interfere with the performance of time critical actions or immediate actions of event procedures.
4. ROs shall continue to maintain awareness of plant parameters during the brief.
5. The CRS should notify crew members of the intent to conduct a Plant Status Brief 1 – 2 minutes before the brief occurs.
6. A Plant Status Brief is performed by the CRS utilizing the following format:
   a. **B** *BEGIN*:
      - Announce "Attention in the Control Room for a Plant Status Brief"
      - Ensure all affected personnel are attentive, signified by each raising a hand
   b. **R** *RECAP:*
      - Provide overview of sequence of events and current plant status
      - Identify major equipment failures
      - Identify procedures that are in effect
      - Discuss status of actions performed outside the MCR

95. G2.2.13 SRO 001

Given the following plant conditions:

- The letdown line has been removed from service and cleared for maintenance to install several new vent valves.
- An approved test procedure has been provided with the work order package to perform the line hydrostatic test following installation.
- A clearance boundary change must be implemented to initiate the system hydrostatic test.

Which ONE (1) of the following completes the statements below?

IAW OPS-NGGC-1301, Equipment Clearance, each boundary change shall be authorized by __(1)__. The __(2)__ will provide concurrence for Maintenance personnel to introduce fluids into the clearance boundary for the system hydrostatic test.

A. (1) an SRO
   (2) WCC SRO

B. (1) the CRS only
   (2) CRS

C.✓ (1) an SRO
   (2) CRS

D. (1) the CRS only
   (2) WCC SRO

The correct answer is C.

A. Incorrect. SRO is correct for boundary change but WCC SRO is incorrect for the concurrence.

B. Incorrect. Maintenance Supervisor is incorrect for the boundary change.

C. Correct. Reference - OPS-NGGC-1301 Section 9.1.21 and 9.2.4.1.5. SRO is required for the boundary change and CRSS is required for the concurrence to introduce the fluids into the system.

D. Incorrect. Maintenance Supervisor is incorrect for the boundary change and WCC SRO is incorrect for the concurrence.

Question: 95
Tier/Group: 3 (SRO)
K/A Importance Rating: RO 4.1 SRO 4.3

K/A: 2.2 Equipment Control
2.2.13 Knowledge of tagging and clearance procedures.

Reference(s): OPS-NGGC-1301
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: RNP Bank
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 43.3
Comments:

This question meets the K/A because it tests the candidate on SRO responsibilities contained in the Equipment Clearance procedure.

41. If an Emergent Equipment Clearance is needed and the clearance computer program is unavailable, the appropriate attachments in the back of this procedure may be used to manually prepare the clearance.

    A Clearance Log Sheet (Attachment 1) will be filled out for tracking purposes starting with clearance number YY-XXXX. (YY = current year and XXXX will be sequential numbered starting at 0001). Once the computer system has been restored, all information should be entered into the computer.

    Any tags that are still hanging should be replaced with the computer generated tags.

    The manually generated clearance forms should be saved until the computer generated clearance has been closed and all pertinent information captured.

42. Specific Plant Addendums should be referenced for all employees working under this procedure.

43. Attachment 10, Clearance Process Checklist, is required to be used for all clearances generated in conjunction with this procedure.

44. The introduction of fluids, either gas or liquid, into a system within a clearance boundary may be performed by work groups other than Operations provided it is according to an approved procedure and with the concurrence of the CRS or SM. The location at which the fluid is to be introduced into the system shall be specified in the procedure.

45. Any approved clearance older than 6 months, must be evaluated to ensure that it meets the intent of the current revision to this procedure. If not, it must be re-written using the current revision.

46. When equipment that operates at temperatures above ambient is removed from service, it should be allowed to cool prior to starting work or precautionary measures, such as wearing the appropriate PPE, shall be taken to eliminate the potential burn hazard.

9.5 Clearance Development: Lift Checklist

1. During Lift Checklist development, a Clearance Preparer shall determine:
   - Restoration positions. The restoration position should be based on current plant status / conditions. This may include reviewing applicable procedure steps and/or valve lineups to determine plant status.
   - Specify the proper restoration sequence, paying particular attention to restoring in an order to prevent personnel injury or equipment damage.
   - Independent Verification requirements using OPS-NGGC-1303
2. If relying on a system alignment for restoration, the procedure and section or valve lineup being relied on should be specified in the Special Instructions. **[R18]**

> **NOTES:**
> 1. Boundary Change Checklists may be prepared prior to receiving Attachment 6, Boundary Change Form.
> 2. Steps 3.0 and 4.0 of Attachment 6 may be performed concurrently.
> 3. Hang and lifts shall be on separate checklists unless there are no holders.

9.6 Clearance Development: Boundary Change

1. Use Attachment 6, Boundary Change Form, for all boundary changes.
2. A Clearance Preparer shall develop a boundary change in accordance with Attachment 10 and identifies all affected clearance orders.
3. A second Clearance Preparer who is a Licensed Operator shall verify a boundary change is adequate.
4. Each boundary change shall be authorized by an SRO. This authorization is indicated by the Checklist Status being "Distributed" and indicates the SRO has verified the following:
   - Plant conditions are correct for the boundary change
   - The boundary change will not adversely impact plant operation
   - Applicable compensatory actions have been initiated
   - The Control Room has been notified, as necessary
5. The boundary change is then assigned to a Tag Hanger for implementation. The Tag Hanger shall initial each step and print their Passport short name at least once with their initials for each Checklist.
6. Workers and holders shall perform Zero Energy Checks after any change to the clearance.

96. G2.2.18 SRO 001

Given the following plant conditions:

- Two trains of RHR are **OPERABLE**

- RCS is at 135°F and has been drained to -30 inches standpipe level for RCP seal replacement

- Maintenance has requested that both CV Personnel Hatch doors be opened to allow cables and hoses to be routed through the doors to support maintenance activities

Which ONE (1) of the following completes the statements below?

IAW OMM-033, IMPLEMENTATION OF CV CLOSURE, CV Personnel Hatch doors can be opened provided CV closure can be implemented within ___(1)___hours. The hoses and cables can be routed through the hatch provided ___(2)___ for removal.

A.✓ (1) 0.5
(2) the hoses and cables have quick disconnects

B. (1) 4
(2) the hoses and cables have quick disconnects

C. (1) 0.5
(2) dedicated personnel are stationed inside the CV with tools needed

D. (1) 4
(2) dedicated personnel are stationed inside the CV with tools needed

The correct answer is A.

A. Correct. Per OMM-033, CV closure time for all penetrations is 30 minutes with the conditions provided in the stem. Hoses and cables blocking the door must be capable of immediate removal using quick disconnects, clamps, etc..

B. Incorrect. 4 hours is the maximum time of isolation for the Equipment Hatch. Plausible because the Personnel Hatch is a smaller penetration and it could be assumed that because it has a smaller release area that it could be assigned the longer duration for isolation.

C. Incorrect. Per OMM-033, CV closure time for all penetrations is 30 minutes with the conditions provided in the stem. Penetrations must be capable of being isolated from outside containment so personnel if required to be stationed would not be stationed inside containment. Plausible because there are requirements in some cases to have personnel stationed at the penetration and a lot penetrations can be isolated from either side.

D. Incorrect. 4 hours is the maximum time of isolation for the Equipment Hatch. Plausible because the Personnel Hatch is a smaller penetration and it could be assumed that because it has a smaller release area that it could be assigned the longer duration for isolation. Penetrations must be capable of being isolated from outside containment so personnel if required to be stationed would not be stationed inside containment. Plausible because there are requirements in some cases to have personnel stationed at the penetration and a lot penetrations can be isolated from either side.

Question: 96
Tier/Group: 3 (SRO)
K/A Importance Rating: RO 2.6 SRO 3.8

K/A: Equipment Control
G2.2.18: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc..

Reference(s): OMM-033
Proposed References to be provided to applicants during examination: None
Learning Objective: RCS012
Question Source: RNP Bank
Question History: 09 NRC EXAM
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 43.5
Comments: This question meets the K/A because the candidate must know the requirements for allowing maintenance to perform work and the restrictions required when shutdown.

Previous Version of Question

QUESTIONS REPORT

for ILC-09 NRC Written Exam Questions

1. G2.2.18 002

Given the following:

- Unit in Mode 5 for RCP seal replacement.
- RCS is at 135°F and has been drained to -30 inches standpipe level.
- Reactor has been shutdown for 28 hours.
- Maintenance has requested that both CV Personnel Hatch doors be opened to allow cables and hoses to be routed through the doors to support maintenance activities.

Which ONE (1) of the following describes the requirements that must be met for the CV Personnel Hatch IAW OMM-033, Implementation of CV Closure?

CV Personnel Hatch doors......

A. cannot be opened until 67 hours 5 minutes have elapsed since reactor shutdown.

B.✓ can be opened provided CV closure can be implemented within 30 minutes and the hoses / cables are provided with quick disconnects for removal. Closure must be implemented from outside of the CV.

C. can be opened provided CV closure can be implemented within 4 hours and the hoses / cables are provided with quick disconnects for removal. Closure must be implemented from outside of the CV.

D. can be opened provided CV closure can be implemented within 30 minutes. However, the hoses / cables that impede the closing of either door cannot be allowed.

3.5 It is the responsibility of the Work Group Supervisor to [SOER 09-1, Recommendation 11]:

3.5.1 Prepare the CV Closure Exception Permit.

3.5.2 Provide knowledgeable personnel on a 24 hour basis, available on site **AND** properly briefed, to secure all penetrations held open under this procedure that are his direct responsibility.

NOTE:	Power or air operated tools are acceptable, however closure shall be capable of being performed entirely from outside the CV (except for the Fuel Transfer Tube IAW Section 8.2) with only manual hand tools in the event that air or electrical power is **NOT** available.

3.5.3 Ensure that the following are available at the site of the open penetration:

- A device providing a positive form of closure for the open penetration, capable of withstanding at least 19 psia CV pressure, and being installed in a timely manner.
- All tools required to install the closure device.

3.5.4 Ensure that the tools and materials required to isolate each open penetration are inspected each shift and discrepancies corrected.

3.5.5 Ensure that the personnel designated to secure penetrations, are familiar with the installation method for all types of closure devices that fall under their responsibility.

3.5.6 Ensure that personnel who are responsible for closure of penetrations are familiar with this procedure.

3.5.7 Ensure the lead person designated to secure the penetration on each shift signs the Attachment 10.3, Penetration Closure Responsibility Log, prior to **OR** during turnover before the off-going designated person leaves the site. These actions should be completed within 1 hour of watch relief.

3.6 The work leader for the particular activity is responsible for having the tools, materials, and manpower required to isolate the penetration at the work site and for notifying the WCC of any turnover of responsibility.

3.7 Individuals designated as responsible for verifying penetrations are closed are responsible for notifying the WCC SRO to identify another individual if job responsibilities or temporary absences prevent performing the function.

5.15 In the event that the Equipment Hatch has been removed **AND** the CV Purge is lost due to reasons **NOT** involving a loss of a safety function (See OMM-046) **OR** a fuel handling accident is **NOT** in progress, **THEN** deployment of the Equipment Hatch Membrane is **NOT** required.

5.16 An aggregate impact assessment shall be performed by the WCC SRO using Attachment 10.5 when a new CV Closure Exception Permit is initiated **OR** a change in status of an existing CV Closure Exception Permit occurs.

5.17 Pre-outage scheduling and risk reviews should have considered aspects of open penetrations aggregate assessment. However, emergent issues or delayed activities may impact the aggregate assessment, and the number of penetrations open at any one time may be limited by the resources or methods available to effect closure.

5.18 A CV purge will remain in operation during any core alteration or movement of irradiated fuel assemblies while the equipment hatch is removed, even if time since shutdown is greater than 4 days.

5.19 Plant Aligned for Natural Circulation Table

Plant Aligned for Natural Circulation	Time After Shutdown	Time to Boil	Equipment Hatch
Yes	≥ 16 hours	N/A	May be removed and the runway installed. 4 hour Permit. 16 bolts must be installed and torque to ≥397 ft-lbs in ≤ 4 hours. (CV Closure Exception Permit for CV Equipment Hatch with Outside Runway)
No	<4 days	N/A	Must be installed.
No	≥ 4 days	< 41 minutes	Must be installed.
No	≥ 4 days	≥41 minutes AND < 4 hours	May be removed but must be installed by Time to Boil. 4 bolts must be installed and torque to ≥ 397 ft-lbs in ≤ Time to Boil. (CV Closure Exception Permit for CV Equipment Hatch)
No	≥ 4 days	≥ 4 hours	May be removed and the runway installed. 4 hour permit. 4 bolts must be installed and torque to ≥ 397 ft-lbs in ≤ 4 hours. (CV Closure Exception Permit for CV Equipment Hatch with Outside Runway)

REFERENCE USE

8.0 INSTRUCTIONS

8.1 Determining Penetration Closure Times

NOTE:	CV Closure time is **NOT** applicable with the core fully off loaded to the Spent Fuel Building.

8.1.1 **IF** the following conditions are satisfied, **THEN** allowed CV Closure Time is 30 minutes for all penetrations except the CV Equipment Hatch:

1. Two Trains of RHR are OPERABLE **AND**
   - Reactor Coolant System average temperature is less than or equal to 200°F
   - Reactor Coolant System level is above -36 inches

   OR

2. One Train of RHR OPERABLE with refueling cavity level between 16 and 29 inches as indicated on the Refueling Cavity Level Indicator **AND** Reactor Coolant System average temperature is less than or equal to 200°F.

3. **IF** NEITHER condition above is met, **THEN** CV Closure Time shall be determined from Plant Curve 3.5, Time to CV Closure, in the Station Curve Book.

8.1.2 **IF** the estimated time to close the open penetration exceeds 30 minutes, **THEN** an evaluation should be performed for the open penetration **AND** Operations Manager approval on Attachment 10.1 will be required.

8.1.3 CV Personnel Hatch

1. **WHEN** opening the CV Personnel Hatch, **THEN** at least one of the doors will be capable of being closed.
2. Any equipment impeding the closing of one of the doors shall be located in such a way that it can be immediately removed from the opening through the use of quick disconnects, clamps, etc.

8.1.4 **IF** the equipment hatch will be removed, **THEN** refer to Precaution and Limitation 5.18 for determination of required closure time

97. G2.3.13 SRO 001

Given the following plant conditions:

- A Site Area Emergency has been declared due to a LOCA outside containment
- All Emergency Response Organization facilities are staffed and functional
- The TSC has determined that the leak can be isolated
- Expected exposure to isolate the leak is > 5 Rem TEDE
- An operator has been briefed and is awaiting approval for entry into the Auxiliary Building to isolate the leak

Which ONE (1) of the following completes the statement below?

Authorization of the exposure ___(1)___ be delegated to the Radiological Control Director. If authorized, the operator is allowed to receive up to ___(2)___ REM for this entry.

A. (1) can
(2) 10

B. (1) can
(2) 25

C✓ (1) can NOT
(2) 10

D. (1) can NOT
(2) 25

The correct answer is C

A) Incorrect. This is the wrong authorizer. Plausible since the person can perform many of the same functions with concern to radiological controls as the Site Emergency Coordinator IAW EPOSC-04. However, this person cannot authorize individuals to exceed 5 REM TEDE in a year. 5 REM is incorrect. Plausible since this is the maximum limit workers can receive for normal activities.

B) Incorrect. This is the wrong authorizer. Plausible since the person can perform many of the same functions with concern to radiological controls as the Site Emergency Coordinator IAW EPOSC-04. However, this person cannot authorize individuals to exceed 5 REM TEDE in a year. 10 REM is correct for repair efforts during a casualty.

C) Incorrect. This is the correct authorizer. 5 REM is incorrect. Plausible since this is the maximum limit workers can receive for normal activities.

D) Correct. The SEC can authorize this entry and approve the individual exceeding the 5 EM TEDE in a year.

NRC Exam

Question: 97
Tier/Group: 3
K/A Importance Rating: RO 3.2 SRO 3.7
K/A: G2.3.4: Knowledge of radiation exposure limits under normal or emergency conditions.
References: EPOSC-04, EPCLA-1
Proposed references to be provided to applicants during the Exam: None
Learning Obj:
Question Source: RNP bank (Modified G2.3.4 002)
Question History: No NRC Exams
Question Cognitive Level: Low
10 CFR part 55: 43.4

Meets the K/A because the student must know radiation exposure limits under emergency conditions and requirements for approval.

2.21 RNP-RA/05-0082, Response to NRC Bulletin 2005-02, Emergency Preparedness and Response Actions for Security-Based Events

2.22 NUREG-0654, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

2.23 EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, U.S. Environmental Protection Agency, Washington, D.C., May 1992

2.24 NRC Regulatory Issue Summary 2005-08, Range of Protective Actions For Nuclear Power Plant Incidents

2.25 NRC Interim Compensatory Measures Order, Section B.5.b

2.26 NRC Regulatory Issue Summary 2007-02, Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events

2.27 NEI 99-01 Revision 4, Methodology for Development of Emergency Action Levels

2.28 CAPR 258209, GE Declaration during Graded Exercise

2.29 EMG-NGGC-0005, Activation Of The Emergency Response Organization Notification System

2.30 EPEOF-10, Recovery Manager And Recovery Operations

2.31 NSIR/DPR-ISG-01, Interim Staff Guidance

2.32 EPRAD-03, Dose Projections

2.33 NRC RIS Issue Summary 2008-26, Clarified Requirements of Title 10CFR50.54(y) When Implementing 10CFR50.54(x) to Depart From a License Condition or Technical Specification

3.0 RESPONSIBILITIES

3.1 The Site Emergency Coordinator (SEC) has immediate and unilateral authority to implement this procedure.

3.2 The SEC may not delegate:

3.2.1 The decision to notify offsite authorities;

3.2.2 Making offsite Protective Action Recommendations (PAR);

3.2.3 Classifying or terminating the emergency;

3.2.4 Authorizing exposures in excess of 10 CFR 20 limits during a declared emergency.

8.6.6 **IF** the monitored dose rates **OR** stay times encountered during the entry exceed the limits set forth for the mission, **THEN** the team will immediately communicate with OSC and seek further direction.

8.6.7 Once the mission has been completed, the team will follow established monitoring **AND** personnel decontamination procedures **OR** take other actions specified by the RCD.

8.7 Emergency Worker Dose Limits

8.7.1 Regulatory limits shall be observed for planned radiation exposures to emergency workers **UNLESS** the SEC **AUTHORIZES** the individual to exceed 5 Rem TEDE in a year.

8.7.2 The table shown below identifies the Emergency worker dose limits. In addition to the categories listed in the table doses should be limited as follows:

1. The lens of the eye should be limited to three (3) times the stated TEDE value.

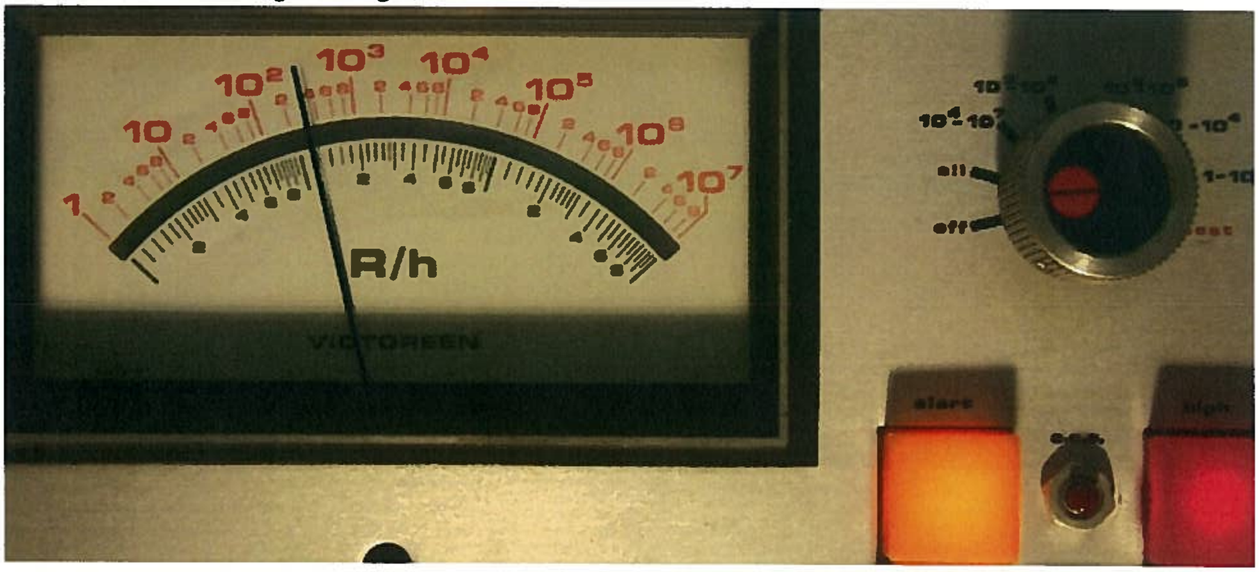
2. Any other organ (including skin and body extremities) should be limited to ten (10) times the stated TEDE value.

Dose Limit	Activity	Condition
5 REM	All	
10 REM	Repair and reentry efforts	Lower dose not practicable
25 REM	Lifesaving or protection of large populations	Lower dose not practicable
>25 REM	Lifesaving or protection of large populations	Only on a voluntary basis to persons fully aware of the risks involved

8.7.3 There may be situations where saving a life is not the issue, **AND** it is necessary to enter a hazardous area under repair/reentry efforts, to protect valuable installations, **OR** to make the facility more secure against events which could lead to radioactivity releases (e.g., assessment actions, entry of damage repair parties who are to repair valve leaks, **OR** add iodine-fixing chemicals to spilled liquids).

98. Given the following plant conditions:

-A Large Break LOCA has occurred
-Containment pressure is 32 psig and rising
-HVH-1 and HVH-2 have tripped
-Containment Sump level is rising
-R-32A, CV High Range, has the current indication:



Which ONE (1) of the following completes the statement below?

Based on the information provided a __(1)__ must be declared due to __(2)__.

(References Provided)

A. (1) Site Area Emergency
   (2) Loss of the Reactor Coolant AND Fuel Cladding Barriers

B. (1) General Emergency
   (2) Loss of the Reactor Coolant, Fuel Cladding AND Containment Barriers

C. (1) Site Area Emergency
   (2) Loss of the Reactor Coolant Barrier AND a Potential Loss of the Containment Barrier

D. (1) General Emergency
   (2) Loss of the Reactor Coolant AND Fuel Cladding Barrier AND Potential Loss of the Containment Barrier

The correct answer is A

A) Correct. A SAE will be declared. R-32A indicates approximately 300 R/hr. This correlates to a loss of Fuel Cladding Barrier due to item #3 and a loss of Reactor Coolant System Barrier due to item #1. This gives you a loss of any two barriers per FS1.1.

B) Incorrect. A GE is incorrect. Plausible because R-32A indicates approximately 300 R/hr. This correlates to a loss of Fuel Cladding Barrier due to item #3 and a loss of Reactor Coolant System Barrier due to item #1. If the student misinterprets sump level rising for response not consistent with LOCA conditions (#2 of Containment Barrier), this would get them to a GE.

C) Incorrect. A SAE will be declared. R-32A indicates approximately 300 R/hr. This correlates to a loss of Fuel Cladding Barrier due to item #3 and a loss of Reactor Coolant System Barrier due to item #1. The Potential loss of containment barrier is plausible if the student mistakes HVH-1and HVH-2 tripping for < one full train of depressurization equipment operating. CV spray is still operating and so is the "B" train of HVH.

D) Incorrect. A GE is incorrect. Plausible because R-32A indicates approximately 300 R/hr. This correlates to a loss of Fuel Cladding Barrier due to item #3 and a loss of Reactor Coolant System Barrier due to item #1. The Potential loss of containment barrier is plausible if the student mistakes HVH-1and HVH-2 tripping for < one full train of depressurization equipment operating. CV spray is still operating and so is the "B" train of HVH.

Question: 98
Tier/Group: 3
K/A Importance Rating: SRO 2.9
K/A: Radiation Control
G2.3.5: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
Reference(s): Sim/Plant design, Hot Conditions EAL Matrix
Proposed References to be provided to applicants during examination: Hot conditions EAL Matrix
Learning Objective: Objective EAL-004
Question Source: RNP Bank
Question History: ILC 11-1 NRC exam. This question was modified so a previously wrong answer is now correct. Removed the loss of off-site power from the stem. Removed the loss of B EDG. Added HVH-2 tripping as well as HVH-1. Also added containment sump level status to aid in the distractors.
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.11 / 41.12 / 43.4 / 45.9)
Comments:

This question meets the K/A because the student must use the rad monitoring system (R-32A) to aid in making an EAL Classification which is an SRO task. The EAL Classification is what makes this an SRO question.

99. G2.4.23 SRO 001

Given the following plant conditions:

-The crew is in EPP-28, LOSS OF ULTIMATE HEAT SINK, for a total loss of the intake structure

-Subsequently a loss of all AC power occurs

Which ONE (1) of the following completes the statement below?

The CRS ___(1)___ transition to EPP-1, LOSS OF ALL AC POWER, because ___(2)___ .

A.✓ (1) will **NOT**
(2) EPP-28 contains actions to deal with a loss of all AC power

B. (1) will
(2) EPP-1 contains actions to deal with a loss of all AC power

C. (1) will
(2) EPP-1 has priority over all other EPP procedures

D. (1) will **NOT**
(2) EPP-28 has priority over all other EPP procedures

The correct answer is A

A) Correct. The CRS will not transition to EPP-1. EPP-28 has a note in the beginning of it saying that it has higher priority over EPP-1. The basis for this is that EPP-28 has actions to take care of a loss of all ac.

B) Incorrect. The CRS will transition to EPP-1 is correct. Plausible because EPP-1 is a higher priority EPP than all other EPP's except EPP-28. EPP-1 does not have priority over all procedures. Plausible because EPP-1 does contain the actions to deal with a loss of all AC power.

C) Incorrect. The CRS will transition to EPP-1 is incorrect. Plausible because EPP-1 is a higher priority EPP than all other EPP's except EPP-28. EPP-1 does not have priority over only the EPP procedures. Plausible because in most procedures, you cannot follow their mitigation strategy without power to the components. EPP-28 is the one exception that takes into account a loss of all ac power.

D) Incorrect. The CRS will not transition to EPP-1 is correct. EPP-28 does not have priority over all the EPP procedures, plausible because there are two procedures that it doesn't have priority over, EPP-21/25. EPP-28 sends you to EPP-21 to energize your pressurizer heaters and EPP-25 to energize supplemental plant equipment.

Question: 99
Tier/Group: 3
K/A Importance Rating: SRO 4.4
K/A: Emergency Procedures / Plan
G2.4.23: Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.
Reference(s): Sim/Plant design,EPP-28 and its basis document
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 4 of EPP-28 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.10 / 43.5 / 45.13)
Comments:

This question meets the K/A because the student must prioritize which EPP to use and know the reason why(basis) to use that procedure. This question is SRO level because they must decide which procedure to use to mitigate this event.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

Subsequent steps may require access to Non-vital Areas. Access to these areas prior to nullification of the threat could cause loss of personnel.

****

NOTE

- FRPs, EPP Foldouts, EPP-1, AOP-014, AOP-020, AND AOP-022 are NOT applicable for this event. Transition OR use of these procedures should NOT be made unless otherwise directed in this procedure.
- Any time MCC-6 is deenergized, coordination of field activities will require the use of portable radios OR cell phones. It is recommended that the DS Radios OR Fire Protection radios be used for this communication.

1. **Check Reason For Entry - LOSS OF INTAKE STRUCTURE**

Go to the Section for the current Plant Mode:

MODE	SECTION
3	Section E
4	Section F
5	Section G
6	Section H
Defueled	Section I

RNP STEP **BASIS**

Main Body – LOIS MODES 1 THROUGH 3

C1 STEP BASIS

Subsequent steps throughout the procedure contain local manual operator actions outside the Control Room. Prior to performing these steps the threat should be nullified. Previous steps in AOP-034 have directed the Control Room to have personnel seek shelter. An exact time can not be determined for when the threat will be nullified. The intent of the caution is to remind the Operator that prior to sending an operator to perform an action; he should have a reasonable expectation that the Operator can reach the area and accomplish the task.

1N1 STEP BASIS

The note explains that FRBs, EPP Foldouts, EPP-1, AOP-014, AOP-020, and AOP-022 are not applicable for this event and that these procedures should not be implemented except as directed by this procedure. These procedures assume that the UHS is available and will not provide any meaningful strategies for this event. Where EPP-1, AOP-014, AOP-020, and AOP-022 have actions or individual attachments which will provide assistance, this procedure will direct their implementation. Any individual steps that need implementation have been included in this procedure. This note is applicable for the entire procedure; therefore it has been placed at the beginning of the procedure.

2N1 STEP BASIS

Most of the events that this procedure is intended to mitigate could involve a loss of 480V Bus E-2 and MCC-6. The Plant PA is powered from a bus fed by MCC-6, therefore under these conditions the plant PA will not be available. The note recommends that the Fire Protection or DS Radios be used for coordination of field activities. The plant cell phone repeaters are powered from non vital power and are not expected to last for longer than 1 hour following a loss of power. The DS/Fire Protection radios will provide a more reliable form of communications. The DS radios and fire protection radios both operate on the same frequency.

1-2 STEP BASIS

These are decision steps to control the applicable portion of the procedure to be performed. Since the time frames for the two components of interest are so divergent, the procedure has been divided into different sections (see general). If both the intake and the dam have been lost the step is arranged such that the intake section is performed since this is much more limiting. Step one checks for a loss of Intake or Dam. The RNO (LOTG) sends the operator to the appropriate Section for LOTG dependent on the Mode at the time of the event. Step 2 sends the operator to the appropriate section for a loss of the intake dependent on Mode.

NOTE: The remainder of the steps in the Main Body or for a LOIS event for Modes 1 through 3.

3 STEP BASIS

This step ensures the MSIV and Bypass valves are closed to limit the amount of cooldown. Based on simulator observations, when the turbine is tripped from greater than approximately 25% power, auxiliary feedwater is initiated from S/G low level resulting in additional cooling and a lower PZR level. In this circumstance when direction is given to stop RCPs later in the procedure, an SI is initiated due to steam line delta pressure. Isolating the steam lines early and establishing control of auxiliary feedwater flow prior to stopping RCPs helps to minimize the heat loss and reduce the likelihood for an SI initiation.

C4 STEP BASIS

Based on calculation RNP-M/MECH-1769, Operations needs to provide well water to the running EDG within 40 minutes after it starts to assure continued operation for the EDG or shut it down, and valve in well water to the CCW heat exchanger within one hour into the event to prevent CCW temperature from exceeding 125°F. The 125°F CCW temperature is based on ensuring the integrity of RCP seals. This caution provides the times available for establishing the alternate cooling from well water for operator awareness.

100. G2.4.30 SRO 001

Given the following plant conditions:

- Today at 0100 LCO 3.4.13, RCS OPERATIONAL LEAKAGE, was entered due to an 8 gpm leak from a crack on the spray line penetration weld at the PZR
- The crew commenced a shutdown at 0200
- The plant reached MODE 3 at 0800

Which ONE (1) of the following completes the statement below?

Notifications must be made to the NRC by ___(1)___ and ___(2)___.

(REFERENCES PROVIDED)

A. (1) 0300
   (2) 0800

B.✓ (1) 0600
   (2) 0800

C. (1) 0300
   (2) 1200

D. (1) 0600
   (2) 1200

The correct answer is B.

A. Incorrect. 0300 is incorrect because the notification requirement is 4 hours vice 1. Plausible because RCS leakage is also an E-Plan classification, however the value is 10 gpm for pressure boundary leakage.

B. Correct. Commencing a Technical Specification required shutdown requires a notification to the NRC within 4 hours IAW AP-030 based on 10CFR requirements. The shutdown commenced at 0200 therefore the 4 hour notification must be made by 0600. Another notification is required to be made to the NRC within one 1 hour of exceeding the LCO shutdown time requirements per the E-Plan under EAL Matrix SU3.1. The LCO was entered at 0100 with a 6 hr requirement to be in Mode 3 which expired at 0700. This requires an Unusual Event classification. The NRC is required to be notified as soon as possible not to exceed one hour which would be 0800.

C. Incorrect. 0300 is incorrect because the notification requirement is 4 hours vice 1. Plausible because RCS leakage is also an E-Plan classification, however the value is 10 gpm for pressure boundary leakage. 1200 is incorrect because the E-Plan classification requires a one hour notification. 1200 is plausible because a notification is required IAW AP-030 for completion of a TS required shutdown, however it is a 60 day LER.

D. Incorrect. 1200 is incorrect because the E-Plan classification requires a one hour notification. 1200 is plausible because a notification is required IAW AP-030 for completion of a TS required shutdown, however it is a 60 day LER.

Question: 100
Tier/Group: 3 (SRO)
K/A Importance Rating: RO 2.7 SRO 4.1

K/A: Emergency Procedures/Plan
G2.4.30: 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.

Reference(s): AP-030, EAL Matrix, TS 3.4.13
Proposed References to be provided to applicants during examination: EAL Matrix, AP-030 , Attachment 11.1-11.3, LCO 3.4.13
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 43.2, 5
Comments:

This question meets the K/A because the candidate must determine the required notifications to an external agency (NRC) based on conditions given in the stem.

You have completed the test!

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC			
10 CFR 50.72 states that immediate reports shall be made to the NRC Operations Center of these Emergency Events via the NRC Emergency Telecommunications System (ETS) as specified in the Emergency Plan. 10 CFR 50.72 additionally identifies Non-Emergency Events which are to be reported within one-hour, four-hours, or eight hours to the NRC. ETS Telephones, which are identified, are located in the Control Room, the TSC, and the EOF. In the event that the ETS is not available, 10 CFR 50.72(a)(2) permits the use of commercial telephone.			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
NOTE: 10 CFR 50.72 recognizes the Emergency Plan and its four Emergency Classes of Unusual Event, Alert, Site Area Emergency and General Emergency.			
EMERGENCIES 10 CFR 50.72(a)(i) 10 CFR 30.32(i)(3)(viii) 10 CFR 40.31(i)(3)(viii) 10 CFR 72.75(a)	Emergency Unusual Event Alert Site Area Emergency General Emergency ISFSI	HBRSEP shall notify the NRC of the declaration of any of the Emergency Classes specified in the Emergency Plan. (See EPNOT-01)	– Declaration of an Unusual Event, Alert, Site Area Emergency, or General Emergency. – Discovery of an event that should have resulted in an Emergency Classification, but no emergency was declared. – Discovery that a declared emergency exceeded the Emergency Action Levels for a higher emergency declaration, but the higher classification was not declared.
ERDS ACTIVATION 10 CFR 50.72(a)(4)	ERDS Emergency	HBRSEP shall activate the ERDS as soon as possible but not later than one hour after declaring an Alert, Site Area Emergency, or General Emergency.	– An Alert, Site Area Emergency, or General Emergency is declared.
DEVIATION FROM TS (10 CFR 50.54(X)) 10 CFR 50.72(b)(1)	Deviation Departure License Condition	Any deviation from the TS authorized pursuant to 10 CFR 50.54(x).	– Intentional deviation from an approved plant procedure in order to preserve plant safety 10 CFR 50.54(x).

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SHUTDOWN REQUIRED BY TS 10 CFR 50.72(b)(2)(i)	Shutdown TS Shutdown Power Reduction	The initiation of any shutdown required by the TS.	– Reactor is in MODEs 1 or 2 and the Control Room takes action to reduce power (i.e., negative reactivity insertion) in order to comply with a Required Action to be in MODE 3 within a Completion Time. Reduction in power for some other purpose than compliance with the shutdown requirement is not reportable. MODE changes required by TS when reactor is in MODEs 3, 4, or other non-power conditions, are not reportable. – If allowed outage time plus required shutdown time to MODE 3 is less than the expected restoration time of the LCO and power is reduced in anticipation of the required shutdown, the shutdown is reportable.
ECCS DISCHARGE INTO RCS 10 CFR 50.72(b)(2)(iv)(A)	ECCS Actuation Safety Injection	Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	– Manual or automatic Safety Injection System actuation in response to a valid signal that resulted in or should have resulted in discharge into the reactor coolant system.
RPS INITIATION (MANUAL/AUTOMATIC) DURING OPERATION 10 CFR 50.72(b)(2)(iv)(B)	RPS Actuation Reactor Protection System RPS Reactor Trip	Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	- Manual or automatic reactor trip from critical through RTP of 100%. Trips which occur as part of planned evolutions in accordance with procedures are not reportable.•

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 75 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

Provided Reference

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 75 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Reference Provided

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC			
10 CFR 50.72 states that immediate reports shall be made to the NRC Operations Center of these Emergency Events via the NRC Emergency Telecommunications System (ETS) as specified in the Emergency Plan. 10 CFR 50.72 additionally identifies Non-Emergency Events which are to be reported within one-hour, four-hours, or eight hours to the NRC. ETS Telephones, which are identified, are located in the Control Room, the TSC, and the EOF. In the event that the ETS is not available, 10 CFR 50.72(a)(2) permits the use of commercial telephone.			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
NOTE: 10 CFR 50.72 recognizes the Emergency Plan and its four Emergency Classes of Unusual Event, Alert, Site Area Emergency and General Emergency.			
EMERGENCIES 10 CFR 50.72(a)(i) 10 CFR 30.32(i)(3)(viii) 10 CFR 40.31(i)(3)(viii) 10 CFR 72.75(a)	Emergency Unusual Event Alert Site Area Emergency General Emergency ISFSI	HBRSEP shall notify the NRC of the declaration of any of the Emergency Classes specified in the Emergency Plan. (See EPNOT-01)	– Declaration of an Unusual Event, Alert, Site Area Emergency, or General Emergency. – Discovery of an event that should have resulted in an Emergency Classification, but no emergency was declared. – Discovery that a declared emergency exceeded the Emergency Action Levels for a higher emergency declaration, but the higher classification was not declared.
ERDS ACTIVATION 10 CFR 50.72(a)(4)	ERDS Emergency	HBRSEP shall activate the ERDS as soon as possible but not later than one hour after declaring an Alert, Site Area Emergency, or General Emergency.	– An Alert, Site Area Emergency, or General Emergency is declared.
DEVIATION FROM TS (10 CFR 50.54(X)) 10 CFR 50.72(b)(1)	Deviation Departure License Condition	Any deviation from the TS authorized pursuant to 10 CFR 50.54(x).	– Intentional deviation from an approved plant procedure in order to preserve plant safety 10 CFR 50.54(x).

Reference Provided

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

HBRSEP shall immediately notify the NRC Operations Center via ETS as soon as practical and in all cases within one hour of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SAFETY LIMIT, LIMITING SAFETY SYSTEM SETTING EXCEEDED 10 CFR 50.36(c)(1)(i)(A) UFSAR Section 17.3A, Paragraph 3.1.a	Safety Limit Limiting Safety System Setting	If any safety limit is exceeded, shut down the reactor. HBRSEP shall notify the NRC [within 1 hour via ETS per 10 CFR 50.72(a)(1), See Emergency Plan Procedures]. Operation must not be resumed until authorized by the NRC. NRC Region II must also be notified within 1 hour and the Vice President - Robinson Nuclear Plant within 24 hours	– The limits of TS Figure 2.1.1-1 are exceeded.
SAFETY SYSTEM DOES NOT FUNCTION AS REQUIRED 10 CFR 50.36(c)(1)(ii)(A)	ESF RPS Limiting Safety System Setting	HBRSEP shall notify the NRC if the automatic safety system [to correct an abnormal situation before a safety limit is exceeded] has been determined not to function as required.	– A failure mechanism is discovered that indicates that the RPS will not function to trip the reactor under certain required conditions.

Reference Provided

ATTACHMENT 11.1
Page 3 of 9
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the ETS within one hour* after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
THEFT/UNLAWFUL DIVERSION OF SNM OR SPENT FUEL SHIPMENT 10 CFR 73.71(a)(1)	SNM Spent Fuel Security Safeguards	Any discovery of the loss of any shipment of SNM or spent fuel, and within one hour after recovery of or accounting for such lost shipment	– Shipment Emergency Event
THEFT/UNLAWFUL DIVERSION OF SNM 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(1)	Theft of SNM Diversion Security Safeguards	Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause: (1) A theft or unlawful diversion of SNM	– Shipment Emergency Event
SABOTAGE OF PLANT EQUIPMENT 10 CFR 73.71(b)(1) 10 CFR 73, Appendix G, I(a)(2)	Sabotage Damage to Plant SNM Spent Fuel Security Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (2) Significant physical damage to a power reactor...or its equipment or carrier equipment transporting nuclear fuel or spent nuclear fuel, or to the nuclear fuel or spent fuel a facility or carrier possesses.	– Shipment Emergency Event – Security Event (SEC-NGGC-2147)

* In response to NRC Bulletin 2005-02, RNP committed to make an accelerated call to the NRC within approximately 15 minutes following discovery of an imminent threat or attack against the station. The primary purpose is to allow for the NRC to timely notify other licensees of a potential common threat. The accelerated call should not be allowed to interfere with plant or personnel safety, physical security response, or notification of local law enforcement agencies. The information provided in the accelerated call can be limited to:

- Site name
- Emergency Classification – if already determined – do not delay call for the purpose of classifying
- Nature of the threat – briefly described, if known, including the type of attack (e.g., armed assault by land, water or aircraft) and the attack status (e.g., imminent, in progress, or repelled)

Reference Provided

ATTACHMENT 11.1
Page 4 of 9
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SECURITY SAFEGUARDS EVENTS

HBRSEP shall notify the NRC Operations Center via the ETS within one hour* after discovery of the safeguards events described as follows (10 CFR 73.71(b)(1)):

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
UNAUTHORIZED TAMPERING WITH PLANT EQUIPMENT 10 CFR 73, Appendix G, I(a)(3)	Unauthorized Use Tampering Security System Safeguards	[Any event in which there is reason to believe that a person has committed or caused, or attempted to commit or cause, or has made a credible threat to commit or cause:] (3) Interruption of normal operation of HBRSEP through the unauthorized use of or tampering with its machinery, components, or controls including the security system.	– Security Event (SEC-NGGC-2147)
ENTRY OF UNAUTHORIZED PERSON INTO PROTECTED OR VITAL AREA 10 CFR 73, Appendix G, I(b)	Unauthorized Entry Security Safeguards	An actual entry of an unauthorized person into a protected area, material access area, controlled access area, vital area, or transport.	– Security Event (SEC-NGGC-2147)
FAILURE, DEGRADATION, OR DISCOVERED VULNERABILITY OF SAFEGUARD SYSTEM 10 CFR 73, Appendix G, I(c)	Degradation Vulnerability Safeguards Unauthorized Undetected Access Security	Any failure, degradation, or the discovered vulnerability in a safeguard system that could allow unauthorized or undetected access to a protected area, material access area, controlled access area, vital area or transport for which compensatory measures have not been employed.	- Procedure SEC-NGGC-2147
INTRODUCTION OF CONTRABAND INTO VITAL OR PROTECTED AREA 10 CFR 73, Appendix G, I(d)	Contraband Unauthorized Security Safeguards	The actual or attempted introduction of contraband into a protected area, material process area, vital area, or transport.	- Contraband applies to items that could be used to commit radiological sabotage as defined in 10 CFR 73.2.

* See footnote on the previous page regarding a goal for a 15 minute call to the NRC in regard to an imminent security threat or attack.

Reference Provided

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM HBRSEP shall immediately notify the NRC Operations Center via ETS, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
EXTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, OR SNM (5X ANNUAL LIMIT) 10 CFR 20.2202(a)(1)	Byproduct Source SNM Exposure Dose Release Occupational	Notwithstanding any other requirements for notification, immediately notify the NRC of any event involving byproduct, source, or SNM possessed by HBRSEP that may have caused or threatens to cause any of the following conditions: 1. An individual to receive: (i) A total effective dose equivalent of 25 rems or more; or (ii) An eye dose equivalent of 75 rems or more; or (iii) A shallow dose equivalent to the skin or extremities of 250 rads or more; or 2. The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	

Reference Provided

ATTACHMENT 11.1
Page 6 of 9
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SOURCE, BYPRODUCT AND SNM HBRSEP shall immediately notify the NRC Operations Center via ETS, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
INTERNAL EXPOSURE FROM BYPRODUCT, SOURCE, SNM (>5X OCCUPATIONAL LIMIT) 10 CFR 20.2202(a)(2)	Intake Ingestion Release Source Byproduct SNM	The release of radioactive material, inside or outside the restricted area, so that, had an individual been present for 24 hours, the individual could have received an intake five times the occupational annual limit on intake.	
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - ISFSI HBRSEP shall immediately notify the NRC Operations Center via ETS, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
ISFSI - ACCIDENTAL CRITICALITY OR LOSS OF SNM 10 CFR 72.74	ISFSI Criticality SNM Loss	The licensee shall notify the NRC Operations Center via ETS within one hour of discovery of accidental criticality or any loss of SNM.	– Unusually high radiation readings discovered in the vicinity of the ISFSI that could indicate possibility of a criticality event
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - SNM SHIPMENTS HBRSEP shall immediately notify the NRC Operations Center via ETS, when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
LOST OR UNACCOUNTED SHIPMENT OF SNM 10 CFR 70.52(b) 10 CFR 73.71(a)(1)	Shipment Loss SNM Spent Fuel Diversion Safeguards Security	HBRSEP shall notify the NRC Operations Center via the ETS within one hour after discovery of any loss of any shipment of SNM or spent fuel or any incident in which an attempt has been made, or is believed to have been made, to commit a theft or unlawful diversion of SNM.	– Shipment Emergency Event – Security Event (SEC-NGGC-2147)
LOST OR UNACCOUNTED SHIPMENT OF SNM - RECOVERY 10 CFR 73.71(a)(1)	Recovery Accounting Shipment SNM Security Safeguards	HBRSEP shall notify the NRC Operations Center via the ETS within one hour after recovery of, or accounting for, any lost shipment of SNM.	

Reference Provided

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FOLLOW-UP

With respect to the telephone notifications made under paragraphs (a) and (b) of 10 CFR 50.72 or paragraphs (a), (b), (c), or (d) of 10 CFR 72.75, in addition to making the required initial notification, HBRSEP shall during the course of the event immediately report:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(1) 10 CFR 72.75(f)(1)	Degradation Emergency Class Change Update Termination ISFSI	(i) any further degradation in the level of safety of the plant or ISFSI or other worsening conditions, including those that require the declaration of any of the Emergency Classes, if such a declaration has not been previously made, or (ii) any change from one Emergency Class to another, or (iii) a termination of the Emergency Class.	– Refer to EPNOT-01
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(2) 10 CFR 72.75(f)(2)	Result Evaluation Effectiveness Unknown ISFSI	(i) the results of ensuing evaluations or assessments of plant or ISFSI conditions, (ii) the effectiveness of response or protective measures taken, and (iii) information related to plant or ISFSI behavior that is not understood.	
FOLLOW-UP NOTIFICATION 10 CFR 50.72(c)(3) 10 CFR 50.72.75(f)(3)	Open Continuous Communication ISFSI	Maintain an open, continuous communication channel with the NRC Operations Center upon request by the NRC.	– Refer to EPNOT-01

Reference Provided

ATTACHMENT 11.1
Page 8 of 9
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

IMMEDIATE (ONE HOUR) NOTIFICATIONS - NRC REGION II OFFICE HBRSEP shall immediately notify the final delivery carrier and, by telephone and telegram, mailgram, or facsimile, the NRC Region II Office when:			
EVENT	**KEY WORDS**	**REQUIREMENT**	**EXAMPLES**
THEFT/UNLAWFUL DIVERSION OF TRITIUM 10 CFR 30.55(c)	Incident Theft Tritium Attempt Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft of more than 10 curies of tritium (outside of spent fuel) at any one time or more than 100 curies of tritium in one calendar year	– 10 Curies of tritium discovered missing from the Chemistry Laboratory, and reason exists to suspect that the tritium was stolen
THEFT/UNLAWFUL DIVERSION OF SOURCE MATERIAL 10 CFR 40.64(c)	Incident Attempt Theft Diversion Source Security Safeguards	Any incident in which an attempt has been made or is believed to have been made to commit a theft or unlawful diversion of more than 15 pounds of Source Material at any one time or 150 pounds of Source Material in any one calendar year	– A source assembly is discovered missing from a new fuel shipment.
SHIPPING PACKAGE RADIOACTIVELY CONTAMINATED 10 CFR 20.1906(d)(1)	Contamination Shipment	Removable radioactive surface contamination exceeds the limits of 10 CFR 71.87	– New or Spent Fuel Shipment Cask arrives with surface contamination in excess of limits.
SHIPPING PACKAGE EXCEEDING EXTERNAL DOSE RATE LIMITS 10 CFR 20.1906(d)(2)	Radiation Dose Rate Shipment	External radiation levels exceeds of the limits of 10 CFR 71.47	– New or Spent Fuel Shipment Cask arrives with external radiation levels in excess of limits.

Reference Provided

IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - FFD The NRC Region II Administrator must be notified immediately by telephone of the following:			
NRC EMPLOYEE NOT FIT FOR DUTY 10 CFR 26.77(c)	Alcohol Influence Substance NRC employee FFD Fitness for Duty	If HBRSEP has a reasonable belief that an NRC employee or NRC contractor may be under the influence of any substance, or is otherwise unfit for duty, the licensee or other entity may not deny access but shall escort the individual. In any such instance, the licensee or other entity shall immediately notify the Region II Administrator by telephone, followed by written notification (e.g., e-mail or fax) to document the oral notification. If the Region II Administrator cannot be reached, the licensee or other entity shall notify the NRC Operations Center.	
IMMEDIATE (ONE HOUR) NOTIFICATIONS TO THE NRC - IAEA The NRC Director, NRR or Director, NMSS must be notified immediately by telephone of the following:			
SURPRISE VISIT OF IAEA OFFICIAL 10 CFR 75.8(c)	IAEA International Atomic Energy Agency Credential	HBRSEP shall immediately communicate by telephone, within one hour with respect to the credentials of any person who claims to be an IAEA representative and shall accept written or electronic confirmation of the credentials from the NRC.	– If the IAEA representative's credentials have not been confirmed by the NRC, the licensee shall not admit the person until the NRC has confirmed the person's credentials. The licensee, shall notify the Commission promptly, by telephone, whenever an IAEA representative arrives at a facility or location without advance notification.

Reference Provided

ATTACHMENT 11.2
Page 1 of 3
FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
SHUTDOWN REQUIRED BY TS 10 CFR 50.72(b)(2)(i)	Shutdown TS Shutdown Power Reduction	The initiation of any shutdown required by the TS.	– Reactor is in MODEs 1 or 2 and the Control Room takes action to reduce power (i.e., negative reactivity insertion) in order to comply with a Required Action to be in MODE 3 within a Completion Time. Reduction in power for some other purpose than compliance with the shutdown requirement is not reportable. MODE changes required by TS when reactor is in MODEs 3, 4, or other non-power conditions, are not reportable. – If allowed outage time plus required shutdown time to MODE 3 is less than the expected restoration time of the LCO and power is reduced in anticipation of the required shutdown, the shutdown is reportable.
ECCS DISCHARGE INTO RCS 10 CFR 50.72(b)(2)(iv)(A)	ECCS Actuation Safety Injection	Any event that results or should have resulted in emergency core cooling system (ECCS) discharge into the reactor coolant system as a result of a valid signal except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	– Manual or automatic Safety Injection System actuation in response to a valid signal that resulted in or should have resulted in discharge into the reactor coolant system.
RPS INITIATION (MANUAL/AUTOMATIC) DURING OPERATION 10 CFR 50.72(b)(2)(iv)(B)	RPS Actuation Reactor Protection System RPS Reactor Trip	Any event or condition that results in actuation of the reactor protection system (RPS) when the reactor is critical except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation.	- Manual or automatic reactor trip from critical through RTP of 100%. Trips which occur as part of planned evolutions in accordance with procedures are not reportable.

Reference Provided

FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

If not reported under paragraphs (a) or (b)(1) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases, within four hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRESS RELEASES AND GOVERNMENT NOTIFICATIONS 10 CFR 50.72(b)(2)(xi) 10 CFR 72.75(b)(2)	News Release Press Radio Television Fatality Environment Public Health and Safety Release ISFSI	Any event or situation, related to the health and safety of the public or on-site personnel, or protection of the environment, for which a news release is planned or notification to other government agencies has been or will be made. Such an event may include an on-site fatality or inadvertent release of radioactively contaminated materials.	– Any News release concerning - A fatality, - Inadvertent release of radioactively contaminated materials to public areas - unusual or abnormal releases of radioactive effluents, or - Information associated with an Emergency Event except when the ERO is activated (EPNOT-01). – Notification to other government agencies concerning: - A fatality on site, - Health and safety of the public or site personnel, - Inadvertent release of radioactively contaminated materials to public areas, - Discovered endangered species kill.

Reference Provided

ATTACHMENT 11.2
Page 3 of 3
FOUR HOUR NOTIFICATIONS TO THE NRC

FOUR HOUR NOTIFICATIONS TO THE NRC

HBRSEP shall notify the NRC Operations Center via ETS as soon as possible but not later than 4 hours after the discovery of any of the following events or conditions involving sources or spent fuel.

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OR THEFT OF LICENSED MATERIAL (>1000X 10 CFR 20 LIMITS) 10 CFR 20.2201	Loss Theft Missing Licensed Radioactive Material Recovery	Immediately notify the NRC, after its occurrence becomes known, any lost, stolen, or missing licensed material in an aggregate quantity equal to or greater than 1,000 times the quantity specified in [10 CFR 20] Appendix C under such circumstances that it appears to HBRSEP that an exposure could result to persons in unrestricted areas. Follow-up written report required within subsequent 30 days. Note – If the lost, stolen, or missing source exceeds a "Quantity of Concern" as specified in HPP-018, then the NRC desires to be notified within 4 hours of any subsequent recovery of the source.	– A radiography source is discovered missing. The source is licensed to the radiography contractor. If the contractor does not make the required notification, HBRSEP should notify the NRC Operations Center via ETS.
ISFSI - DEPARTURE FROM LICENSE CONDITION 10 CFR 72.75(b)(1)	ISFSI Emergency Departure Deviation Health and Safety License Condition	An action taken in an emergency that departs from a condition or a technical specification contained in a license or certificate of compliance issued under 10 CFR 72 when the action is immediately needed to protect the public health and safety and no action consistent with license conditions or technical specifications that can provide adequate or equivalent protection is immediately apparent.	– Action taken in an emergency that departs from procedure that is deemed necessary to prevent releases or radiation doses to the public in excess of 10 CFR 20 limits (See PRO-NGGC-0200).

Reference Provided

EIGHT HOUR NOTIFICATIONS TO THE NRC

EIGHT HOUR NOTIFICATIONS TO THE NRC

If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
PRINCIPAL SAFETY BARRIERS SERIOUSLY DEGRADED 10 CFR 50.72(b)(3)(ii)(A)	Degraded Safety Barriers Fission Product Barrier	Any event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded;	– Fuel cladding failures in the reactor, or in the storage pool, that exceed expected values, or that are unique or widespread, or that are caused by unexpected factors, and would involve a release of significant quantities of fission products – Cracks and breaks in the piping or reactor vessel, or major components in the reactor coolant system that have safety relevance (steam generators, reactor coolant pumps, valves, etc.) – Significant welding or material defects in the RCS – Low temperature overpressure transients where the pressure temperature limits are violated – Loss of relief and/or safety valve functions during operation – Loss of Containment function or integrity - Complete loss of containment integrity function including (1) containment leakage rate greater than allowed value per SR 3.6.1.1 (i.e., entry into LCO 3.6.1 Condition A), (2) loss of containment penetration isolation functional capability (i.e., both barriers), or loss of containment spray capability
UNANALYZED PLANT CONDITION 10 CFR 50.72(b)(3)(ii)(B)	Safety Function Unanalyzed Condition	Any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety.	– OTΔT setpoints are declared inoperable due to summator module lag constants. The channel response time exceeded the value assumed in the accident analysis (analytical limits). – Accumulation of voids in systems designed to remove heat from the reactor that could inhibit the ability to adequately remove heat from the core, particularly under natural circulation conditions. – Any power level excursion above 2346 MWt should be evaluated to determine if the condition posed an unanalyzed condition that significantly degrades nuclear plant safety. Operation slightly in excess of 2346 MWt for short periods are not expected to trigger the 10 CFR 50.72(b)(3)(ii)(B) criterion.

Reference Provided

EIGHT HOUR NOTIFICATIONS TO THE NRC

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If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
LOSS OF EMERGENCY ASSESSMENT, OFF-SITE RESPONSE, OR COMMUNICATIONS CAPABILITY 10 CFR 50.72(b)(3)(xiii)	Selective Signaling System Sirens ETS ERFIS ERDS	Any event that results in a major loss of emergency assessment capability, off-site response capability, or communications capability (e.g., significant portion of control room indication, ETS, or off-site notification system).	– Loss of 15 or more of 59 Public Warning Sirens as indicated on the siren activation system for a period of at least 30 minutes at any one time – Loss of greater than 50% of communications capability (i.e., offsite communications systems which include the Selective Signaling System, the Essex System and the Local Government Radio System) – Loss of Emergency Assessment Capability. This may include planned or unplanned losses of an Emergency Response Facility (ERF). Typically, this would be the TSC but may include the EOF. **(1)** – ETS communications function unavailable. This does not apply to minor interruptions in site or corporate telecommunications systems. It is intended to apply to serious conditions during which the telecommunication system can no longer fulfill the requirements of the Emergency Plan or provide ETS functionality. **(1)** – Loss of commercial telephone system to the extent that required communications could not be made to official offsite locations (e.g., EOCs, Warning Points) – Inoperability of ERFIS and ERDS is not capable of being restored within one hour. **(1)**

(1) See Attachment 11.16, Guidance On Reporting Loss Of Emergency Assessment Or Communications Capability, for additional guidance.

Reference Provided

EIGHT HOUR NOTIFICATIONS TO THE NRC

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If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
RPS/SAFETY SYSTEM INITIATION (MANUAL/AUTOMATIC) 10 CFR 50.72(b)(3)(iv)(A)(B)	Manual Automatic Actuation Engineered Safety Feature ESF Valid Clearance RPS Actuation Reactor Protection System RPS Reactor Trip	Any event or condition that results in valid actuation of any of the systems listed below except when the actuation results from and is part of a pre-planned sequence during testing or reactor operation. The systems to which the requirements of this paragraph apply are: (1) Reactor protection system (RPS) including: reactor scram and reactor trip. (2) General containment isolation signals affecting containment isolation valves in more than one system or multiple main steam isolation valves (MSIVs). (3) Emergency core cooling systems (ECCS) for pressurized water reactors (PWRs) including: high-head, intermediate-head, and low-head injection systems and the low pressure injection function of residual (decay) heat removal systems. (4) PWR auxiliary or emergency feedwater system. (5) Containment heat removal and depressurization systems, including containment spray and fan cooler systems. (6) Emergency AC electrical power systems, including: emergency diesel generators (EDGs)	– Auxiliary Feedwater initiation/actuation – Reactor Trip (Manual/Automatic) while subcritical – Reactor Trip while critical is reportable per Attachment 11.2 – EDG start due to an undervoltage trip signal on emergency bus E1 or E2 – A single train of Containment Isolation actuates – A valid signal for Containment Ventilation Isolation occurs --------------------------------------- Valid actuations are those actuations that result from "valid signals" or from intentional manual initiation, unless it is part of a preplanned test. Valid signals are those signals that are initiated in response to actual plant conditions or parameters satisfying the requirements for the initiation of the safety function of the system. They do not include actuations which are the result of other signals. (NUREG 1022) Invalid actuations are, by definition, those that do not meet the criteria for being valid. Thus invalid actuations include actuations that are not the result of valid signals and are not intentional manual actuations. Except for actuations of the Reactor Protection System (RPS) when the reactor is critical or in MODE 1, invalid actuations are not reportable by telephone under 10 CFR 50.72. In addition, invalid actuations are not reportable under 10 CFR 50.73 in any of the following: - The invalid actuation occurred when the system is already properly removed from service. This means all requirements of plant procedures for removing equipment from service have been met. It includes required clearance documentation, equipment and control board tagging, and properly positioned valves and power supply breakers. - The invalid actuation occurs after the safety function has already been completed. An example would be RPS actuation after the control rods have already been inserted into the core.

Reference Provided

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If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
CONDITION THAT COULD PREVENT FULFILLMENT OF SAFETY FUNCTIONS 10 CFR 50.72(b)(3)(v)	Loss of Safety Function Residual Heat Mitigation Shutdown Generic Setpoint Drift Engineering Evaluation Operability Determination Common Mode Failure	Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to: (A) Shut down the reactor and maintain it in a safe shutdown condition; (B) Remove residual heat; (C) Control the release of radioactive material, or (D) Mitigate the consequences of an accident. Events covered in this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported in accordance with this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.	– Loss (inoperability) of both Trains, e.g., ECCS, Low Temperature Overpressure Protection System, or Lake Robinson water level below LCO 3.7.8 limit - Loss of one Train of required equipment, and the cause of the failure could fail the other train, and there is a reasonable expectation that the other train would not fulfill its safety function if required 1) Contaminated lubrication fluid degrades SI Pump operation (a single condition could prevent fulfillment of a safety function if both trains could be reasonably expected to be inoperable). 2) EDG Air Start Solenoids (if it demonstrates a design, procedural, or equipment deficiency that could prevent the fulfillment of a safety function, i.e., if both diesels are susceptible to same problem) – Multiple Control Rod failures (if failure prevented the fulfillment of a safety function) – Operator action to inhibit the RPS (actions would prevent fulfillment of a safety function)

Reference Provided

ATTACHMENT 11.3
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EIGHT HOUR NOTIFICATIONS TO THE NRC

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If not reported as a declaration of an Emergency Class under paragraph (a) of 10 CFR 50.72, HBRSEP shall notify the NRC Operations Center via ETS as soon as practical and in all cases within eight hours of the occurrence of any of the following:

EVENT	KEY WORDS	REQUIREMENT	EXAMPLES
ISFSI - DEFECT IMPORTANT TO SAFETY 10 CFR 72.75(c)(1)	ISFSI Defect Safety	A defect in any spent fuel storage structure, system, or component that is important to safety.	– A defect discovered in the design or construction of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits.
ISFSI - REDUCTION IN EFFECTIVENESS 10 CFR 72.75(c)(2)	ISFSI Confinement Reduction Effectiveness	A significant reduction in the effectiveness of any spent fuel storage cask confinement system during use.	– Wear or degradation of ISFSI units that could result in releases or radiation doses to the public in excess of 10 CFR 20 limits.
TRANSPORT OF CONTAMINATED INJURED PATIENT 10 CFR 50.72(b)(3)(xii) 10 CFR 72.75(c)(3)	Contaminate Injured Person Medical Transport Rescue Hospital ISFSI	Any event requiring the transport of a radioactively contaminated person to an off-site medical facility for treatment.	– Any event requiring the transport of a radioactively contaminated or potentially contaminated (NUREG 1022) person to an off-site medical facility for treatment.