DUKE ENERGY®

H.B. Robinson

ILC-13 NRC Licensing Exam
Written Exam
Book 1 of 2

ROBINSON

1. 008 AK2.03 001
Given the following plant conditions:

-Plant is at 100% RTP

-RCS pressure is 2280 psig

-PC-444J, PZR PRESSURE, is in Automatic at 65% ouput

-PCV-455A, PZR SPRAY 444G, controller is in Automatic at 25% ouput

-PCV-455C, PZR PORV, partially opens

-The OAC takes action and manually closes PCV-455C

-**NO** further operator actions have been taken

-Current RCS pressure is 2150 psig

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

PCV-455A, PZR SPRAY 444G, controller output indication currently reads ___(1)___ .
PC-444J output indication currently reads ___(2)___.

A. (1) 25%
(2) 65%

B. (1) 25%
(2) 0

C. (1) 0
(2) 65%

D.✓ (1) 0
(2) 0

The correct answer is D.

A) Incorrect. PCV-455A is still in automatic. PC-444J will close this spray valve based on the RCS pressure, causing the controller output to read 0. Plausible if the student doesn't realize that PC-444J controls this valve. PC-444J will not read 65%. Plausibile because this controller would read 65% if it was in manual.

B) Incorrect. PCV-455A is still in automatic. PC-444J will close this spray valve based on the RCS pressure, causing the controller output to read 0. Plausible if the student doesn't realize that PC-444J controls this valve or if they don't realize that this valve is in automatic. PC-444J would read 0 is correct.

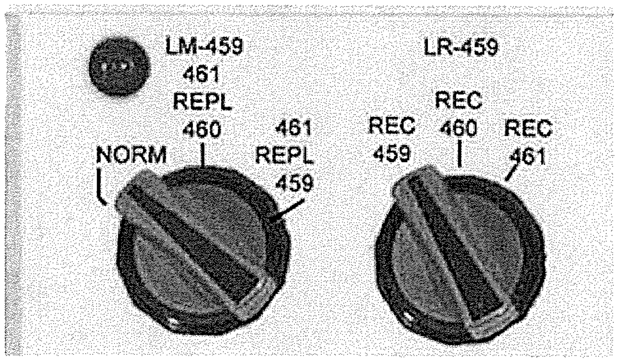
C) Incorrect. PCV-455A controller output will read 0. PC-444J will not read 65%. Plausibile because this controller would read 65% if it was in manual.

D) Correct. PCV-455A is still in automatic. PC-444J will close this spray valve based on the RCS pressure, causing the controller output to read 0. PC-444J will read 0 is correct. This controller is in automatic and will read 0 based on RCS pressure being 2150 psig.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 2.5 SRO 2.4
K/A: 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
AK2: Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following:
AK2.03: Controllers and positioners
References: Sim/Plant design, Student Text for PZR/PRT
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 8 of PZR/PRT Lesson Plan
Question Source: New
Question History: New
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.7/45.7)

Meets the K/A because the student is required to know how the controllers for the spray valves function based on RCS pressure.

Figure 12 Level Transmitter Switches



PRESSURE

There are eight pressure transmitters on the PZR.

1. PT-445

   - provides a signal for operating power operated relief valve PCV-456.
   - provides a high and low pressure alarm.
   - provides pressure indication on the RTGB.

2. PT-444

   - provides pressure indication on the RTGB and at the motor driven auxiliary feedwater pump station.
   - provides signal to a proportional - plus - reset controller (PC-444J) on the RTGB.
   - provides signal to PC-444J for operating power operated relief valve PCV-455C.
   - provides signal to PC-444J for spray valves PCV-455A and B.
   - provides signal to PC-444J for heater control and high controller output alarm.

3. PT-455, 456, and 457

Are the three PZR pressure protection channels that supply the following to reactor protection and safeguards:

2. 009 EK1.02 001

Given the following plant conditions:

-"A" EDG is OOS

-Reactor power is at 100% RTP

Subsequently:

-A fault on 4Kv bus 2 leads to a reactor trip

-Upon the reactor trip, a small break LOCA occurs

-The crew is in EOP-E-1, LOSS OF REACTOR OR SECONDARY COOLANT

-CETC's read 547°F

-Max. CV pressure reached was 3.4 psig

-RCS Wide Range Pressure reads 1292 psig

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

The RCS subcooling is ___(1)___ . Based off the indications above, RCP trip criteria for EOP-E-1 Foldout is ___(2)___ .

A. (1) 29.6°F
(2) **NOT** met

B. (1) 31°F
(2) met

C. (1) 29.6°F
(2) met

D.✓ (1) 31°F
(2) **NOT** met

The correct answer is D.

A) Incorrect. 29.6°F is incorrect. Plausible if the student did not convert PSIG to PSIA. RCP Trip criteria is not met in this scenario. The RCP trip criteria is not met. Plausible with this subcooling if the students are unsure of the RCP trip criteria requiring at least one SI pump and capable of delivering flow to the core AND RCS subcooling being < than 30°F. If they think this is an OR statement, this would be correct.

B) Incorrect. 31°F is the correct subcooling. The RCP trip criteria is not met. Plausible with this subcooling if the students are unsure of the RCP trip criteria requiring at least one SI pump and capable of delivering flow to the core AND RCS subcooling being < than 30°F. If they think this is an OR statement, this would be correct. Also, the candidate could think that adverse numbers are in effect due to CV pressure at 3.4 psig. The adverse number for subcooling with respect to RCP Trip Criteria is 50°F.

C) Incorrect. 29.6°F is incorrect. Plausible if the student did not convert PSIG to PSIA. RCP trip criteria is not met. Plausible since there is at least one SI pump running and the students calculated subcooling is <30°F.

D) Correct. 31°F is the correct subcooling. The RCP trip criteria is not met. The scenario has at least one SI pump running and capable of delivering flow and the RCS Subcooling is >30°F.

RCP trip criteria out of EOP-E-1:
"BOTH of the following satisfied:

-SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW

AND

-RCS subcooling based on core exit TCs - LESS THAN 30°F [50°F]"

The Subcooling limit recently changed from 35°F to 30°F.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.5 SRO 4.2
K/A: 009 Small Break LOCA
EK1: Knowledge of the operational implications of the following concepts as they apply to the small break LOCA:
EK1.02: Use of steam tables
References: Sim/Plant design, EOP-E-1 Foldout criteria
Proposed references to be provided to applicants during the Exam: Steam Tables
Learning Obj: Objective 6 for EOP-E-1 lesson plan
Question Source: New
Question History: New
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.8/41.10/45.3)

Meets the K/A because the students have to use the steam tables to calculate subcooling, then based off that subcooling, determine whether or not the RCP's should be tripped. Tripping or not tripping the RCP's is the operational implication part of the K/A.

1. RCP TRIP CRITERIA
   IF either condition listed below occurs, THEN trip all RCPs:
   - Containment Isolation Phase B - ACTUATED

     OR
   - BOTH of the following satisfied:
     - SI pumps - AT LEAST ONE RUNNING AND CAPABLE OF DELIVERING FLOW

       AND
     - RCS subcooling based on core exit TCs - LESS THAN 30°F [50°F]

2. SI TERMINATION CRITERIA
   IF all conditions listed below occur, THEN reset SPDS and Go To EPP-7, SI TERMINATION, Step 1:
   - RCS subcooling based on core exit TCs - GREATER THAN 35°F [55°F]
   - Total feed flow to intact S/Gs - GREATER THAN 300 gpm

     OR

     Narrow range level in at least one intact S/G - GREATER THAN 8% [18%]
   - RCS pressure - GREATER THAN 1650 PSIG [1700 PSIG]
   - RCS pressure - STABLE OR RISING
   - PZR level - GREATER THAN 14% [31%]

3. SI REINITIATION CRITERIA
   IF either condition listed below occurs, THEN manually start SI and RHR pumps as necessary to restore RCS subcooling and PZR level:
   - RCS subcooling based on core exit TCs - LESS THAN 35°F [55°F]

     OR
   - PZR level - CANNOT BE MAINTAINED GREATER THAN 14% [31%]

4. SECONDARY INTEGRITY CRITERIA
   IF any S/G pressure is lowering in an uncontrolled manner OR has completely depressurized AND that S/G has NOT been isolated, THEN reset SPDS and Go To EOP-E-2, FAULTED STEAM GENERATOR ISOLATION, Step 1.

5. EOP-E-3 TRANSITION CRITERIA
   IF any S/G level rises in an uncontrolled manner OR has abnormal radiation, THEN perform the following:
   a. Manually start SI and RHR pumps as necessary.
   b. Reset SPDS and Go To EOP-E-3, STEAM GENERATOR TUBE RUPTURE, Step 1.

6. COLD LEG RECIRCULATION SWITCHOVER CRITERIA
   IF RWST level lowers to less than 27%, THEN reset SPDS and Go To EPP-9, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

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

3. 011 EG2.4.21 001

Given the following plant conditions:

-A Seismic Event coincident with a Large Break LOCA has occurred

-CV pressure is currently 11 psig

-"B" CV Spray Pump has tripped

-All other ESF components operated properly

-CV Sump level is 380 inches and rising

-CV Radiation levels are <5 r/hr

FRP-J.1, RESPONSE TO HIGH CONTAINMENT PRESSURE
FRP-J.2, RESPONSE TO CONTAINMENT FLOODING

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

Based off the indications above, the crew meets entry conditions for ____(1)____. One of the Major Action Categories of this procedure is ____(2)____ .

A. (1) FRP-J.1
(2) Verify Containment Isolation and Heat Removal

B. (1) FRP-J.1
(2) Check for and Isolate Faulted Steam Generator

C. (1) FRP-J.2
(2) Determine the radioactivity level of the sump fluid

D. (1) FRP-J.2
(2) Try to Identify Unexpected Source of Sump Water and Isolate It if Possible

The correct answer is D.

A) Incorrect. FRP-J.1 is the incorrect procedure. Plausible if students are unaware of the CV sump level requirements. The second part is incorrect. Plausible since this is a Major Action Category from FRP-J.1.

B) Incorrect. FRP-J.1 is the incorrect procedure. Plausible if students are unaware of the CV sump level requirements. The second part is incorrect. Plausible since this is a Major Action Category from FRP-J.1.

C) Incorrect. The crew meets FRP-J.2 entry requirements based on CV Sump Level being >375 inches. The second part is incorrect. Plausible because this is a step in FRP-J.2.

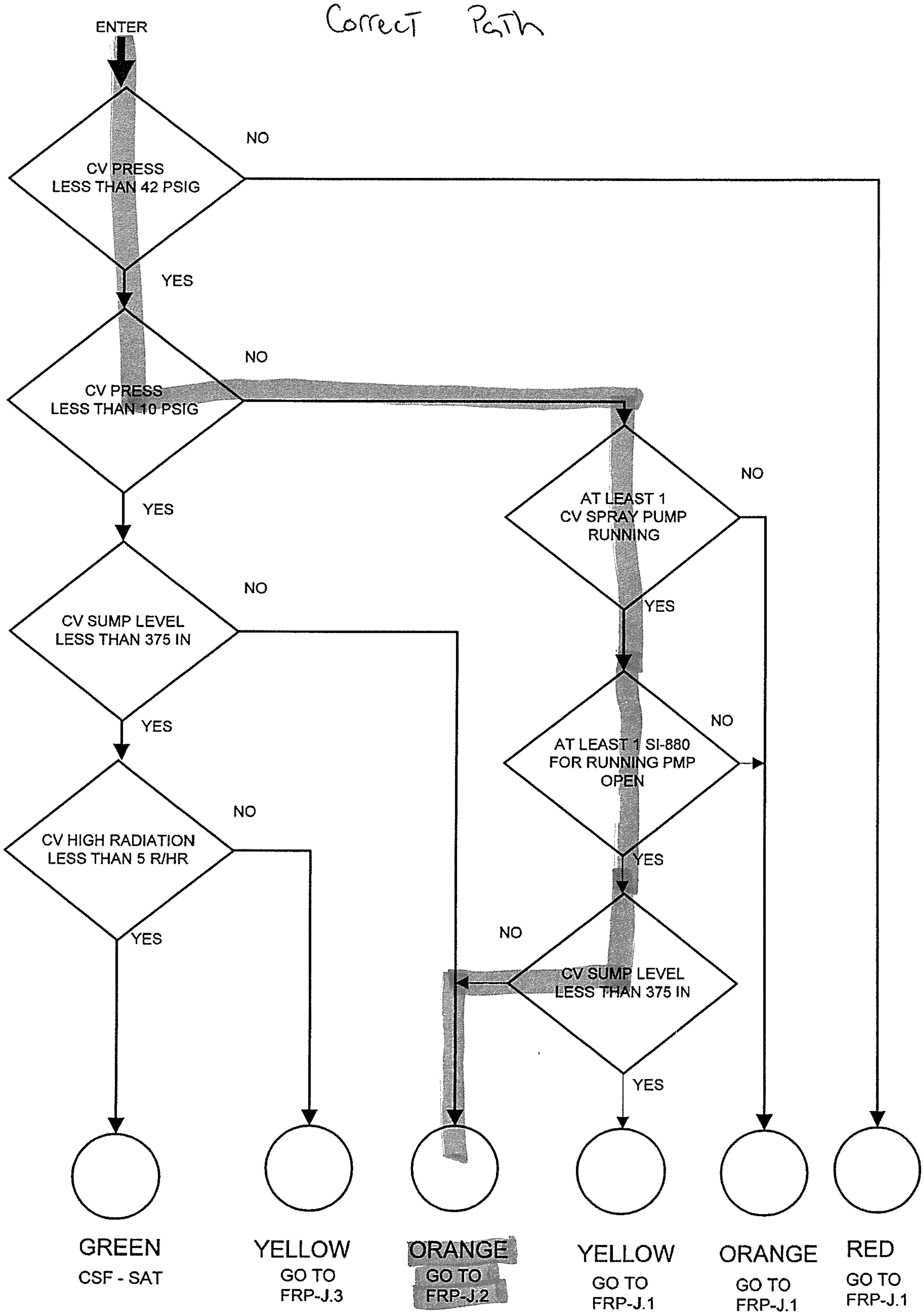
D) Correct. The crew meets FRP-J.2 entry requirements based on CV Sump Level being >375 inches. The second half is correct, this is a Major Action Category from FRP-J.2

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.0 SRO 4.6
K/A: 011 Large Break LOCA
G2.4.21: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
References: Sim/Plant design, CSFST's, FRP-J.1 BD, FRP-J.2 BD
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 2 for FRP-J.2 lesson plan
Question Source: New
Question History: New
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.7/43.5/45.12)

This meets the K/A because the student has to determine the correct FRP to enter based on their knowledge of parameters and logic in the scenario.

CSF-5, CONTAINMENT

Correct Path



Correct

DISCUSSION (From the WOG FR-Z.2 Basis Document)

1. INTRODUCTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, is a Function Restoration Guideline (FRG) that provides procedural guidance when the containment level is greater than flood level.

There is only one explicit transition to guideline FR-Z.2. It is from the Critical Safety Function Status Tree F-0.5, CONTAINMENT, on an ORANGE priority when containment sump level is greater than flood level.

After all the actions in guideline FR-Z.2 are completed, the operator is instructed to return to the guideline and step in effect.

2. DESCRIPTION

Guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING, provides actions to respond when the containment level is greater than design flood level. This level is significant since the critical systems and components, which are necessary to ensure an orderly safe plant shutdown and provide feedback to the operator regarding plant conditions, are normally located above the design flood level. Therefore, the guideline FR-Z.2 is entered from the Containment Status Tree on an ORANGE priority when this design flood level is exceeded.

The primary purpose of the containment sump area is to collect the water injected into the containment or spilled from the reactor coolant system following an accident. The water collected in the containment sump is then available for long term core and/or containment cooling via the emergency core cooling or containment spray recirculation systems. In addition, the containment sump collects the injected or spilled water into areas such that vital systems or components will not be flooded and thus rendered inoperable.

The maximum level of water in the containment following a major accident generally is based upon the entire water contents of the reactor coolant system, refueling water storage tank, condensate storage tank, and SI accumulators. This water volume approximates the maximum water volume introduced into the containment following a LOCA plus a steamline or feedline break inside containment.

An indicated water level in the containment greater than the maximum expected volume (design basis flood level) is an indication that water volumes other than those represented by the above noted volumes have been introduced into the containment. Also, the high water level provides an indication that potential flooding of critical systems and components needed for plant recovery may occur.

The actions in this guideline attempt to identify any unexpected source of water and isolate it if possible. Beyond that the plant engineering staff is consulted to determine if transfer of containment sump water to other tanks is appropriate.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-Z.2 is to provide actions to respond to containment flooding.

The following subsections provide a summary of the major action categories of operator actions and the key utility decision points for guideline FR-Z.2, RESPONSE TO CONTAINMENT FLOODING.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-Z.2 is given below in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN FR-Z.2

- o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible
- o Notify Plant Engineering Staff of Sump Level and Activity Level

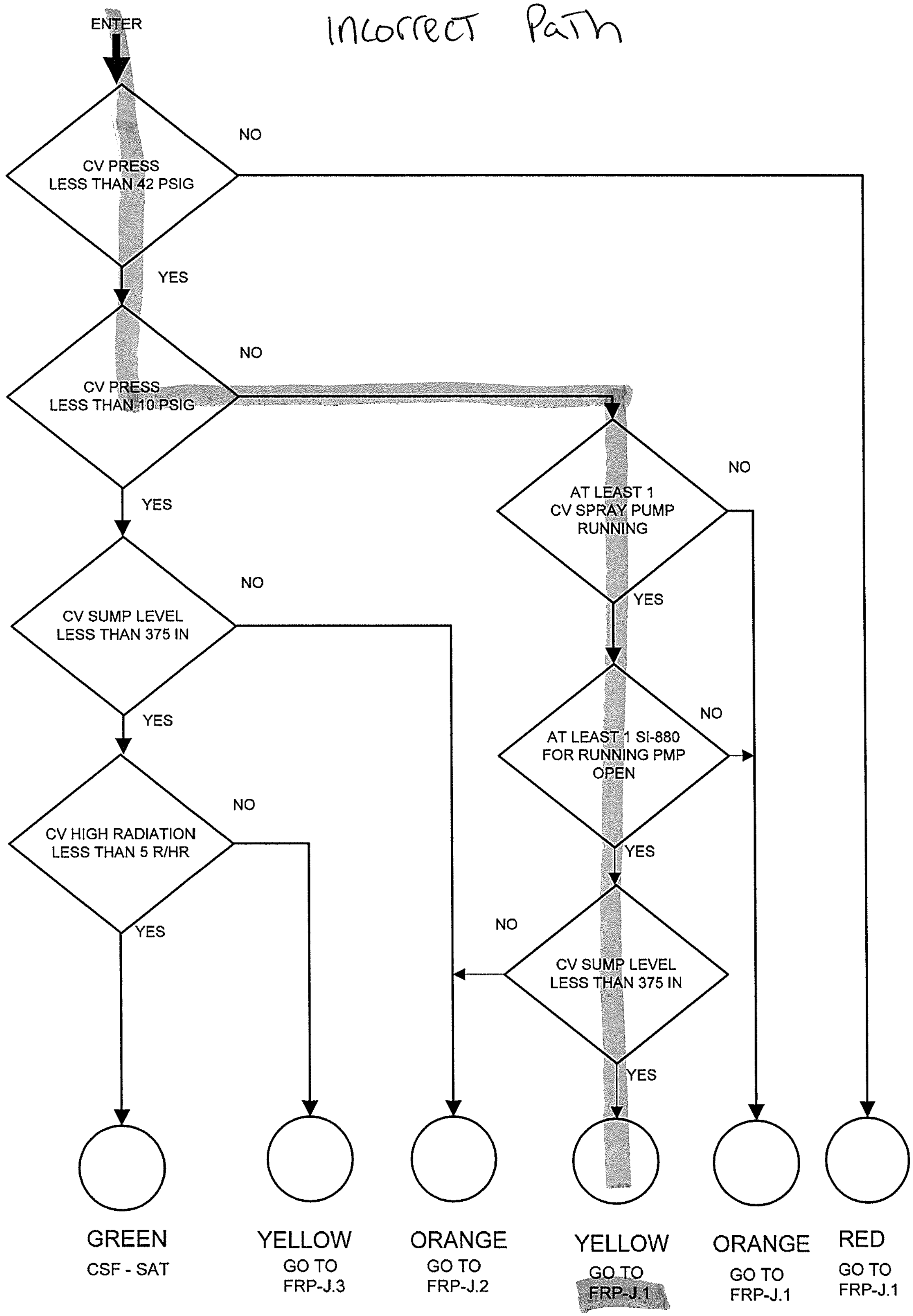
o Try to Identify Unexpected Source of Sump Water and Isolate It if Possible

The first action in this guideline is to try to identify the source of water which is causing containment flooding and isolate it. The concern regarding flooding is that critical plant components needed for plant recovery could be damaged and rendered inoperable.

o Notify Plant Engineering Staff of Sump Level and Activity Level

By knowing the sump level and activity level, the plant engineering staff can determine if the excess water can be transferred to storage tanks located outside containment.

Incorrect Path



3.2 Key Utility Decision Points incorrect for Part c(2)

There is one key utility decision point in this guideline when the utility must determine an appropriate course of action. In Step 3, the operator informs the plant engineering staff of the sump level and activity level in the sump to obtain a recommended action. The plant engineering staff will direct any further actions to be taken by the operator to address the containment water level. Actions to prevent further flooding or to transfer sump water to storage tanks may be provided.

STEP SPECIFIC DESCRIPTION AND RNP DIFFERENCES

The following pages will provide the RNP step number, the ERG step number, the WOG basis for each step where applicable, the differences between the ERG and RNP step, and the Category of deviation (SSD).

RNP STEP	WOG STEP	BASIS/DIFFERENCES
PEC	PEC	WOG BASIS N/A, there is no WOG basis description for the PEC, other than the general description. RNP DIFFERENCES/REASONS No significant differences. SSD DETERMINATION This is not an SSD.
1-7	1	WOG BASIS To identify unexpected source of water in sump BASIS: This step instructs the operator to try to identify the unexpected source of the water in the containment sump. Containment flooding is a concern since critical plant components necessary for plant recovery may be damaged and rendered inoperable. A water level greater than the design basis flood level provides an indication that water volumes other than those represented by the emergency stored water sources (e.g., RWST, accumulators, etc.) have been introduced into the containment sump. Typical sources which penetrate containment are service water, component cooling water, primary makeup water and demineralized water. All possible plant specific sources which penetrate containment should be included in this step. These systems provide large water flow rates to components inside the containment and a major leak or break in one of these lines could introduce large quantities of water into the sump. Identification and isolation of any broken or leaking water line inside containment is essential to maintaining the water level below the design basis flood level. RNP DIFFERENCES/REASONS The RNP step has been split into multiple steps in order to provide the plant specific list as directed by the ERG. The first item added by the RNP step is for a faulted or ruptured S/G. This could be a large source of water to the CV, especially if the S/G has not yet been isolated. If a faulted or ruptured S/G is present the procedure verifies it has been isolated. Note that for a faulted S/G, the UFSAR assumes that the S/G has been isolated at no later than 10 minutes into the event. The RNP step has provided direction to stop all RCPs prior to isolating CCW into the CV to protect the RCPs. RNP has provided a check for a Fire Main and Primary Water leak as part of the plant specific list. SSD DETERMINATION This is an SSD per criterion 4 and 10.

Incorrect Major Action Categories

already contains steps for checking hydrogen concentrations and starting the recombiners.

Note that for the design basis large LOCA, hydrogen generation is postulated to occur due to a small amount of cladding oxidation, radiolysis of water in the core region, and oxidation of aluminum and zinc in the containment. The hydrogen buildup from these sources is very slow (typically on the order of several days to several weeks) and is already addressed by the hydrogen control steps in E-1 and other guidelines where a LOCA may exist (e.g., ES-1.2, ECA-3.1, ECA-3.2). Addressing hydrogen in FR-Z.1 for the reference plant is not appropriate since there is no relationship between high containment pressure and a containment challenge due to a hydrogen burn.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-Z.1 is to provide actions to respond to an excessively high containment pressure.

The following subsections provide a summary of the major action categories of operator actions for guideline FR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-Z.1 is given below in the form of major action categories. These are discussed below in more detail.

MAJOR ACTION CATEGORIES IN FR-Z.1

- o Verify Containment Isolation and Heat Removal
- o Check for and Isolate Faulted Steam Generator

o Verify Containment Isolation and Heat Removal

The first group of steps in this guideline ensures that the automatic actions, which are very important during a high containment pressure condition, have occurred. They include containment isolation Phase A, containment ventilation isolation, containment spray actuation and containment isolation Phase B, operation of containment fan coolers and main steamline isolation. The operator should verify these automatic actions have occurred and if not, he should take manual action.

o Check for and Isolate a Faulted Steam Generator

High containment pressure could result from a steamline break inside containment. Therefore, if a steamline break is detected, the operator is instructed to isolate the faulted steam generator(s) to try to eliminate the release of mass and energy which is causing the high containment pressure.

STEP SPECIFIC DESCRIPTION AND RNP DIFFERENCES

The following pages will provide the RNP step number, the ERG step number, the WOG basis for each step where applicable, the differences between the ERG and RNP step, and the Category of deviation (SSD).

PEC PEC WOG BASIS

N/A, there is no WOG basis description for the PEC, other than the general description.

RNP DIFFERENCES/REASONS

No significant differences.

SSD DETERMINATION

This is not an SSD.

4. 022 AK3.07 001

Given the following plant conditions:

-The following conditions are at time 12:30:00:

-The Reactor is at 100% RTP

-All control systems are in normal alignments

-Charging flow has been rising and is currently at 50 gpm

-TI-140, REGEN HX LTDN OUTLET TEMP, is lowering

-VCT level is lowering

-The following conditions are at time 12:45:00:

-RCS Pressure is 2235 psig

-Charging pump discharge pressure is 2100 psig

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

The leak is located __(1)__ of the Regenerative Heat Exchanger.

The crew takes action to close HIC-121, Charging Flow, IAW AOP-018, REACTOR COOLANT PUMP ABNORMAL CONDITIONS, Section C, LOSS OF SEAL INJECTION. The purpose of doing this is to __(2)__.

A.✓ (1) downstream
(2) isolate the charging line leak

B. (1) downstream
(2) maintain minimum RCP Seal Injection Flow

C. (1) upstream
(2) isolate the charging line leak

D. (1) upstream
(2) maintain minimum RCP Seal Injection Flow

The correct answer is A.

A) Correct. The leak is downstream of the Regenerative Heat Exchanger. This is indicated by the TI-140 lowering. AOP-018 Basis document has you close HIC-121 if there is a rupture. There is a rupture and this is indicated by Charging pump discharge pressure being less than RCS Pressure.

B) Incorrect. The leak is downstream of the Regenerative Heat Exchanger. The purpose is not to maintain minimum RCP Seal Injection Flow. Plausible since you do shut HIC-121 in other parts of AOP-018 to maintain minimum RCP Seal Injection Flow.

C) Incorrect. The leak is not upstream of the Regenerative Heat Exchanger. Plausible if the students do not understand the indications given to them. AOP-018 Basis document has you close HIC-121 if there is a rupture. There is a rupture and this is indicated by Charging pump discharge pressure being less than RCS Pressure.

D) Incorrect. The leak is not upstream of the Regenerative Heat Exchanger. Plausible if the students do not understand the indications given to them. The purpose is not to maintain minimum RCP Seal Injection Flow. Plausible since you do shut HIC-121 in other parts of AOP-018 to maintain minimum RCP Seal Injection Flow.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.0 SRO 3.2
K/A: 022 Loss of Reactor Coolant Makeup
AK3: Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:
AK3.07: Isolating charging
References: Sim/Plant design,AOP-018 and its Basis Document
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of AOP-018
Question Source: New
Question History: New
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.5/41.10/45.6/45.13)

Meets the K/A because the questions asks the reason for isolating charging during a loss of reactor coolant makeup, which is why do you shut HIC-121.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

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NOTE

"Charging Line" in the subsequent step is defined as the Charging Line downstream of HCV-121, CHARGING FLOW valve.

*26. **Check Location Of Break - IN CHARGING LINE**

RNO: IF break is found in the Charging Line, THEN Observe the NOTE prior to Step 27 and Go To Step 27.

Go To Step 33.

NOTE

Throttling any RCP SEAL WATER FLOW CONTROL VALVE with Charging isolated will only shift flow to the other RCPs AND raise Charging Pump discharge pressure.

27. **Perform The Following:**

a. Close HIC-121, CHARGING FLOW

b. Fully Open the RCP SEAL WATER FLOW CONTROL VALVEs

- CVC-297A
- CVC-297B
- CVC-297C

28. **Check RCP SEAL WATER FLOW CONTROL VALVEs - FULLY OPEN**

RNO: WHEN the RCP SEAL WATER FLOW CONTROL VALVEs are fully open, THEN Go To Step 29.

Correct

Step	Description	Section C
27-29	These steps are reached if a break has been diagnosed on the Charging Line downstream of HIC-121. This break may be isolated by closing HIC-121 and Seal Injection restored. HIC-121 is closed to isolate the source of the leakage (partially isolates, the local actions started at the previous step should continue to provide full isolation). The RCP Seal Water Flow Control valves are fully opened to assure system head is low enough to preclude lifting the Charging Pump Discharge Safety valves. Following start of a pump at minimum speed no attempt is made to throttle high flows to assure less than 13 gpm on each RCP. In this case total flow is checked and maintenance notified to adjust minimum speed on the pump.	
30	Following restart of the Charging Pump and restoration of Seal Injection, the makeup System is restarted to provide normal makeup .	
31	Excess Letdown is placed in service in order to provide a means of inventory control prior to restart of a Charging Pump. An attachment has been provided instead of OP-301-1 due to conflicts with this procedure and the section of OP-301-1 that places Excess Letdown in service.	
32	This step transitions the operator to sections of the procedure to return the system to normal.	
33	This step is reached should a leak have been diagnosed other than downstream of HCV-121. The step acts as a hold-point until the leak location has been identified because subsequent actions are based on whether or not the leak can be isolated. The 1st part of the step provides for the hold point with the appropriate transitions for the leak location in the RNO. Note that the term "readily" identified is used in the RNO. This term is used in order to allow CRS judgment for the time spent looking for the leak location vs the plant conditions present while looking for the leak. It is expected that if plant conditions are seriously degrading, not much time will be available. The leak location will determine the amount of time required to locate the leak. It is expected that leaks inside the Auxiliary Building will be easily identified. Leaks inside the CV, however, may take an excessive amount of time to identify. For leaks that take an excessive amount of time to identify the CRS should move on in the procedure. If subsequent diagnosis shows the leak to be isolable, then actions may be taken to restore charging/seal injection from outside of this procedure or via reentry to the procedure.	

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

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33.	Check RCP Seal Injection Flow - BETWEEN 8 GPM AND 13 GPM	Locally throttle RCP SEAL WATER FLOW CONTROL VALVE(s) to obtain flow to each RCP between 6 gpm and 20 gpm. • CVC-297A • CVC-297B • CVC-297C IF required to maintain minimum flow, THEN throttle HIC-121, CHARGING FLOW Valve while maintaining Charging Pump Discharge pressure less than 2500 PSIG.

NOTE

Localized boiling in the affected RCP Thermal Barrier may cause FCV-626, THERM BAR FLOW CONT Valve, to close.

34.	**Check FCV-626, THERM BAR FLOW CONT Valve - CLOSED**	Go To Step 40.
*35.	**Check Thermal Barrier Cooler - INTACT**	WHEN this procedure is completed, THEN Go To AOP-014, Component Cooling Water System Malfunction. Go To Step 40.
36.	**Check APP-002-D2, CV ISOL PHASE B Alarm - EXTINGUISHED**	Go To Step 40.

Incorrect

Step	Description	SECTION A

25-26 Low RCP #1 Seal Leakoff flow is also an indication of abnormal conditions. This step checks if the #1 Seal Leakoff flow is less than 0.8 gpm (alarm setpoint). If flow exists below this rate, these steps provide instruction for more frequent RCP monitoring and to notify Engineering personnel for Westinghouse consultation.

27 This step provides diagnostic instruction for a #3 Seal problem (and also #2 Seal problem). If a standpipe level alarm exists, then either of these seals could have a problem.

28-29 These steps will differentiate between #2 Seal and #3 Seal problems and provide the actions and transitions needed. A #2 Seal problem of a magnitude necessary to cause the high standpipe alarm should have already been noted and addressed under low #1 Seal leakoff steps. The step will provide a transition to the section dealing with high #2 Seal Leakoff to serve as a back up for the previous section. Should the high alarm NOT be present, then a low alarm is present. Repeated low alarms are a symptom of a failed #3 Seal. The RNO provided direction to contact Engineering to notify them of Seal Status.

30-32 This series of steps begins the cleanup section of the procedure that restores the system to a somewhat normal configuration. Tripping RCP B or C will affect Pressurizer Spray flow. If both RCP B and C are running and one of the pumps is tripped, the PZR Spray valve on the loop for the tripped pump must be closed to prevent diverting spray flow from the operating loop to the non-operating loop. These steps will provide the diagnostics and actions to assure that the idle loop spray valve is closed if its pump is stopped while the other loop remains running.

33 Normal operation of the RCP Seal Injection is checked in this step as a final check prior to returning to the normal operating procedures. If required, the RNO provides instructions to restore RCP Seal Injection to the expanded operating parameters allowed by OP-301-1, Chemical and Volume Control System (Infrequent Operation), which states "it is not necessary to readjust RCP seal injection flows to normal range of 8 to 13 GPM for evolutions which will only last for several hours, provided the seal injection flow is maintained within 6 to 20 GPM for RCP continuous operation" (CR 473019). If HIC-121 is required to be throttled, the Operator must monitor Charging Pump discharge pressure and maintain below 2500 psig to prevent lifting the Charging Pump Discharge Safety valves. (CR 95-1752)

N34 This note reminds the Operator of conditions that may have caused FCV-626 to close. For large #1 Seal failures, this is expected to occur.

5. 025 AK2.02 001

Given the following plant conditions:

-The plant is in mid loop operation to repair a S/G primary manway leak

-RCS level is -68 inches and rising very slowly

-RHR pump "A" is in service at 3500 gpm

-The operator notices that RHR flow and pressure are oscillating

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

IAW AOP-020, LOSS OF RESIDUAL HEAT REMOVAL (SHUTDOWN COOLING), the crew will reduce RHR flow to ______ in an attempt to stabilize the RHR oscillations.

A. 3250 gpm

B. 3000 gpm

C. 2800 gpm

D.✓ 1500 gpm

The correct answer is D.

A) Incorrect. Plausible since this is a flow that you can throttle RHR flow to IAW AOP-020 (3000-3750). Done when the cause of the loss of an RHR pump is known and the crew restarts the pump.

B) Incorrect. Plausible since this is a flow that you throttle RHR flow to IAW AOP-020. Done when the cause of the loss of an RHR pump is known and the crew restarts the pump.

C) Incorrect. Plausible since this is a flow that you throttle RHR flow to IAW AOP-020. This flow is used when the loss of an RHR pump is known the crew restarts the pump and the RCS level is below reduced inventory.

D) Correct. AOP-020 has you reduce RHR flow to 1500 GPM.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.3 SRO 3.2
K/A: 025 Loss of Residual Heat Removal System (RHRS)
AK2: Knowledge of the interrelations between the Loss of Residual Heat Removal System and the following:
AK2.02: LPI or Decay Heat Removal/RHR pumps
References: Sim/Plant design, AOP-020 and its basis document
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3 for AOP-020 lesson plan
Question Source: New
Question History: New
Question Cognitive Level: Low/Memory
10 CFR part 55: (CFR 41.7/45.7)

Meets the K/A because the student must know where to reduce flow to in order to try and stop the RHR pumps from cavitating.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

****

CAUTION

Changes in RCS pressure may result in inaccuracies in RCS Loop Standpipe indications.

****

NOTE

Cavitation will be indicated by oscillations in RHR flow and pressure, accompanied by noises from the RHR Pumps. (SOER 88-03, Rec 3)

22. **Determine If RHR Pump Cavitation Is Occurring As Follows:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
a. Check RCS level - ABOVE -72 INCHES (69% RVLIS FULL RANGE)	a. Perform the following: 1) Verify both RHR Pumps STOPPED. 2) Go To Section A, Loss Of RHR While At Reduced Inventory.
b. Check the following RHR indications - CAVITATION PRESENT • FI-605, RHR TOTAL FLOW AND • Running RHR Pump Discharge Pressure Indication o RHR Pump A - PI-602A o RHR Pump B - PI-602B	b. Go To Step 24.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

(Page 13 of 37)

****

CAUTION

While fuel is in the reactor and unborated water may be injected into the RCS, an operating RCP or RHR flow greater than 2800 gpm is required to ensure the assumptions of the boron dilution analysis are satisfied. (Ref UFSAR 15.4.6.2)

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
23.	**Perform the following:**	
	a. Adjust FC-605, RHR HX BYPASS FLOW Controller, to reduce RHR flow to 1500 gpm.	
	b. Check Cavitation - ELIMINATED	b. Perform the following: 1) Initiate CV Closure Using OMM-033, CV Closure. 2) Verify both RHR Pumps STOPPED. 3) Go To 58.
	c. Attempt to Adjust FC-605, RHR HX BYPASS FLOW Controller, to raise RHR flow to greater than 2800 gpm.	
	d. Check RHR flow status • Flow greater than 2800 gpm AND • Cavitation Eliminated	d. Perform the following: 1) Reduce RHR flow to 1500 gpm 2) Refer to Technical Specifications • ITS 3.4.6, 3.4.7, 3.4.8, 3.9.4, 3.9.5
	e. Return to procedure and step in effect	

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	(CONTINUED)	
	f. Check current RCS level - ABOVE REDUCED INVENTORY	f. Perform the following: 1) Slowly adjust FC-605 to obtain 1500 gpm flow on FI-605, RHR TOTAL FLOW 2) Establish 2800 gpm to 3000 gpm RHR Total Flow as follows: a) Set FC-605 potentiometer to 0.5 b) Place FC-605, RHR HX BYPASS FLOW Controller in AUTO c) Slowly adjust FC-605 potentiometer 2800 to 3000 gpm flow on FI-605, RHR TOTAL FLOW 3) Go To Step 7.i
	g. Adjust FC-605, RHR HX BYPASS FLOW Controller, To Restore Flow Between 3000 gpm And 3750 gpm	
	h. Restore FC-605, RHR HX BYPASS FLOW Controller to AUTO as follows: 1) Adjust FC-605 potentiometer to 1.5 2) Place FC-605, RHR HX BYPASS FLOW Controller in AUTO 3) Adjust FC-605 potentiometer establish RHR flow between 3000 gpm And 3750 gpm	

(CONTINUED NEXT PAGE)

6. 027 AA1.02 001

Given the following plant conditions:

-The reactor is at 100% RTP

-PI-444, CH I PRZR PRESS, reads 2310 psig and rising

-PI-455, 456 and 457, PROT CH I (II and III) PRZR PRESS, all read approximately 2210 psig and lowering

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

To combat this, the OAC will take PC-444J to MANUAL and __(1)__ its output to energize all PZR heaters.

After PC-444J is adjusted, PI-458, Calibration PZR Pressure, will indicate __(2)__ psig for maximum proportional heater output.

A. (1) raise
(2) -15

B. (1) raise
(2) +15

C✓ (1) lower
(2) -15

D. (1) lower
(2) +15

The correct answer is C.

A) Incorrect. RCS pressure will lower due to the heaters being deenergized and spray valves open. The OAC will lower the output on PC-444J to shut the spray valves get the heaters full on. Plausible if the students are unsure of which way PC-444J gets controlled for heaters and spray control. The second half of distractor is correct.

B) Incorrect. The OAC will lower the output on PC-444J to shut the spray valves and to get the heaters full on. Plausible if the students are unsure of which way PC-444J gets controlled for heaters/spray/PORV control. Candidate may think that PI-458 will indicate a positive value to indicate maximum proportional heater output.

C) Correct. The OAC will lower the output on PC-444J in order to shut the PORV and restore pressure with the heaters being full on. PI-458 will inidate -15 psig for a maximum proportional heater output.

D) Incorrect. The first part of the distractor is correct. The second half of the distractor is plausible if the candidate thinks that PI-458 will indicate a positive value to inidicate maximum proportional heater output.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.1 SRO 3.0
K/A: 027 Pressurizer Pressure Control System (PZR PCS) Malfunction
AA1: Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions:
AA1.02: SCR-controlled heaters in manual mode
References: Sim/Plant design,PZR Student Text, AOP-025
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 2 AOP-025 Lesson Plan, Objective 3 of PZR/PRT Lesson Plan
Question Source: New
Question History: New
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.7/45.5/45.6)

Meets the K/A because the student has to determine which way to operate PC-444J in order to control the heaters appropriately in manual mode.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

Pressurizer Pressure Transmitter Failure

(Page 1 of 2)

> **NOTE**
>
> Steps 1 and 2 are Immediate Action steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Determine If PZR PORVs Should Be Closed:**	
	a. Check PZR pressure - LESS THAN 2335 PSIG	a. Verify OPEN at least one PZR PORV and associated PORV BLOCK Valve: • PCV-455C AND RC-536 OR • PCV-456 AND RC-535 WHEN RCS pressure is less than 2335 psig, THEN perform Step 1.b. Go To Step 2.
	b. Verify Both PZR PORVs - CLOSED	b. IF any PZR PORV can NOT be closed, THEN close its PORV BLOCK Valve.
2.	**Control The PZR SPRAY VALVES AND PZR Heaters To Restore RCS Pressure To The Desired Control Band**	
3.	**Make PA Announcement For Procedure Entry**	

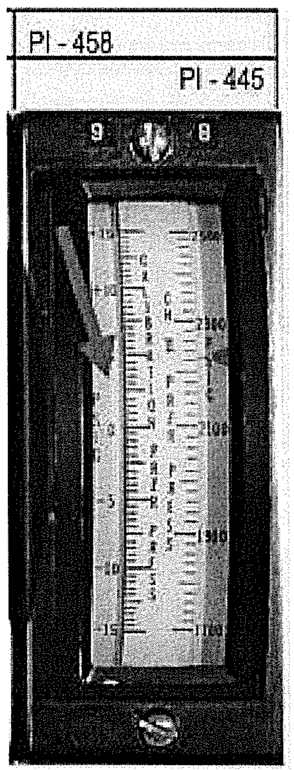
PRESSURE ERROR SIGNAL

1. PI-458, Calibration PZR Pressure, scale is +15 psig to -15 psig

Indicates pressure error signal to the proportional heaters Proportional heaters are either full on or full off. As pressure error signal decreases, i.e. less positive or more negative, the fraction of time the heaters are on (the period) increases.

- For a + 9 psig error signal the period is 0.45
- For a + 0 psig error signal the period is 0.71
- For a minus 9 psig error signal the period is 0.89

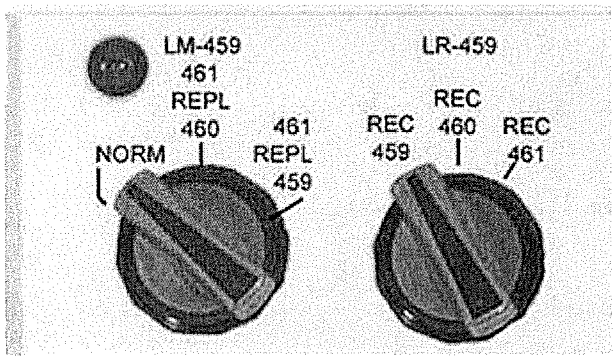
Figure 15 Pressure Error Signal Indicator



SAFETY VALVE POSITION INSTRUMENTATION

An acoustic accelerometer is mounted on the outlet of each of the three safety valves. These accelerometers monitor the sound of fluid passing through the valves and provide electrical signals proportional to the flows. These signals are used to provide relative valve position indications in the Cable Spread Room on Instrument Cabinet "A" and RTGB.

Figure 12 Level Transmitter Switches



PRESSURE

There are eight pressure transmitters on the PZR.

1. PT-445
   - provides a signal for operating power operated relief valve PCV-456.
   - provides a high and low pressure alarm.
   - provides pressure indication on the RTGB.
2. PT-444
   - provides pressure indication on the RTGB and at the motor driven auxiliary feedwater pump station.
   - provides signal to a proportional - plus - reset controller (PC-444J) on the RTGB.
   - provides signal to PC-444J for operating power operated relief valve PCV-455C.
   - provides signal to PC-444J for spray valves PCV-455A and B.
   - provides signal to PC-444J for heater control and high controller output alarm.
3. PT-455, 456, and 457

Are the three PZR pressure protection channels that supply the following to reactor protection and safeguards:

7. 029 EA1.02 001

Given the following plant conditions:

-The crew attempted to trip the reactor and it failed to trip

-They are initiating emergency boration of the RCS

-The Boric Acid Pump aligned for blend failed to start

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

To initiate emergency boration, the crew will open __(1)__ . The emergency boration can be secured when PR Channels are <5% ___(2)___ a Negative IR Startup Rate.

A. (1) MOV-350
(2) or

B. (1) MOV-350
(2) and

C.✓ (1) LCV-115B and close LCV-115C
(2) and

D. (1) LCV-115B and close LCV-115C
(2) or

The correct answer is C.

A) Incorrect. MOV-350 cannot be used since the Boric Acid Pump is not running. Plausible since this is the first choice in methods to emergency borate per FRP-S.1. PR Channels are <5% or Negative Startup Rate is incorrect. Both need to be met per FRP-S.1. Plausible since a negative SUR alone will clear all Orange and Red terminals on your CSFST.

B) Incorrect. MOV-350 cannot be used since the Boric Acid Pump is not running. Plausible since this is the first choice in methods to emergency borate per FRP-S.1. The second part is correct. FRP-S.1 requires both PR Channels <5% and a Negative SUR.

C) Correct. LCV-115B is the correct valve to open since the Boric Acid Pump is not running so they can't use the preferred method of MOV-350. LCV-115C needs to be shut or you will not get flow from the RWST to the Charging pump suction, it will be blocked by the flow from the VCT. FRP-S.1 requires both PR Channels <5% and a Negative SUR.

D) Incorrect. LCV-115B is the correct valve to open since the Boric Acid Pump is not running so they can't use the preferred method of MOV-350. LCV-115C needs to be shut or you will not get flow from the RWST to the Charging pump suction, it will be blocked by the flow from the VCT. PR Channels are <5% or Negative Startup Rate is incorrect. Both need to be met per FRP-S.1. Plausible since a negative SUR alone will clear all Orange and Red terminals on your CSFST.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.6 SRO 3.3
K/A: 029 Anticipated Transient Without Scram (ATWS)
EA1: Ability to operate and monitor the following as they apply to a ATWS:
EA1.02: Charging pump suction valves from RWST operating switch
References: Sim/Plant design, FRP-S.1
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 6 of FRP-S.1 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.7/45./45.6)

Meets the K/A because the student needs to determine from the conditions which Charging pump suction valves from the RWST to operate to initiate emergency boration and what conditions to monitor to secure the emergency boration during an ATWS.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Initiate Emergency Boration Of RCS As Follows:**	
	a. Verify Charging flowpath established as follows:	
	1) CVC-310B, LOOP 2 COLD LEG CHG - OPEN	1) Open CVC-310A, LOOP 1 HOT LEG CHG.
	2) HIC-121, CHARGING FLOW Controller - DEMAND SIGNAL AT 0%	
	b. Verify At Least Two Charging Pumps - RUNNING AT FULL SPEED	
	c. Verify Boric Acid Pump aligned for blend - RUNNING Correct incorrect	c. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT <u>OR</u> • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET. 3) Go To Step 4.f.

(CONTINUED NEXT PAGE)

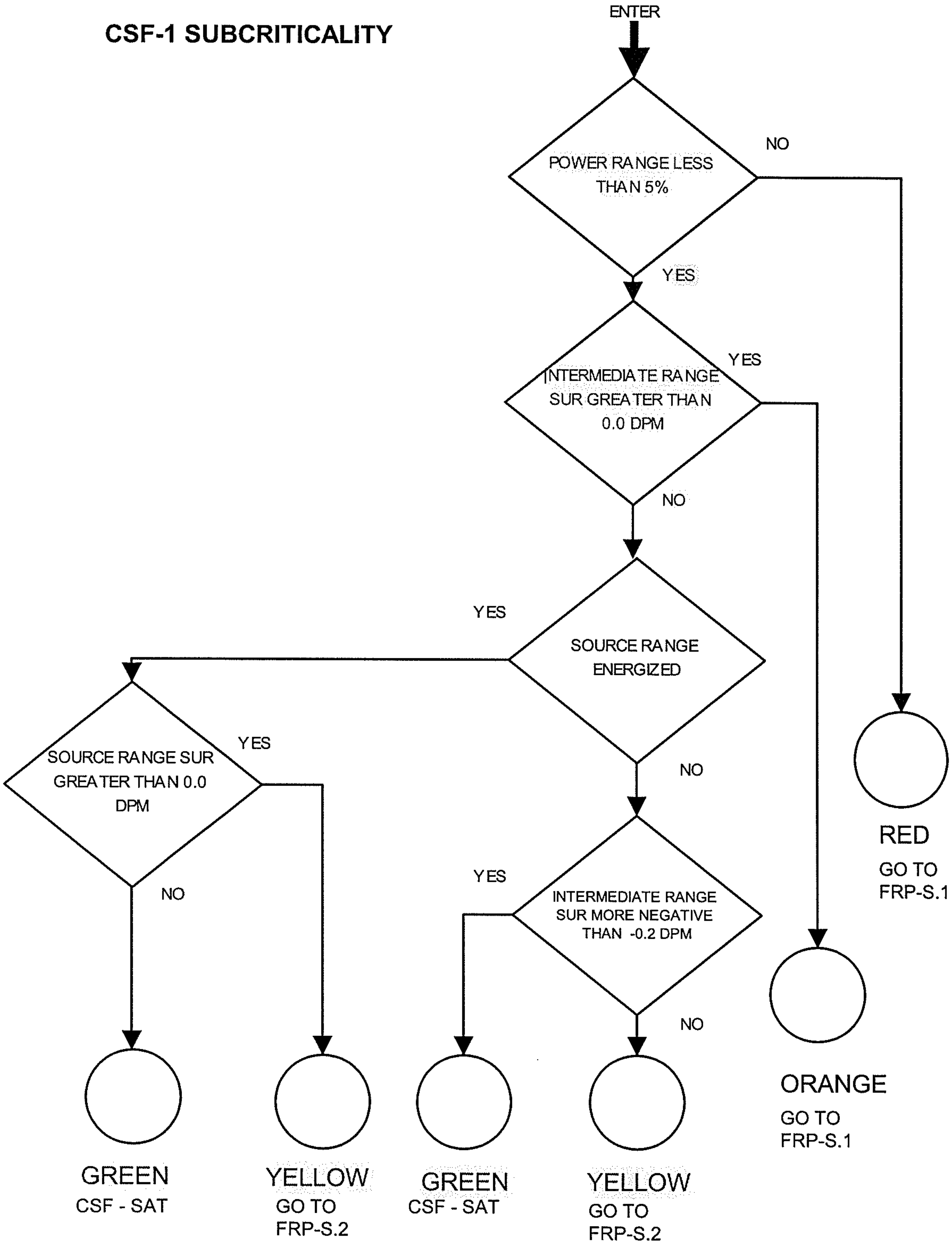
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	(CONTINUED) incorrect	
	d. Verify from the RTGB MOV-350, BA TO CHARGING PMP SUCT - OPEN	d. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT OR • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET. 3) Go To Step 4.f.
	e. Check flow on FI-110, BORIC ACID BYPASS FLOW - FLOW INDICATED	e. Perform the following: 1) Open one of the following valves: • LCV-115B, EMERG MU TO CHG SUCT OR • CVC-358, RWST TO CHARGING PUMP SUCTION (locally) 2) Close LCV-115C, VCT OUTLET.
	f. Verify Charging Flow to RCS on FI-122A	

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
5.	**Check CONTAINMENT VENTILATION ISOLATION - INITIATED**	Perform the following: a. Depress H.V. OFF on R-11 OR R-12 to initiate Containment Ventilation Isolation. b. IF any CONTAINMENT VENTILATION ISOLATION Valve fails to close, THEN locally verify penetration is isolated from outside CV.
* 6.	**Check SI - INITIATED**	IF An SI Signal occurs, THEN verify auto start of all SI equipment using Supplement L, while continuing with this procedure. Go To Step 8
7.	**Verify Auto Start Of All SI Equipment Using Supplement L, While Continuing With This Procedure**	
8.	**Check If The Following Trips Have Occurred:**	
	a. Reactor Trip	a. Continue attempts to locally trip Reactor.
	b. Turbine Trip	b. Locally trip Turbine using the Turbine Trip Assembly at Front Standard.
* 9.	**Check If Reactor Is Subcritical:**	
	a. Check Power Range Indication - LESS THAN 5%	a. Go To Step 10.
	b. Check Intermediate Range Indication - NEGATIVE SUR	b. Go To Step 10.
	c. Go To Step 23.	
*10.	**Check CST Level - LESS THAN 10%**	IF CST level lowers to less than 10%, THEN perform Step 11. Go To Step 12.

Correct

CSF-1 SUBCRITICALITY



8. 038 EK3.08 001

Which ONE (1) of the following identifies the reason for securing the RCP's in EOP-E-3, STEAM GENERATOR TUBE RUPTURE?

A. Minimizes the impact upon core heat removal.

B.✓ It insures against possible operator misdiagnosis.

C. Prevents excessive depletion of RCS water inventory.

D. Prevents the possiblity of RCP motor overspeed and catastrophic failure.

The correct answer is B.

A) Incorrect. Plausible because this is correct for a Small Break LOCA

B) Correct. EOP-E-3 Basis document says "RCP trip is required to ensure core cooling for certain small LOCA sizes and conditions. Although RCP trip to ensure core cooling is not necessary for a steam generator tube rupture, RCP trip is required if the specified criteria are met to insure against possible operator misdiagnosis, operator error, or a multiple failure event scenario."

C) Incorrect. Plausible because this is correct for a Small Break LOCA

D) Incorrect. Plausible because this is correct for a LBLOCA.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.1 SRO 4.2
K/A: 038 Steam Generator Tube Rupture (SGTR)
EK3: Knowledge of the reasons for the following responses as the apply to the SGTR:
EK3.08: Criteria for securing RCP
References: Sim/Plant design,
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 8 of EOP-E-3 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.5/41.10/45.6/45.13)

Meets the K/A because the question asks the student the reason for tripping the RCP's during a SGTR.

Correct

EOP STEP: 1 WOG STEP: 1

STEP

Check If RCPs Should Be Stopped:

PURPOSE:

To trip RCPs if required conditions are satisfied.

BASIS:

RCP trip is required to ensure core cooling for certain small LOCA sizes and conditions. Although RCP trip to ensure core cooling is not necessary for a steam generator tube rupture, RCP trip is required if the specified criteria are met to insure against possible operator misdiagnosis, operator error, or a multiple failure event scenario.

KNOWLEDGE:

This step is a continuous action step.

The operator should know the importance of RCP trip when established criteria are exceeded. See document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume.

The RCP trip criteria applies until an operator controlled RCS cooldown is initiated.

DEVIATIONS:

Added plant specific substep a. to transition past the remaining substeps if NO RCPs are running to enhance procedure usage while remaining consistent with the ERG intent to stop all RCPs.

(Continued on next page)

Correct

EOP STEP: 1 WOG STEP: 1

STEP

Check If RCPs Should Be Stopped:

PURPOSE:

To trip RCPs if required conditions are satisfied.

BASIS:

RCP trip is required to ensure core cooling for certain small LOCA sizes and conditions. Although RCP trip to ensure core cooling is not necessary for a steam generator tube rupture, RCP trip is required if the specified criteria are met to insure against possible operator misdiagnosis, operator error, or a multiple failure event scenario.

KNOWLEDGE:

This step is a continuous action step.

The operator should know the importance of RCP trip when established criteria are exceeded. See document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume.

The RCP trip criteria applies until an operator controlled RCS cooldown is initiated.

DEVIATIONS:

Added plant specific substep a. to transition past the remaining substeps if NO RCPs are running to enhance procedure usage while remaining consistent with the ERG intent to stop all RCPs.

(Continued on next page)

Incorrect

EOP STEP: 13 WOG STEP: 22

STEP

Check If RCPs Should Be Stopped:

PURPOSE:

To trip RCPs if required conditions are satisfied.

BASIS:

During accident conditions, there are some situations which warrant RCP trip if the RCPs are running. During the initial stages of a small break loss of coolant accident (SBLOCA), if selected parameter setpoints are reached, the RCPs should be tripped to avoid more serious impacts.

Continuous operation of the RCPs during a LOCA cannot be guaranteed since tripping of the RCPs would occur upon a loss of offsite power or other essential support conditions which can be postulated to occur at any time. The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident. The RCPs should be tripped before RCS liquid inventory is depleted to the point where tripping of the pumps would cause the break to immediately uncover.

For large break LOCAs, the operation of the RCPs has little if any effect during mitigation and recovery. During the initial phases of a large LOCA, the RCPs are continuously powered for some minimum time period to avoid the possibility of RCP motor overspeed since this could lead to the possibility of flywheel fracture and the attendant missile generation problems. For SBLOCAs, the primary concern is not one of mechanical stability of the component but rather one of RCS coolant inventory and the impact upon core heat removal.

Refer to document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume.

KNOWLEDGE:

Importance of RCP trip when required conditions are satisfied.

This step is on the Foldout Page for this procedure.
(therefore, it is not required to be a continuous action step)
(Continued on next page)

Incorrect

EOP STEP: 1

WOG STEP: 1

STEP

Check If RCPs Should Be Stopped:

PURPOSE:

To trip RCPs if required conditions are satisfied.

BASIS:

During accident conditions, there are some situations which warrant RCP trip if the RCPs are running. During the initial stages of a small break loss of coolant accident (SBLOCA), if selected parameter setpoints are reached, the RCPs should be tripped to avoid more serious impacts.

Continuous operation of the RCPs during a LOCA cannot be guaranteed since tripping of the RCPs would occur upon a loss of offsite power or other essential support conditions which can be postulated to occur at any time. The reason for purposely tripping the RCPs during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCPs were tripped for some reason later in the accident. The RCPs should be tripped before RCS liquid inventory is depleted to the point where tripping of the pumps would cause the break to immediately uncover.

For large break LOCAs, the operation of the RCPs has little if any effect during mitigation and recovery. During the initial phases of a large LOCA, the RCPs are continuously powered for some minimum time period to avoid the possibility of RCP motor overspeed since this could lead to the possibility of flywheel fracture and the attendant missile generation problems. For SBLOCAs, the primary concern is not one of mechanical stability of the component but rather one of RCS coolant inventory and the impact upon core heat removal.

Refer to document RCP TRIP/RESTART in the Generic Issues section of the ERG Executive Volume.

KNOWLEDGE:

Importance of RCP trip when required conditions are satisfied.

This step is on the Foldout Page for this procedure.
(therefore, it is not required to be a continuous action step)
(Continued on next page)

9. 040 AA2.02 001

Given the following plant conditions:

-The reactor is at 50% RTP

-A small steam line leak has been reported

-The CRS implements Attachment 10.4, CONTROL BAND AND TRIP LIMIT GUIDANCE, from OMM-022, EMERGENCY OPERATING PROCEDURES USER'S GUIDE

-APP-006-A2, S/G A STM>FW FLOW is in alarm

IAW Attachment 10.4, at what NR S/G level is the crew required to manually trip the reactor?

A. 16%

B. 21%

C. 30%

D.✓ 35%

The correct answer is D.

A) Incorrect. Plausible since this is the automatic reactor trip on S/G low-low level.

B) Incorrect. IAW Attachement 10.4, 21% is the trip value if stm flow>feed flow annunciator was not in.

C) Incorrect. Plausible since this is the automatic reactor trip on S/G low level coincident with stm flow>feed flow.

D) Correct. IAW Attachment 10.4, they will trip at 35% with stm flow>feed flow annunciator lit.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.6 SRO 4.7
K/A: 040 Steam Line Rupture
AA2: Ability to determine and interpret the following as they apply to the Steam Line Rupture:
AA2.02: Conditions requiring a reactor trip
References: Sim/Plant design, OMM-022 Attachment 10.4
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 7E of OMM-022 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 43.5 / 45.13)

Meets the K/A because the student needs to know when to manually initiate a reactor trip.

CONTROL BAND AND TRIP LIMIT GUIDANCE

OMM-022 Section 8.3.1 allows operator manual control during certain plant situations. Control bands and trip limits should be established during these times of manual control. Once the plant is stable the OAC/BOP will repeat the bands and limits in use per this guidance. Shift Manager and CRS concurrence for these pre-determined bands and limits is not required. If the control bands and trip limits are not appropriate for the current plant conditions, then the CRS should establish the appropriate bands and limits per normal protocol.

Parameter	Control Band	Trip Limit	
		Low	High
Steam Generator Level	47-57% incorrect correct	21% and Lowering (Without Steam > Feed annunciator) 35% and Lowering (with a Steam > Feed annunciator)	70% and Rising
Rod Control Stable Plant	T_{AVG} within 2° of T_{REF}	$T_{AVG} \leq 10°$ of T_{REF}	$T_{AVG} \geq 10°$ of T_{REF}
Rod Control Transient Plant	T_{AVG} within 5°of T_{REF}	$T_{AVG} \leq 10°$ of T_{REF} and not under operator control	$T_{AVG} \geq 10°$ of T_{REF} and not under operator control
Pressurizer Level	Within 5% of Reference Level	10%	86%
Pressurizer Pressure	2220 – 2260 PSIG	2050 PSIG	2350 PSIG

 Incorrect

an unnecessary reactor trip for those pressure increases that can be controlled by the PORVs. This trip occurs when 2 out of 3 PZR Pressure Signals exceed the trip setpoint.

b. Setpoint - PC-455A, PC-456A, PC-457A/**2376 psig**

12. Low PZR Pressure Trip

a. The Low PZR Pressure Trip provides protection against violating the DNBR limit due to low pressure. This trip occurs when 2 out of 3 PZR Pressure Signals decreases below the trip setpoint. This trip is automatically blocked below 10% (P-7) and enabled greater than P-7.

This trip is dynamically compensated based on the rate of change in pressure.

b. Setpoint -

- PC-455C, PC-456C, PC-457C
- PM-455A, PM-456A, PM-457A
- Trip Setpoint **1844 psig**
- Lead Time Constant 10 sec.
- Lag Time Constant 1 sec.

13. High PZR Water Level Trip

a. The High PZR Water Level Trip provides a back-up to the High PZR Pressure Trip and prevents the PZR Safety and Relief Valves from relieving water for credible accident conditions. This trip occurs when 2 out of 3 PZR Water Level Signals exceeds the trip setpoint. This trip is automatically blocked below 10% (P-7).

b. Setpoint - LC-459A, LC-460A, LC-461A/**91% of span**

14. Steam/Feedwater Flow Mismatch with Low S/G Level Trip

a. The Steam/Feedwater Flow Mismatch Trip provides protection for the Reactor against an anticipated Loss of Heat Sink. This trip occurs when 1 out of 2 flow elements sense that Feedwater Flow is less than Steam Flow

Incorrect

and 1 out of 2 Steam Generator(S/G) Level Elements decrease below the setpoint in any S/G.

b. Setpoint - FC-478A, FC-478B/**0.64 x 10^6 lbs/Hr**

- FC-488A, FC-488B/**0.64 x 10^6 lbs/Hr**
- FC-498A, FC-498B/**0.64 x 10^6 lbs/Hr**
- AND
- LC-474B, LC-475B/**30% of Span**
- LC-484B, LC-485B/**30% of Span**
- LC-494B, LC-495B/**30% of Span**

15. S/G Low-Low Water Level Trip

a. The S/G Low-Low Water Level trip ensures protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the S/G tubes. The S/Gs are the heat sink for the reactor. In order to act as a heat sink, the S/Gs must contain a minimum amount of water. A narrow range low low level in any S/G is indicative of a loss of heat sink for the reactor. This trip occurs when 2 out of 3 S/Gs Narrow Range Level Elements on 1 out of 3 S/Gs decrease below the setpoint.

b. Setpoint -

- LC-474A, LC-475A, LC-476A/**16%** of Span
- LC-484A, LC-485A, LC-486A/**16%** of Span
- LC-494A, LC-495A, LC-496A/**16%** of Span

16. Safeguards Signal Trip

a. The Engineered Safeguards Signal Trips ensure that the Reactor will be shut down during a severe accident. This trip is initiated if the Engineered Safety Features Actuation System is automatically or manually actuated.

b. Setpoint - Refer to ST-006, Engineered Safety Features System

17. Turbine Trip/Reactor Trip

10. 054 AK1.01 001

Given the following plant conditions:

-The reactor is at 100% RTP

-A large Feedwater Line Break occurs at the piping connection to S/G "A" inside the CV

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

Prior to the Reactor trip, RCS Tavg will __(1)__. The reactor will automatically trip on __(2)__ .

A.✓ (1) rise
(2) Low-Low S/G level

B. (1) lower
(2) Low-Low S/G level

C. (1) rise
(2) Low S/G Level with Steam Flow > Feed Flow

D. (1) lower
(2) Low S/G Level with Steam Flow > Feed Flow

The correct answer is A.

A) Correct. RCS Tavg will rise due to there being less feedwater getting to the S/G resulting in lower heat removal. The Reactor will trip on Low-Low S/G level.

B) Incorrect. RCS Tavg will rise due to there being less feedwater getting to the S/G. Plausible since RCS Tavg would lower if this break was at the top of the S/G. The Reactor will trip on Low-Low S/G level.

C) Incorrect. RCS Tavg will rise due to there being less feedwater getting to the S/G. The Reactor will trip on Low-Low S/G level. Plausible since the S/G could trip on this if the feedwater break was upstream of the feedflow transmitters.

D) Incorrect. RCS Tavg will rise due to there being less feedwater getting to the S/G. Plausible since RCS Tavg would lower if this break was at the top of the S/G. The Reactor will trip on Low-Low S/G level. Plausible since the S/G could trip on this if the feedwater break was upstream of the feedflow transmitters.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.1 SRO 4.3
K/A: 054 Loss of Main Feedwater (MFW)
AK1: Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW):
AK1.01: MFW line break depressurizes the S/G (similar to a steam line break)
References: Sim/Plant design, Student Text for RPS
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 10B of the RPS Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR part 55: (CFR 41.8 / 41.10 / 45.3)

Meets the K/A because the student needs to know what RCS Tavg will change because of the break and what will trip the Reactor because of this break.

Purpose and Entry Conditions

(Page 2 of 4)

2. **SYMPTOMS AND ENTRY CONDITIONS**

a. The following are symptoms that require a reactor trip, if one has not occurred:

REACTOR TRIP SIGNAL	LOGIC (INTERLOCK)	SETPOINT
SR High Flux	1/2 (P-6 and P-10)	10^5 CPS
IR High Flux	1/2 (P-10)	Current equal to 25%
PR High Flux Low Range	2/4 (P-10)	24%
PR High Flux High Range	2/4	108%
PZR High Pressure	2/3	2376 psig
PZR Low Pressure	2/3 (P-7)	1844 psig
PZR High Level	2/3 (P-7)	91%
Low RCS Flow	2/3 on 2/3 (P-7) 1/3 (P-8)	94.68% rated flow
RCP Breaker	1/1 on 2/3 (P-7) 1/3 (P-8)	Open
RCP Bus Undervoltage	2/3 busses (P-7)	75% (3120 volts)
Over Temperature ΔT	2/3	Variable
Over Power ΔT	2/3	Variable
Safety Injection	Auto or Manual	N/A
Turbine Trip	2/2 SV or 2/3 AST (P-8)	N/A
Low-Low S/G Level	2/3 on 1/3	16%
Low S/G Level With Steam Flow > Feed Flow	1/2 on 1/3 1/2 on 1/3	30% 0.64×10^6 lbm/hr

Correct

incorrect

(CONTINUED NEXT PAGE)

11. 055 EG 2.4.21 001

Given the following plant conditions:

-The crew has just finished their immediate actions of EPP-1, LOSS OF ALL AC POWER

-Narrow Range S/G levels are all 6%

-Total FW Flow to the S/G's is 500 GPM

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

CSFST's ___(1)___ being monitored for information only. A red path ___(2)___ exist on CSF-3, HEAT SINK.

A. (1) are **NOT**
(2) does **NOT**

B.✓ (1) are
(2) does **NOT**

C. (1) are **NOT**
(2) does

D. (1) are
(2) does

Correct

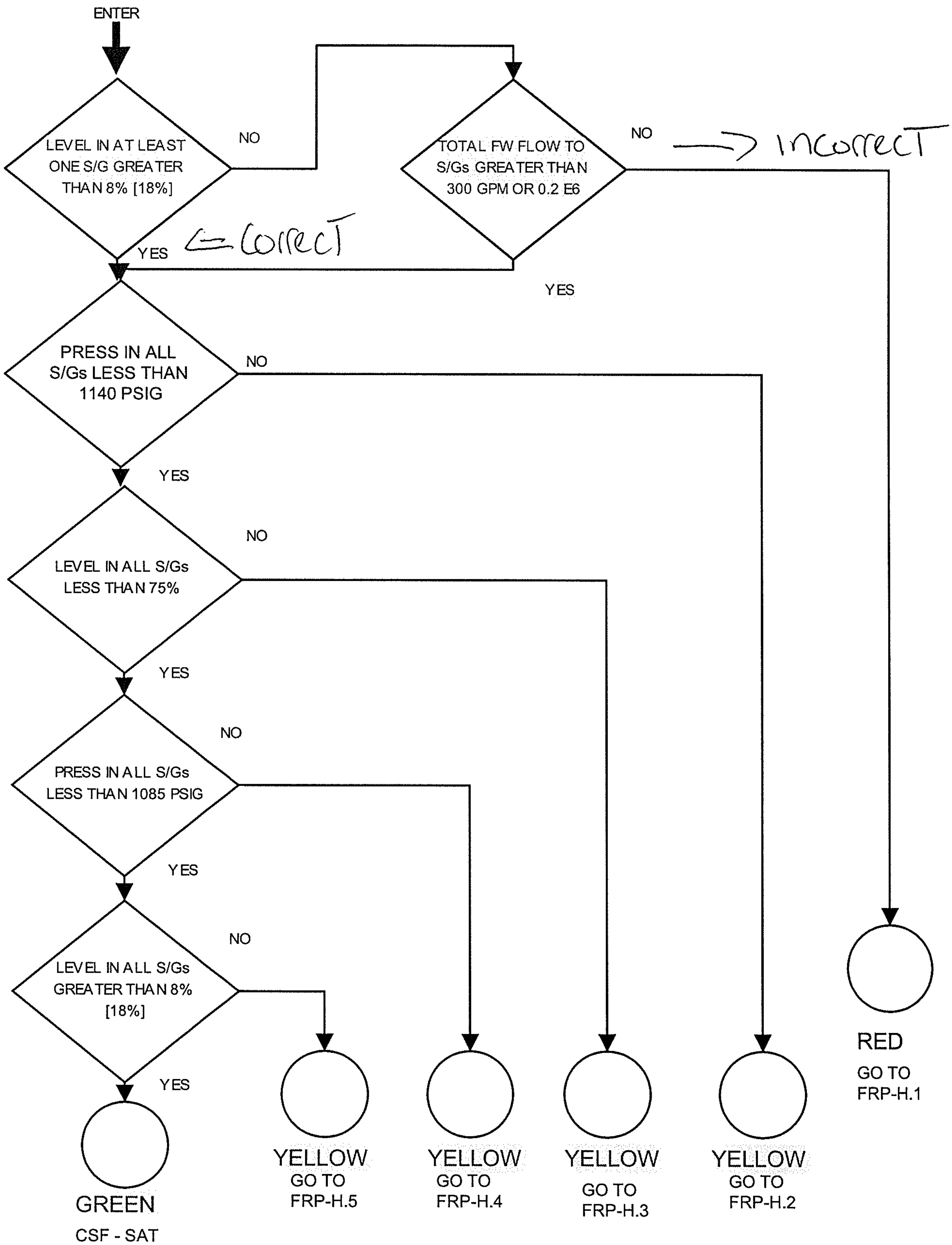
STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- Steps 1 through 3 are Immediate Action steps.
- Critical Safety Function Status Trees are monitored for information only. This procedure is not exited to implement any Functional Restoration Procedure.
- Foldouts and concurrent AOPs should not be implemented during EPP-1, unless implementation is directed by EPP-1, to prevent diluting available resources.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Reactor Trip:** • Check REACTOR TRIP MAIN AND BYP BKRs - OPEN • Check Neutron flux - LOWERING	Trip Reactor.
2.	**Check Both Turbine Stop Valves - CLOSED**	From the RTGB, trip the Turbine. <u>IF</u> the Turbine will <u>NOT</u> trip, <u>THEN</u> close the MSIVs AND MSIV Bypasses.
3.	**Dispatch An Operator To Perform Attachment 6, Restoring AC Power At The DSDG Generator Control Panel.**	

CSF-3, HEAT SINK



The correct answer is B

A) Inorrect. CSFST's are not being monitored for information only is incorrect. After the EDG is powering E1, CSFST's are monitored and the procedure is exited. A red path does not exist on CSF-3. Even though S/G level is <8%, Total FW flow is >300 GPM.

B) Correct. CSFST's are being monitored for information only is correct. Plausible because up until the EDG is started, CSFST's are being monitored for information only. A red path does not exist on CSF-3. Even though S/G level is <8%, Total FW flow is >300 GPM.

C) Incorrect. CSFST's are not being monitored for information only is incorrect. After the EDG is powering E1, CSFST's are monitored and the procedure is exited. A red path does exist on CSF-3 is incorrect. Plausible if the student is unaware of how much total fw flow is required to satisfy the CSFST or if they believe that only S/G level being <8% will give you a red path.

D) Incorrect. CSFST's are being monitored for information only is correct. Plausible because up until the EDG is started, CSFST's are being monitored for information only. A red path does exist on CSF-3 is incorrect. Plausible if the student is unaware of how much total fw flow is required to satisfy the CSFST or if they believe that only S/G level being <8% will give you a red path.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.0 SRO 4.6
K/A: 055 Loss of Offsite and Onsite Power (Station Blackout)
EG 2.4.21: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
Reference(s): Sim/Plant design,
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 2 of FRP-H.1 lesson plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (CFR: 41.7 / 43.5 / 45.12)
Comments:

This question meets the K/A because the student must determine from a set of conditions if there is a red path on CSF-3.

12. 056 AA1.15 001

Given the following plant conditions:

-A Loss of off-site power occurs

-Fifteen seconds later a Large Break LOCA occurs

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

The BOP will expect to see the SWBPs started on the __(1)__ . SWBP suction pressure __(2)__ have to be at least 30 psig for the SWBP's to start.

A. (1) Blackout sequencer
(2) does

B. (1) Blackout sequencer
(2) does **NOT**

C. (1) SI sequencer
(2) does

D✓ (1) SI sequencer
(2) does **NOT**

The correct answer is D.

A) Incorrect. The blackout sequencer did start, however, the SWBP's won't be started until 20 seconds after the blackout signal. Since an SI will occurr due to the LBLOCA, the SI sequencer takes over. The SWBP's will start on the SI Sequencer. Plausible since the Blackout sequencer does start the SWBP's. Suction pressure does not matter for the SI sequencer. Plausible because suction pressure needs to be >30 psig for the SWBP's to start on the blackout sequencer.

B)Incorrect. The blackout sequencer did start, however, the SWBP's won't be started until 20 seconds after the blackout signal. Since an SI will occur due to the LBLOCA, the SI sequencer takes over. The SWBP's will start on the SI Sequencer. Plausible since the Blackout sequencer does start the SWBP's. Suction pressure does not need to be 30 psig to start on the SI sequencer. Plausible if the student doesn't know which sequencer needs to see 30 psig to start the SWBP.

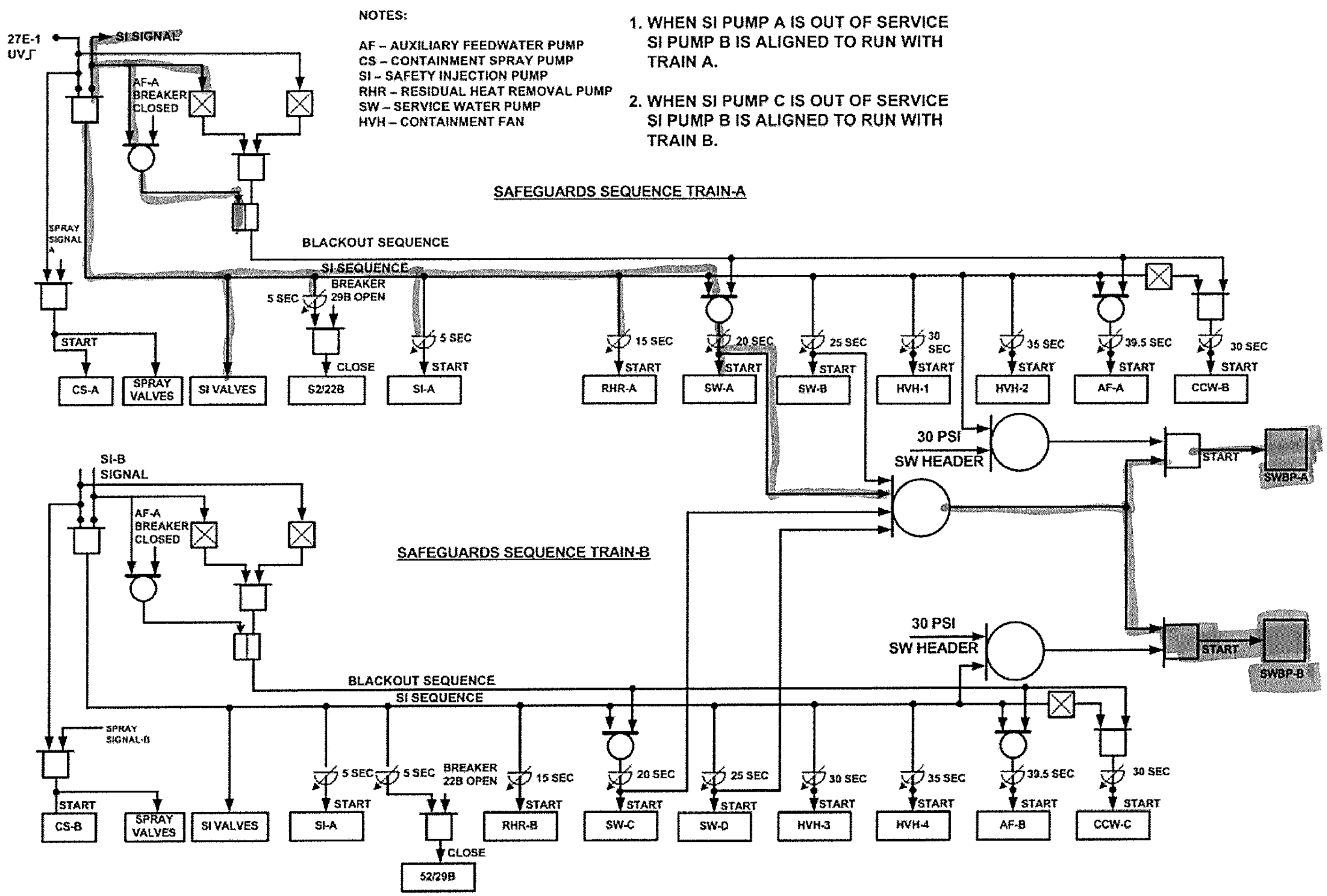
C) Incorrect. The SWBP's do get started on the SI Sequencer. Suction pressure does not need to be 30 psig to start on the SI sequencer. Plausible if the student doesn't know which sequencer needs to see 30 psig to start the SWBP.

D) Correct. The SWBP's do get started on the SI Sequencer. Suction pressure does not matter for the SWBP to start on the SI Sequencer.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.1 SRO 4.3
K/A: 056 Loss of Offsite Power
AA1: Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power:
AA1.15: Service water booster pump
References: Sim/Plant design, Student Text for ESF
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 6 of ESF lesson plan
Question Source: RNP Bank
Question History: None
Question Cognitive Level: High
10 CFR part 55: (CFR 41.7 / 45.5 / 45.6)

Meets the K/A because operators are required to monitor automatic actions of equipment and ensure the correct actions have occurred.

Correct


NOTES:
AF – AUXILIARY FEEDWATER PUMP
CS – CONTAINMENT SPRAY PUMP
SI – SAFETY INJECTION PUMP
RHR – RESIDUAL HEAT REMOVAL PUMP
SW – SERVICE WATER PUMP
HVH – CONTAINMENT FAN
1. WHEN SI PUMP A IS OUT OF SERVICE SI PUMP B IS ALIGNED TO RUN WITH TRAIN A.
2. WHEN SI PUMP C IS OUT OF SERVICE SI PUMP B IS ALIGNED TO RUN WITH TRAIN B.
SAFEGUARDS SEQUENCE TRAIN-A
27E-1
UV
SI SIGNAL
AF-A BREAKER CLOSED
SPRAY SIGNAL A
BLACKOUT SEQUENCE
SI SEQUENCE
BREAKER 29B OPEN
5 SEC
15 SEC
20 SEC
25 SEC
30 SEC
35 SEC
39.5 SEC
START
CLOSE
CS-A
SPRAY VALVES
SI VALVES
52/22B
SI-A
RHR-A
SW-A
SW-B
HVH-1
HVH-2
AF-A
CCW-B
30 PSI SW HEADER
SWBP-A
SWBP-B
SAFEGUARDS SEQUENCE TRAIN-B
SI-B SIGNAL
SPRAY SIGNAL-B
BREAKER 22B OPEN
CS-B
SI-A
52/29B
RHR-B
SW-C
SW-D
HVH-3
HVH-4
AF-B
CCW-C

Incorrect

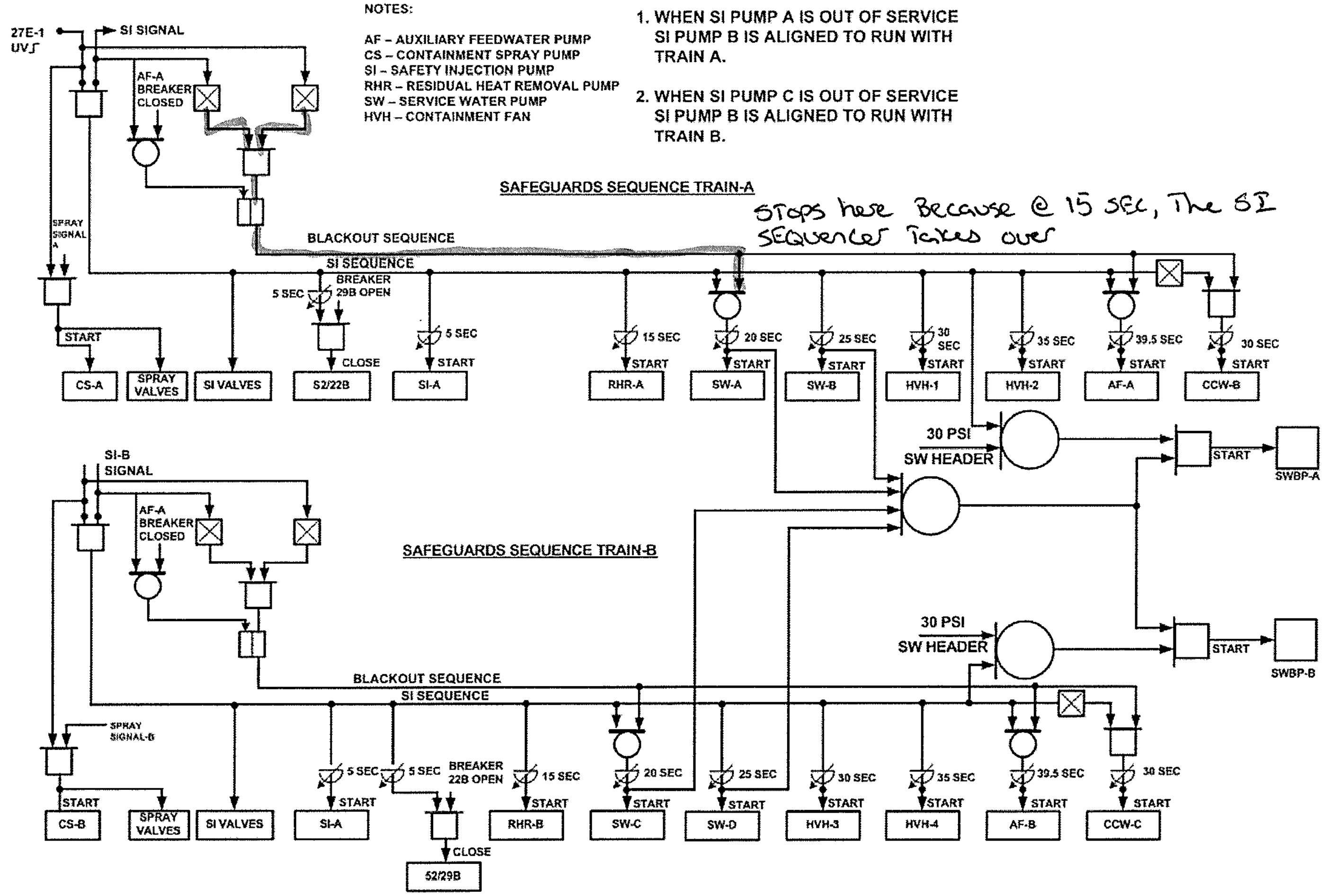
NOTES:

AF – AUXILIARY FEEDWATER PUMP
CS – CONTAINMENT SPRAY PUMP
SI – SAFETY INJECTION PUMP
RHR – RESIDUAL HEAT REMOVAL PUMP
SW – SERVICE WATER PUMP
HVH – CONTAINMENT FAN

1. WHEN SI PUMP A IS OUT OF SERVICE SI PUMP B IS ALIGNED TO RUN WITH TRAIN A.

2. WHEN SI PUMP C IS OUT OF SERVICE SI PUMP B IS ALIGNED TO RUN WITH TRAIN B.

Stops here because @ 15 sec, the SI sequencer takes over



13. 057 AA2.19 001

Given the following plant conditions:

-The reactor is currently at 6% power with the turbine rolling at 1800 rpm IAW GP-005, POWER OPERATION

-Power is lost to Instrument Bus (IB) #1

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

The loss of IB #1 __(1)__ cause a reactor trip. The turbine is __(2)__ .

A.✓ (1) will
(2) tripped

B. (1) will **NOT**
(2) tripped

C. (1) will
(2) **NOT** tripped

D. (1) will **NOT**
(2) **NOT** tripped

The correct answer is A.

A) Correct. The reactor is tripped. IB#1 feeds NIS Cabinet Channel I that will trip IR N35 bistables. 1/2 trip actuated on IR Trip that is still enabled at this time. This will cause direct reactor trip and any reactor trip will trip turbine. The turbine has tripped as a result of the reactor trip.

B) Incorrect. The reactor is tripped. Plausible if the student doesn't realize that they are less than 10% power. The turbine is tripped.

C) Incorrect. The reactor is tripped. IB#1 feeds NIS Cabinet Channel I that will trip IR N35 bistables. 1/2 trip actuated on IR Trip that is still enabled at this time. This will cause direct reactor trip and any reactor trip will trip turbine. The turbine would be tripped. Plausible if the student doesn't recognize that you always get a turbine trip with a reactor trip.

D) Incorrect. The reactor is tripped. Plausible if the student doesn't realize that they are less than 10% power. The turbine would be tripped. Plausible if the student doesn't recognize that you always get a turbine trip with a reactor trip.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 4.0 SRO 4.3
K/A: 057 Loss of Vital AC Electrical Instrument Bus
AA2: Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:
AA2.19: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus
References: Sim/Plant design,
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5/6 of NIS
Question Source: New
Question History:
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 43.5/45.13)

Meets the K/A because the student is required to determine and interpret what automatic actions will occurr due to the loss of IB #1.

Discussion (Continued)

Instrument Power:

On a loss of Instrument Power (Secondary buses), all instrument signals in that Channel will be reduced to a zero state. Thus, for example, Steam Generator I pressure and VCT levels will indicate zero. A zero input signal trips low bistables and inhibits high bistables from providing a trip output to a protection matrix. This will not normally result in, or prevent, protective actuations (Since most matrices are 2/3, a loss of a single channel will change them to 2/2 or 1/2.) Analog Control systems, however, will respond to the signal change.

Control Power:

On a loss of Control Power (Primary bus), all Bistables in that Channel will go to their fail-safe condition. (Exceptions to this are CV Hi-Hi pressure, and the P-6 bistables which are energize to actuate) Thus, the 2/3 matrices will become 1/2, etc. This will not normally result in a protective actuation. However, if for example, a Channel II trip already exists (from some other cause), and Channel I experiences a loss of Control Power, two trips will exist for that protective feature and an actuation will occur. Loss of Primary bus will also result in a loss of Secondary bus (Instrument Power), but the Bistables will trip anyway. However, the indications themselves will fail. This will provide conflicting information to the operator and may cause a plant control response.

2. Safeguard Racks - Control Features:

IB 7 (Train A) and IB 3 (Train B) supply power to the interposing relays for the loads started from their sequencer. A loss of either of these buses will prevent that sequencer from starting its loads. (Note: These loads may still be started manually by the operator after the EDG has loaded the bus)

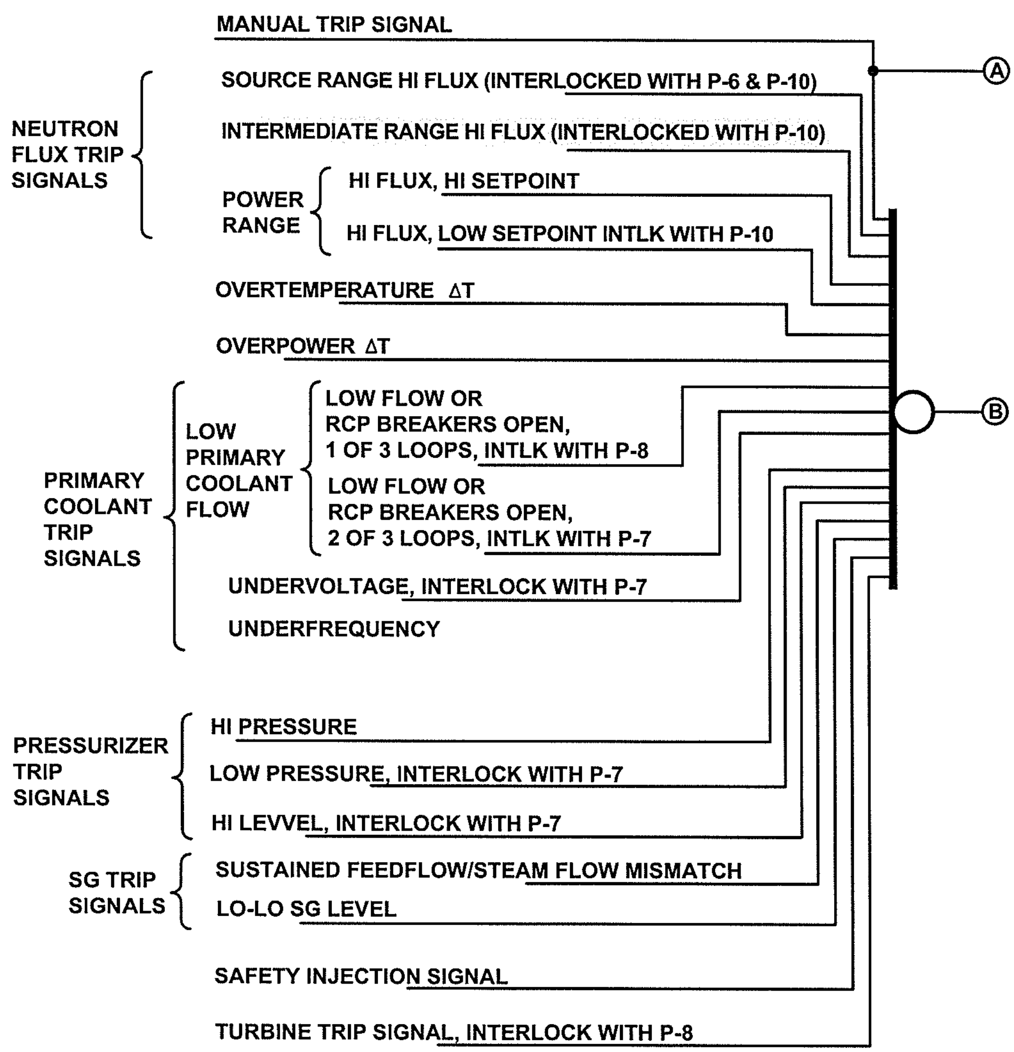
3. Nuclear Instruments:

Bus 1 through 4 supply Instrument and Control power to the Power Range NIS. On a loss of an IB, both sources will be lost to the affected Power Range and all of its components will fail. The Dropped Rod Bistable will fail in the tripped condition and provide a dropped rod alarm.

Bus 1 and 2 supply the Source and Intermediate Ranges. A loss of either of these Buses will cause a SR & IR High Flux Trip. Normally, these 1/2 trips are bypassed, but at low power levels, a Rx Trip will occur.

Reactor Trips

Reactor Trip Signals



14. 058 AK1.01 001

Given the following plant conditions:

-Battery chargers A and B are in service

-Battery chargers A-1 and B-1 are in standby

-12:00 MCC-5 is de-energized

-13:01 APP-036-D3, BATT A/B LO VOLT, alarm is illuminated

-There are multiple indications of loss of control power to components

Subsequently:

-13:05:00 MCC-5 is re-energized

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

Battery charger(s) __(1)__ will need to be manually restarted. As of 13:05:00 the following day, the batteries __(2)__ be fully charged.

A.✓ (1) A-1 only
(2) will

B. (1) A & A-1
(2) will

C. (1) A & A-1
(2) will **NOT**

D. (1) A-1 only
(2) will **NOT**

The correct answer is A.

A) Correct. The battery chargers are sized to charge its partially discharged battery within 24 hours while carrying its normal loads. MCC-5 powers the A train battery chargers. A, the in service battery charger will automatically restart when power is returned. The standby charger, A-1, must be manually restarted.

B) Incorrect. MCC-5 powers the A train battery chargers. A, the in service battery charger will automatically restart when power is returned. The standby charger, A-1, must be manually restarted. Plausible if the students are unsure of which battery chargers automatically restart and which ones need to be manually restarted. The battery chargers are designed to recharge a partially discharged battery within 24 hours while carrying its normal loads.

C) Incorrect. MCC-5 powers the A train battery chargers. A, the in service battery charger will automatically restart when power is returned. The standby charger, A-1, must be manually restarted. Plausible if the students are unsure of which battery chargers automatically restart and which ones need to be manually restarted. The battery chargers are designed to recharge a partially discharged battery within 24 hours while carrying its normal loads. Plausible if the students don't know the design features of the battery chargers.

D) Incorrect. MCC-5 powers the A train battery chargers. A, the in service battery charger will automatically restart when power is returned. The standby charger, A-1, must be manually restarted. The battery chargers are designed to recharge a partially discharged battery within 24 hours while carrying its normal loads. Plausible if the students don't know the design features of the battery chargers.

Question: 14
Tier/Group: 1/1
K/A Importance Rating: RO 2.8 SRO 3.1
K/A: 058 Loss of DC Power
AK1: Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power:
AK1.01: Battery charger equipment and instrumentation
References: Sim/Plant design, DC Student Text, OP-601
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3/10 of DC System lesson plan
Question Source: New
Question History:
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.8 / 41.10 / 45.3)

Meets the K/A because the operational implication is that the battery can supply its normal loads while recharging a partially discharged battery.

CHAPTER 1

Introduction

The DC power system consists of five 125V batteries, each with its own battery charger(s) and DC bus. The five batteries are A, B, C, DS System and DSDG. Two of the batteries, A & B, are safety-related. The battery chargers supply the normal DC loads as well as maintaining each battery fully charged. Each charger has the capacity to supply all normal DC loads and maintain the battery fully charged. For each safety-related station battery, there are two safety-related battery chargers. One battery charger supplies the normal DC loads while the other provides 100% back-up capability.

Each of the two safety-related station batteries is sized to carry its expected shutdown loads following a design basis accident with no battery chargers available for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. Each of the four safety-related chargers has been sized to charge its partially discharged battery within 24 hours while carrying its normal load.

The DC subsystems associated with the Dedicated Shutdown system and DSDG are described in ST-056, Dedicated Shutdown Diesel Generator.

3.0 RESPONSIBILITIES

3.1 Operations personnel are responsible for component manipulations within this procedure.

4.0 PREREQUISITES

4.1 The Electrical Distribution System is in service in accordance with OP-603 to supply power to Motor Control Centers MCC-2 MCC-5, MCC-6, MCC-9, MCC-10, and MCC-12 to support the Battery Chargers, the Battery Room fans, and Battery Room heating and cooling.

5.0 PRECAUTIONS AND LIMITATIONS

5.1 One battery charger for each battery shall be in service so that the batteries will always be at full charge in anticipation of a Loss-of-AC incident.

5.2 The DC distribution system for station battery “A” and “B” shall be maintained operable.

5.3 The Battery Room Ventilation Fans “A” and/or “B” shall be in service.

5.4 Caution should be exercised when working near the Batteries. Rubber gloves, apron and face shields should be worn when handling acid.

5.5 NO SMOKING, BURNING OR WELDING is permitted in the battery rooms.

5.6 Each time a circuit is energized, the applicable Battery Charger (“A” or “A-1”, “B” or “B-1”, or “C”) should be checked for ground indication. OMM-035, Ground Isolation contains a description of Battery Charger ground detector operations.

5.7 All normal precautions pertaining to energized electrical gear shall be observed.

5.8 Inverter input voltage alarm set points are 100 VDC and 140 VDC for Inverters “A” and “B” **AND** 105 VDC and 140 VDC for Inverter “C”.

5.9 In the event that it becomes necessary to disconnect a Station Battery from its DC Bus, Chargers A, A-1, B, and B-1 have the capability to stand alone carrying its associated DC Bus.

5.10 On a loss of A.C. supply, battery chargers in Standby will trip and will require manual restart when A.C. supply is restored.

15. 077 AK3.02 001

Given the following plant conditions:

-The reactor is at 25% RTP

-APP-009-F8, 4KV BUSES LO FREQ, is in alarm

-The crew enters AOP-026, GRID INSTABILITY

-Frequency is now at 58.3Hz

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

The crew will __(1)__ trip the reactor. IAW AOP-026 basis document, the purpose of tripping the RCP's is to __(2)__ .

A. (1) immediately
(2) prevent the automatic reactor trip from low flow

B. (1) wait 5 minutes, and then
(2) prevent the automatic reactor trip from low flow

C✓ (1) immediately
(2) prevent the automatic RCP trip at 58.2 Hz

D. (1) wait 5 minutes, and then
(2) prevent the automatic RCP trip at 58.2 Hz

The correct answer is C.

A) Incorrect. The crew will immediately trip the reactor. Prevent the automatic reactor trip from low flow is incorrect. Plausible since the reactor will trip from low flow at this power if two or more RCP's are tripped.

B) Incorrect. The crew will immediately trip the reactor. Plausible since the crew would wait 5 minutes if the frequency was between 59.0-58.4 Hz. Prevent the automatic reactor trip from low flow is incorrect. Plausible since the reactor will trip from low flow at this power if two or more RCP's are tripped.

C) Correct. The crew will immediately trip the reactor. The purpose of tripping the RCP's in AOP-026 is to prevent the automatic trip at 58.2 Hz.

D) Incorrect. The crew will immediately trip the reactor. Plausible since the crew would wait 5 minutes if the frequency was between 59.0-58.4 Hz. The purpose of tripping the RCP's in AOP-026 is to prevent the automatic trip at 58.2 Hz.

Question:
Tier/Group: 1/1
K/A Importance Rating: RO 3.6 SRO 3.9
K/A: 077 Generator Voltage and Electric Grid Disturbances
AK3: Knowledge of the reasons for the following responses as they apply to Generator Voltage and Electric Grid Disturbances:
AK3.02: Actions contained in abnormal operating procedure for voltage and grid disturbances
References: Sim/Plant design, AOP-026 and its basis document
Proposed references to be provided to applicants during the Exam: None
Question Source: New
Question History:
Learning Obj: Objective 2
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.4, 41.5, 41.7, 41.10 / 45.8)

Meets the K/A because the questions asks for the reason the crew trips the RCP's due to electrical grid disturbances in AOP-026.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

* 1. **DETERMINE If Reactor Trip Is Required As Follows:**

a. CHECK Plant Status - MODE 1 OR 2

a. GO TO Step 3.

b. CHECK Any Of The Following:

correct

- System Frequency - LESS THAN 58.4 HZ

  OR

- Single-Phase Open Circuit Condition indicated by any of the following:
  - Multiple 4KV or 480V components with OVLD/TRIP annunciators in alarm

    OR

  - Current and voltage phase-to-phase imbalance on 4KV or 480V components

    OR

  - Visual observation of damaged equipment

    OR

  - Notification from Load Dispatcher

incorrect

b. IF any of the following indications are observed, THEN PERFORM Step 2.

- Frequency lowers to less than 58.4 Hz

  OR

- Frequency remains between 58.4 Hz and 59.0 Hz for greater than 5 minutes

  incorrect

  OR

- Single-Phase Open Circuit Condition indicated by any of the following:
  - Multiple 4KV or 480V components with OVLD/TRIP annunciators in alarm

    OR

  - Current and voltage phase-to-phase imbalance on 4KV or 480V components

    OR

  - Visual observation of damaged equipment

    OR

  - Notification from Load Dispatcher

GO TO Step 7.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**PERFORM The Following:** a. TRIP the Reactor b. TRIP the RCPs c. GO TO EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure	

NOTE

With Unit #2 separated from the grid, system frequency can be obtained from the Load Dispatcher <u>OR</u> Unit #1 when it is connected to the grid.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 3.	**DETERMINE If Reactor Coolant Pump Trip In Modes 3, 4, <u>OR</u> 5 Is Required As Follows:**	
	a. CHECK system frequency - LESS THAN 58.4 HZ	a. <u>IF</u> frequency lowers to less than 58.4 Hz, <u>THEN</u> PERFORM Steps 3.b and 3.c. <u>IF</u> all running RCPs trip on low frequency, <u>THEN</u> PERFORM Step 3.c. GO TO Step 4.
	b. STOP Running RCPs.	
	c. VERIFY one of the following methods of core cooling in progress: • RHR aligned for core cooling <u>OR</u> • Natural circulation using Supplement E, Natural Circulation Verification.	

Correct

BASIS DOCUMENT, GRID INSTABILITY

INDIVIDUAL STEP DESCRIPTION:

Step	Description
1	This continuous action step has the operator determine if Reactor Trip is required. If the plant is in Modes 1 or 2 AND frequency is less than 58.4 Hz or a single-phase open circuit condition is indicated by multiple overload/trip annunciators on 4KV or 480V components, current and voltage phase-to-phase imbalance on 4KV or 480V components, visual observation of damaged equipment, or notification from the Load Dispatcher, then the operator is directed to step two which will trip the reactor. If the plant is in not in Modes 1 or 2, a transition step proceeds to Step 3. If the plant is in Modes 1 or 2 AND frequency is greater than 58.4 Hz, operation of the Turbine is allowed for only 5 minutes with frequency between 58.4 and 59.0 Hz. The RNO directs the operator to Step 2 if frequency remains in this range for 5 minutes or frequency lowers below 58.4 Hz at any time during the performance of this procedure. If the plant is in Modes 1 or 2 and a reactor trip is not required, a transition is provided to Step 7 while continuing to monitor for indications that would require a reactor trip.
2	This step provides detail as to what is required to shut down the plant in order to protect major equipment from the affects of abnormal frequency or voltage. The step will have the operator trip the Reactor; stop the RCPs and transition to EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure. The RCPs are tripped below 58.4 Hz in an attempt to preempt the automatic trip at 58.2 Hz.
3N	If the unit is in modes 3, 4, or 5 and the generator is separated from the grid, the frequency meter on the RTGB does not sense grid frequency since the sensing point is upstream of the unit OCB's. To determine grid frequency, it is necessary to contact the ECC (Load Dispatcher) or Unit #1 if it is connected to the grid.
3	This continuous action step has the operator determine if reactor coolant pump trip is required. If the plant is in Modes 3, 4, or 5 and frequency is less than 58.4 Hz, the steps have the operator stop the RCPs and verify core cooling is available with either RHR or natural circulation using Supplement E. The RNO also directs performance of the subsequent stopping of the RCPs if frequency lowers below 58.4 Hz at any time during performance of this procedure. The RCPs are tripped below 58.4 Hz in an attempt to preempt the automatic trip at 58.2 Hz. If all the RCPs trip on low frequency, the RNO will also direct the operator verify core cooling is available with either RHR or natural circulation using Supplement E. A transition is provided to Step 4 while continuing to monitor frequency.

Incorrect

Purpose and Entry Conditions

(Page 2 of 4)

2. **SYMPTOMS AND ENTRY CONDITIONS**

a. The following are symptoms that require a reactor trip, if one has not occurred:

REACTOR TRIP SIGNAL	LOGIC (INTERLOCK)	SETPOINT
SR High Flux	1/2 (P-6 and P-10)	10^5 CPS
IR High Flux	1/2 (P-10)	Current equal to 25%
PR High Flux Low Range	2/4 (P-10)	24%
PR High Flux High Range	2/4	108%
PZR High Pressure	2/3	2376 psig
PZR Low Pressure	2/3 (P-7)	1844 psig
PZR High Level	2/3 (P-7)	91%
Low RCS Flow	2/3 on 2/3 (P-7) 1/3 (P-8)	94.68% rated flow
RCP Breaker	1/1 on 2/3 (P-7) 1/3 (P-8)	Open
RCP Bus Undervoltage	2/3 busses (P-7)	75% (3120 volts)
Over Temperature ΔT	2/3	Variable
Over Power ΔT	2/3	Variable
Safety Injection	Auto or Manual	N/A
Turbine Trip	2/2 SV or 2/3 AST (P-8)	N/A
Low-Low S/G Level	2/3 on 1/3	16%
Low S/G Level With Steam Flow > Feed Flow	1/2 on 1/3 1/2 on 1/3	30% 0.64×10^6 lbm/hr

(CONTINUED NEXT PAGE)

16. WE/04 EK2.1 001

Given the following plant conditions:

-The plant is in Mode 3

-The crew has entered EPP-20, LOCA OUTSIDE CONTAINMENT

SI-870 A & B, BIT INJ Valves
RHR-744 A & B, RHR COLD LEG INJ Valves

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

In order to isolate a leak in the Cold Leg Injection piping, EPP-20 will have the crew shut __(1)__ and monitior __(2)__ to see if the leak is isolated.

A. (1) SI-870 A & B
(2) RCS pressure

B. (1) SI-870 A & B
(2) PZR level

C. (1) RHR-744 A & B
(2) PZR level

D✓ (1) RHR-744 A & B
(2) RCS pressure

The correct answer is D.

A) Incorrect. Shutting SI-870 A&B is incorrect. Plausible since by shutting these two valves, it would isolate the SI injection into the cold leg. The RHR system is at a lower pressure and therefore would be more susceptible to a leak. RCS pressure is the parameter EPP-20 has you look at to see if the leak is isolated after shutting the appropriate valves.

B) Incorrect. Shutting SI-870 A&B is incorrect. Plausible since by shutting these two valves, it would isolate the SI injection into the cold leg. The RHR system is at a lower pressure and therefore would be more susceptible to a leak. RCS pressure is the parameter EPP-20 has you look at to see if the leak is isolated after shutting the appropriate valves. Plausible because PZR level would rise if the leak was isolated.

C) Incorrect. The crew will shut RHR-744 A&B. The RHR system is at a lower pressure and therefore would be more susceptible to a leak. RCS pressure is the parameter EPP-20 has you look at to see if the leak is isolated after shutting the appropriate valves. Plausible because PZR level would rise if the leak was isolated.

D) Correct. The crew will shut RHR-744 A&B. The RHR system is at a lower pressure and therefore would be more susceptible to a leak. RCS pressure is the parameter EPP-20 has you look at to see if the leak is isolated after shutting the appropriate valves.

Question: 16
Tier/Group: 1/1
K/A Importance Rating: RO 3.5 SRO 3.9
K/A: WE/04 LOCA Outside Containment
EK2: Knowledge of the interrelations between the (LOCA Outside Containment) and the following:
EK2.1: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
References: Sim/Plant design, EPP-20
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of EPP-20 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.7 / 45.7)

Meets the K/A because the questions asks the interrelations between a small break LOCA and the valves that are shut to isolate the loca.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Determine If Break Is In Cold Leg Injection Piping As Follows:**	
	a. Reset SAFETY INJECTION	
	b. Close RHR COLD LEG INJ Valves while monitoring for an RCS pressure increase: • RHR-744A • RHR-744B	
	c. Check RCS pressure - INCREASING	c. Open RHR COLD LEG INJ Valves: • RHR-744A • RHR-744B Go To Step 5.
	d. Maintain RHR COLD LEG INJ Valves - CLOSED • RHR-744A • RHR-744B	
	e. Notify Plant Operations Staff that the break has been isolated and to determine further recovery actions	
	f. Go To Step 8	

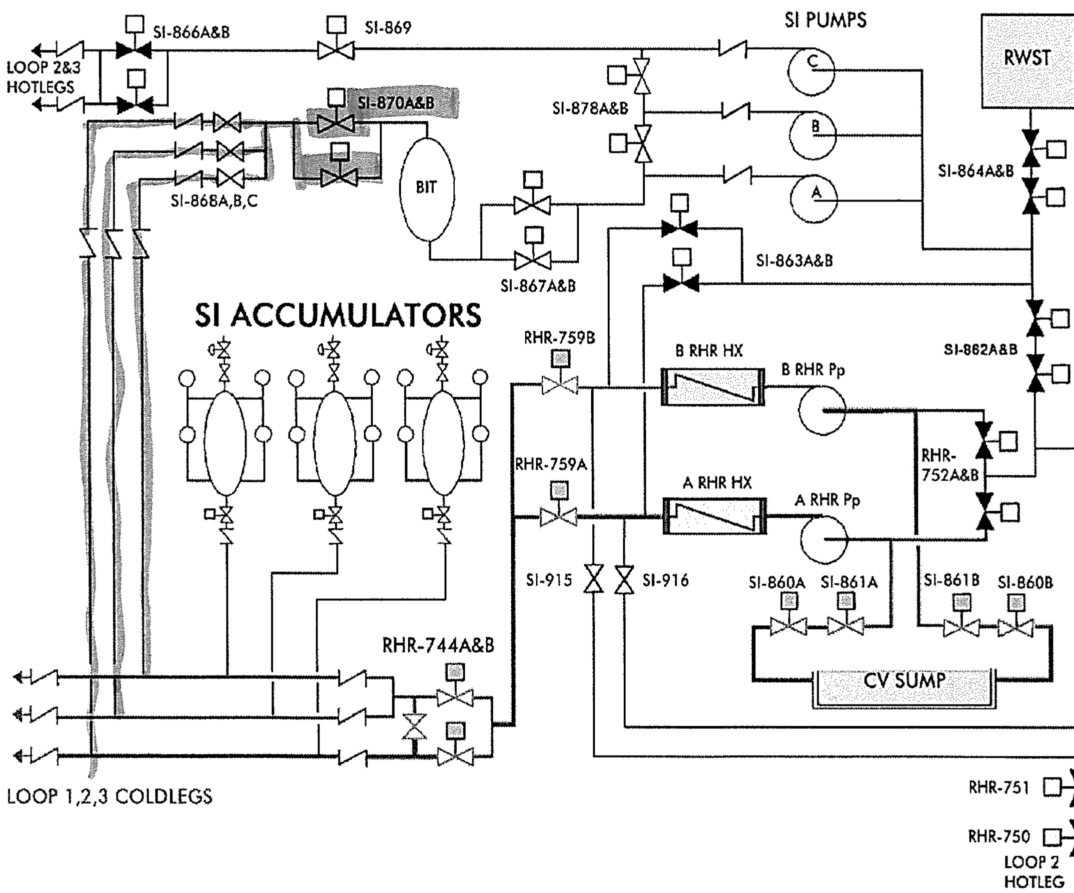
incorrect

Redundancy and segregation of instrumentation and components are incorporated in the design to assure that postulated malfunctions will not impair the ability of the system to meet the design objectives. The system is effective in the event of loss of normal plant auxiliary power coincident with the loss of coolant, and can accommodate the credible failure of any single component or instrument channel to respond actively in the system. During the recirculation phase of a LOCA, the system can accommodate a loss of any part of the flow path, since backup alternate flow path capability is provided.

Pipe whip protection for the ECCS components is provided as is protection against seismic events, protection against missiles, and loads that may result from the effects of a LOCA.

The accumulators, which are passive components, discharge into the cold legs of the reactor coolant piping when RCS pressure lowers to 630-640 psig, thus assuring rapid core cooling for large pipe breaks. They are located inside the containment, but outside the crane wall. Therefore each accumulator is protected against possible missiles.

Figure 1 SI System Shown in Recirculation mode (< 275 psig, EPP 9)



17. WE/05 G2.4.6 001

Which of the following is the **FIRST** mitigation strategy that FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, directs you to attempt?

A. Maximize charging flow to cool the RCS

B. Depressurize the S/G's to raise subcooling

C. Initiation of RCS Bleed and Feed Heat Removal

D✓ Attempt Restoration of Feed Flow To Steam Generators

The correct answer is D.

A) Incorrect. Plausible because this is part of what you do when you setup feed and bleed.

B) Incorrect. Plausible because you do depressurize S/G's, however, it is not to raise subcooling.

C) Incorrect. Plausible because if you can't restore feed flow, FRP-H.1 has you perform feed and bleed. This is the Second strategy that FRP-H.1 has you attempt.

D) Correct. FRP-H.1 has you first attempt to restore feed flow to S/G's.

Question: 17
Tier/Group: 1/1
K/A Importance Rating: RO 4.1 SRO 4.3
K/A: WE/05 Loss of Secondary Heat Sink
G2.4.6: Knowledge of EOP mitigation strategies.
References: Sim/Plant design, FRP-H.1 Basis Document
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3 of FRP-H.1 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 43.5 / 45.13)

Meets the K/A because it asks the students what the the first mitigation strategy is of our Loss of Secondary Heat Sink procedure.

Correct

The focus of the analysis was to determine the additional time available to the operator as a result of eliminating RCP heat from the system before action to initiate bleed and feed became necessary. Thus, the time of two events was used to determine the impact of RCP trip time. The two events are 1) the time when PORVs automatically open as a result of the degraded heat transfer capability of the steam generator and 2) the time when steam generator secondaries dry out.

Table 1 shows a comparison of the three cases. Case 1 represents a situation where steam generators would experience the earliest dryout due to the RCP heat load and Case 2 is where the steam generators would experience the latest dryout. The extension in dryout time from Case 1 to Case 2 is between 7 and 9 minutes, depending upon the indication of dryout that is chosen. The use of the time to PORV opening will have some uncertainty due to the uncertainty in predicting non-equilibrium effects in the pressurizer. However, PORV opening time is probably the best indicator obtainable from the LOFTRAN analysis of the time available until bleed and feed must be initiated.

TABLE 1

IMPACT OF RCP TRIP ON LOSS OF HEAT SINK

PARAMETER	CASE 1*	CASE 2*	CASE 3*
PORVs OPEN	30.75 min	37.83 min	35.80 min
STEAM GENERATOR DRY OUT	33.10 min	42.50 min	40.93 min

* CASE 1: All RCPs Running

CASE 2: All RCPs Tripped at Reactor Trip

CASE 3: All RCPs Tripped 5 Minutes After Reactor Trip

Reactor trip occurred at 28 seconds.

Loss of main feed occurred at 10 seconds.

Case 3, where the RCPs are tripped 5 minutes after reactor trip, is a best estimate expectation of when the operator can be expected to trip RCPs following a reactor trip based on guidance provided in this guideline. Thus, the extension in time to loss of heat sink symptoms is the most realistic that could be expected based on anticipated operator response. The extension to loss of secondary heat sink symptoms is about 5 minutes based on PORV opening time. This compares favorably with the extension already seen between Cases 1 and 2. Thus, operator action to trip RCPs upon entering this guideline for loss of secondary heat sink can appreciably delay the need for bleed and feed and the loss of secondary heat sink. Thus, time can be gained for the operator to establish a means of supplying feedwater.

Delaying the loss of secondary heat sink is not the only reason for tripping RCPs. RCPs running can also reduce the effectiveness of bleed and feed. RCP heat input to the RCS will result in increased steam generation hindering the depressurization of the RCS during bleed and feed. The higher pressure produced by RCP operation will reduce SI flow and increase inventory lost through the PORVs. Therefore, RCPs should be tripped if AFW flow cannot be established immediately after entering this guideline.

3. RECOVERY/RESTORATION TECHNIQUE

The objective of the recovery/restoration technique incorporated into guideline FR-H.1 is to restore and/or maintain adequate secondary heat removal capability and to establish RCS bleed and feed heat removal if secondary heat removal capability cannot be maintained.

The following subsections provide a summary of the major categories of operator actions and key utility decision points for guideline FR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

3.1 High Level Action Summary

A high level summary of the actions performed in FR-H.1 is given below in the form of major action categories. These are described below in more detail.

MAJOR ACTION CATEGORIES IN FR-H.1

- o Attempt Restoration of Feed Flow To Steam Generators
- o Initiation of RCS Bleed and Feed Heat Removal
- o Restore and Verify Secondary Heat Sink

Incorrect

The focus of the analysis was to determine the additional time available to the operator as a result of eliminating RCP heat from the system before action to initiate bleed and feed became necessary. Thus, the time of two events was used to determine the impact of RCP trip time. The two events are 1) the time when PORVs automatically open as a result of the degraded heat transfer capability of the steam generator and 2) the time when steam generator secondaries dry out.

Table 1 shows a comparison of the three cases. Case 1 represents a situation where steam generators would experience the earliest dryout due to the RCP heat load and Case 2 is where the steam generators would experience the latest dryout. The extension in dryout time from Case 1 to Case 2 is between 7 and 9 minutes, depending upon the indication of dryout that is chosen. The use of the time to PORV opening will have some uncertainty due to the uncertainty in predicting non-equilibrium effects in the pressurizer. However, PORV opening time is probably the best indicator obtainable from the LOFTRAN analysis of the time available until bleed and feed must be initiated.

TABLE 1

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Delaying the loss of secondary heat sink is not the only reason for tripping RCPs. RCPs running can also reduce the effectiveness of bleed and feed. RCP heat input to the RCS will result in increased steam generation hindering the depressurization of the RCS during bleed and feed. The higher pressure produced by RCP operation will reduce SI flow and increase inventory lost through the PORVs. Therefore, RCPs should be tripped if AFW flow cannot be established immediately after entering this guideline.

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A high level summary of the actions performed in FR-H.1 is given below in the form of major action categories. These are described below in more detail.

MAJOR ACTION CATEGORIES IN FR-H.1

- Attempt Restoration of Feed Flow To Steam Generators
- Initiation of RCS Bleed and Feed Heat Removal
- Restore and Verify Secondary Heat Sink

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
47.	**Check Charging Flow As Follows:**	
	a. Check Charging Pumps - ALL STOPPED	a. Go To Step 47.c.
	b. Check the following alarms - EXTINGUISHED: • APP-001-C1, RCP THERM BAR COOL WTR HI FLOW AND • APP-001-D1, RCP THERM BAR COOL WTR LO FLOW	b. Perform the following: 1) Verify all RCPs STOPPED. 2) Locally close the following RCP SEAL WATER FLOW CONTROL VALVEs prior to continuing: • CVC-297A • CVC-297B • CVC-297C
	c. Align Charging Pump suction to RWST as follows:	
	1) At the RTGB, verify LCV-115B, EMERG MU TO CHG SUCT - OPEN	1) Open CVC-358, RWST TO CHARGING PUMP SUCTION.
	2) Verify LCV-115C, VCT OUTLET - CLOSED	
	d. Start all available Charging Pumps	
	e. Increase running Charging Pumps speed to maximum	
	f. Verify maximum charging flow on FI-122A	
*48.	**Check RWST Level - LESS THAN 27%**	IF RWST level lowers to below 27%, THEN Go To EPP-9, TRANSFER TO COLD LEG RECIRCULATION. Go To Step 50.
49.	**Go To EPP-9, TRANSFER TO COLD LEG RECIRCULATION**	

incorrect

RNP STEP | WOG STEP | BASIS/DIFFERENCES

RNP DIFFERENCES/REASONS

The previous step placed instrument air in the override position. Once Phase A has been reset this can be restored to normal. The step checks for switch position since the previous step would not have been performed if air pressure could not be established.

SSD DETERMINATION

This is an SSD per criterion 10.

N46 N/A WOG BASIS

N/A, this step is not in the WOG ERG.

RNP DIFFERENCES/REASONS

This note has been included to remind the operator of the location of the load listing for all components to be started on an EDG.

SSD DETERMINATION

This is an SSD per criterion 10.

46 18 WOG BASIS

PURPOSE: To verify adequate power to run charging pumps

BASIS:

Charging pump operation will provide charging flow to cool the RCS in addition to SI flow. In order to start the charging pumps, adequate power must be available. If offsite power is available, the electrical system has enough power to supply all safeguards loads in addition to the charging pumps. If offsite power is not available, then the diesel generators may only be sized to supply the safeguards equipment and not the charging pumps in addition to the safeguards equipment. It may then be necessary to secure non-essential loads to establish capacity to run the charging pumps. For the plants that have adequate diesel capacity to run both the safeguards equipment and the charging pumps, this step is not necessary.

RNP DIFFERENCES/REASONS

The RNP step provides additional detail.

SSD DETERMINATION

This is an SSD per criterion 10.

47 19 WOG BASIS

PURPOSE: To start charging pumps, if not running, and establish maximum charging flow

BASIS:

Maximum charging flow to the RCS is established in order to provide additional cooling for the RCS. This additional cooling is especially important if the RCS pressure is above the high-head SI pump shutoff head since the charging pumps would be the only feed path capable of injecting at these pressures. If charging pumps are already running, then seal cooling is adequate and maximum charging flow can be established. If pumps are not running, then seal injection flow is lost, and the only remaining source of seal cooling is CCW. If CCW flow to the RCP thermal barrier is also lost, then the seal is assumed to be already heated up. Rather than initiate the slow and tedious process of reestablishing seal cooling at this time, the seal injection flow path is isolated to allow the charging pumps to be started and maximum charging flow to be established.

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

The SI Accumulator Discharge Valves must be closed when S/G pressure is less than 240 psig to prevent nitrogen injection into the RCS.

****

24. **Depressurize Selected S/G With The Lowest Level To Less Than 600 PSIG By Dumping Steam At Maximum Rate Using One Of The Following Methods Listed In Order Of Preference:**

    - Steam Dump to condenser via the pressure control mode

      OR

    - Steam Line PORVs controlled by Instrument Air

      OR

    - Steam Line PORVs controlled by Nitrogen per Attachment 2 of AOP-017, Loss of Instrument Air

INCORRECT

RNP STEP	WOG STEP	BASIS/DIFFERENCES
20-27	6	WOG BASIS

PURPOSE: To direct the operator in establishing condensate flow as an alternative (or supplement) to AFW and main FW flow

BASIS:

The condensate system is the next source of water readily available to the operator for use in reestablishing the secondary heat sink.

In order to depressurize at least one SG to less than the shutoff head pressure of the condensate system pumps, the RCS must be depressurized below (A.06) psig to allow blocking of the low steamline pressure SI and low PRZR pressure SI signals. If these signals were allowed to actuate, feedline and steamline isolation actuation signals may have to be reset. Feedline isolation may still occur on a reactor trip signal coincident with the low Tavg signal.

Auxiliary spray is used to depressurize the RCS, if letdown is in service, since it provides a maximum cooling to the primary system while allowing no loss of primary water inventory. Normal spray is not available since RCPs are stopped (Step 3). If letdown is not in service, PRZR PORVs are used to avoid thermal stresses to the auxiliary spray nozzles. However, if the PRZR PORVs cannot be used, auxiliary spray must be used.

Depressurization of the SG(s) is accomplished through the condenser steam dump, PORV, or other means if required. Footnote (O.09) defines the steam generator pressure requirement that will allow the condensate pump to provide adequate feedwater flow for decay heat removal. Minimum condensate flow for condensate pump protection, which is provided by a recirculation line flow control valve, is typically much greater than the flow required to remove decay heat. Reducing SG pressure to the condensate pump discharge header pressure for recirculation would permit the condensate pumps to inject into the SG with adequate feed flow for decay heat removal.

The optimum number of SGs to depressurize to less than O.09 psig, in the case of the reference plant, is one because certain benefits are realized by depressurizing only one SG. The likelihood of reaching the criteria for initiation of RCS bleed-and-feed is reduced because only a single SG is steamed. Additionally, the accompanying reduction in pressurizer level and RCS subcooling is less severe, which in turn reduces the likelihood that manual SI actuation will be required based on degraded plant conditions. Thus, before the SG is depressurized it should be isolated from the other SGs.

KNOWLEDGE:

At least one SG should be depressurized to a pressure that allows the condensate pump to deliver flow at least equal to that of which is used for decay heat removal. Providing condensate pump flow equal to the minimum flow used for recirculation, satisfies the flow requirement.

18. WE/11 EA2.2 001

Given the following plant conditions:

-The crew has implemented EPP-15, Loss of Emergency Coolant Recirculation

-The crew is initiating an RCS cooldown to cold shutdown

-The following table is a plot of the RCS cooldown:

TIME	RCS T_{COLD}	TIME	RCS T_{COLD}
0800	547°F	0945	425°F
0815	530°F	1000	395°F
0830	520°F	1015	382°F
0845	505°F	1030	364°F
0900	498°F	1045	340°F
0915	478°F	1100	320°F
0930	447°F	1115	300°F

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

The cooldown rate limit ___(1)___ been exceeded. At ___(2)___ RWST level, the crew is required to secure all running ECCS pumps taking a suction from the RWST.

A. (1) has
(2) 27%

B.✓ (1) has
(2) 9%

C. (1) has **NOT**
(2) 27%

D. (1) has **NOT**
(2) 9%

The correct answer is B.

A) Incorrect. EPP-15 has you cooldown at a rate <100°F. This would mean they have exceeded the cooldown rate at 1000. RWST level of 27% is incorrect. Plausible because at 27%, you normally transition to EPP-9, TRANSFER TO LONG TERM RECIRCULATION. In EPP-9, you secure certain ECCS pumps and realign the systems for long term recirc.

B) Correct. EPP-15 has you cooldown at a rate <100°F. This would mean they have exceeded the cooldown rate at 1000. This is the level in which the RWST is considered empty.

C) Incorrect. EPP-15 has you cooldown at a rate <100°F. This would mean they have exceeded the cooldown rate at 1000. Plausible if they miscalculated the cooldown. RWST level of 27% is incorrect. Plausible because at 27%, you normally transition to EPP-9, TRANSFER TO LONG TERM RECIRCULATION. In EPP-9, you secure certain ECCS pumps and realign the systems for long term recirc.

D) Incorrect. EPP-15 has you cooldown at a rate <100°F. This would mean they have exceeded the cooldown rate at 1000. Plausible if they miscalculated the cooldown. T.S 3.4.3, RCS Pressure and Temperature (P/T) Limits has you restore parameters to within limits in 30 Minutes.

Question: 18
Tier/Group: 1/1
K/A Importance Rating: RO 3.4 SRO 4.2
K/A: WE/11 Loss of Emergency Coolant Recirculation
EA2: Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation)
EA2.2: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
References: Sim/Plant design, EPP-15
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of EPP-15 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR part 55: (CFR 43.5 / 45.13)

Meets the K/A because from the information given, the student has to determine/interpret whether or not the crew has violated the cooldown rate IAW EPP-15. The student must also adhere to EPP-15 requirements of when to secure the ECCS pumps.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
8.	**Verify The Following CV RECIRC FANs - RUNNING** • HVH-1 • HVH-2 • HVH-3 • HVH-4	
* 9.	**Check RWST Level - LESS THAN 9%**	IF RWST level drops to less than 9%, THEN Go To Step 51. Go To Step 11.
10.	**Go To Step 51.**	
11.	**Place The CONTAINMENT SPRAY Key Switch To The OVRD/RESET Position**	

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*48.	**Check If RCPs Must Be Stopped:**	
	a. Check the following: • APP-001-E2, RCP #1 SEAL LEAKOFF LO FLOW - ILLUMINATED OR • APP-001-F2, RCP SEAL WTR LO ΔP - ILLUMINATED	a. IF APP-001-E2 APP-001-F2 Illuminate, THEN perform Step 48.b. Go To Step 49
	b. Stop affected RCPs	
49.	**Check RCS Temperature - GREATER THAN 200°F**	Go To Step 65.
50.	**Check RWST Level - LESS THAN 9%**	Go To Step 1.
51.	**Stop All Of The Following Pumps:** • RHR Pumps • SI Pumps • CV Spray Pumps • Charging Pumps	

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- A differential pressure of 210 psid across the RCP number 1 seals is necessary for continued RCP operation.
- RCS cooldown should be completed as quickly as possible since the RCS may continue to depressurize to a value that may not support differential pressure across the RCP number 1 seals.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
19.	**Initiate RCS Cooldown To Cold Shutdown As Follows:**	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F IN THE LAST 60 MINUTE	
	b. Maintain RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
	c. Check RHR System - ALIGNED FOR CORE COOLING	c. Go To Step 19.f.
	d. Cooldown using RHR System	
	e. Go To Step 25	
	f. Check intact S/Gs - AT LEAST ONE AVAILABLE FOR RCS COOLDOWN	f. <u>IF</u> RHR System unavailable, <u>THEN</u> use a faulted S/G for RCS cooldown.
	g. Check steam dump to Condenser - AVAILABLE	g. Dump steam from S/Gs using STEAM LINE PORVs. Go To Step 20.
	h. Dump steam to Condenser from S/Gs	
20.	**Check RCS Hot Leg Temperatures - LESS THAN 543°F**	<u>WHEN</u> RCS hot leg temperatures less than 543°F, <u>THEN</u> Go To Step 21.

Incorrect

Purpose and Entry Conditions

(Page 1 of 1)

1. **PURPOSE**

   This procedure provides the necessary instructions for transferring the safety injection system and containment spray system to the recirculation mode.

2. **ENTRY CONDITIONS**

   When RWST level lowers to less than 27%.

- END -

incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

5.	**Stop Pumps To Obtain The Following Conditions:**
	
	• SI Pumps - ONE RUNNING
	• RHR Pumps - ALL STOPPED
	• Charging Pumps - ALL STOPPED
	• CV Spray Pumps - MAXIMUM ONE RUNNING
		Dispatch the BOP Operator to locally trip ANY pump that fails to stop:
		
		• Bus E-1
		  • SI Pump A - CMPT 21C
		  • RHR Pump A - CMPT 22A
		  • CV Spray Pump A - CMPT 19A
		  • SI Pump B - CMPT 22B
		  • Charging Pump B - CMPT 21B
		
		• Bus E-2
		  • SI Pump C - CMPT 23B
		  • RHR Pump B - CMPT 26B
		  • CV Spray Pump B - CMPT 25C
		  • SI Pump B - CMPT 29B
		  • Charging Pump C - CMPT 23A
		
		• DS Bus
		  • Charging Pump A - CMPT 34B

<u>NOTE</u>

Attachment 1 will locally close valves that have lost power due to an electrical train failure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

6.	**Close The DISCH Valves Associated With Any Stopped CV Spray Pump:**
	
	• CV SPRAY PUMP A
	  • SI-880A
	  • SI-880B
	
	• CV SPRAY PUMP B
	  • SI-880C
	  • SI-880D
		<u>IF</u> a valve has failed <u>AND</u> failure is <u>NOT</u> due to an electrical train failure, <u>THEN</u> Dispatch an Operator to locally close the valve.

19. 028 AA2.03 001

Given the following plant conditions:

-Reactor is at 100% RTP

-"A" & "B" Charging pumps are OOS

-HIC-121, CHARGING FLOW VALVE, controller indicates 100% output

-CVC-297 A/B/C are full open

-Seal Injection flow are all about 6.6 GPM

CVC-297A, RCP "A" SEAL WATER FLOW CONTROL VALVE
CVC-297 B, RCP "B" SEAL WATER FLOW CONTROL VALVE
CVC-297 C, RCP "C" SEAL WATER FLOW CONTROL VALVE
LT-459, CH I PRZR LEVEL
LT-461, CH III PRZR LEVEL

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

Based off the conditions above, LT- __(1)__ has failed low. ITS LCO 3.4.17, Chemical and Volume Control System, requirements __(2)__ met.

A. (1) 461
(2) are **NOT**

B✓ (1) 459
(2) are **NOT**

C. (1) 461
(2) are

D. (1) 459
(2) are

The correct answer is B

A) Incorrect. LT-461 is currently not the controlling channel. In a normal line-up, LT-459 is the selected controlling channel. Plausible if LT-461 was selected as the controlling channel, the same indications would apply. The second part of the distractor is correct since ITS LCO 3.4.17 is NOT met since the charging pumps are not operable.

B) Correct. LT-459 is the controlling channel normally selected for control. If it fails low, the crew will enter AOP-025 and close HIC-121 and fully open CVC-297 A/B/C to restore RCP seal injection flows while attempting to minimize PZR level rise. ITS LCO 3.4.17 requirements are NOT met because there are two charging pumps not operable.

C) Incorrect. LT-461 is currently not the controlling channel. In a normal line-up, LT-459 is the selected controlling channel. Plausible if LT-461 was selected as the controlling channel, the same indications would apply. The crew will enter LCO 3.4.17. Plausible because the students may focus in on seal injection flows being greater than the TS limit and forget that you need two charging pumps to meet the requirements of this LCO.

D) Incorrect. LT-459 is the controlling channel normally selected for control. If it fails low, the crew will enter AOP-025 and close HIC-121 and fully open CVC-297 A/B/C to restore RCP seal injection flows while attempting to minimize PZR level rise. The crew will enter LCO 3.4.17. Plausible because the students may focus in on seal injection flows being greater than the TS limit and forget that you need two charging pumps to meet the requirements of this LCO.

Question: 19
Tier/Group: 1/2
K/A Importance Rating: RO 2.8 SRO 3.3
K/A: 028 Pressurizer (PZR) Level Control Malfunction
AA2: Ability to determine and interpret the following as they apply to
the Pressurizer Level Control Malfunctions:
AA2.03: Charging subsystem flow indicator and controller
References: Sim/Plant design, AOP-025, LCO 3.4.17
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective of AOP-025 Lesson Plan, Objective 3 PZR/PRT lesson plan
Question Source: New
Question History:
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 43.5 / 45.13)

Meets the K/A because from the indications of flow(RCP Seal Injection) and controller position(HIC-121), the student needs to determine what PZR level control(LT-459) has malfunctioned.

Correct

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Chemical and Volume Control System (CVCS)

LCO 3.4.17 Reactor Coolant Pump (RCP) seal injection shall be OPERABLE, with:

a. Two charging pumps shall be OPERABLE; and

b. Two Makeup Water Pathways from the Refueling Water Storage Tank (RWST) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required charging pump inoperable.	A.1 Restore required charging pump to OPERABLE status.	24 hours
B. One Makeup Water Pathway from the RWST inoperable.	B.1 Restore Makeup Water Pathway from the RWST to OPERABLE status.	24 hours*
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

(continued)

* The Completion Time for Required Action B.1 to Restore Makeup Water Pathway from the RWST to OPERABLE status is allowed to be 72 hours during Operating Cycle 26.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Seal injection to any RCP not within limit. AND Both required charging pumps inoperable.	D.1 Initiate action to restore seal injection to affected RCP(s). AND	Immediately
	D.2 Be in MODE 3. AND	6 hours
	D.3 Cool down and depressurize the RCS to a pressure of < 1400 psig.	12 hours
E. Seal injection to any RCP not within limit. AND At least one charging pump OPERABLE.	E.1 Initiate action to restore seal injection to affected RCP(s) AND	Immediately
	E.2 Be in MODE 3. AND	6 hours
	E.3 Be in MODE 5.	36 hours
F. Both Makeup Water Pathways from the RWST inoperable.	F.1 Be in MODE 3. AND	6 hours
	F.2 Be in MODE 5.	36 hours

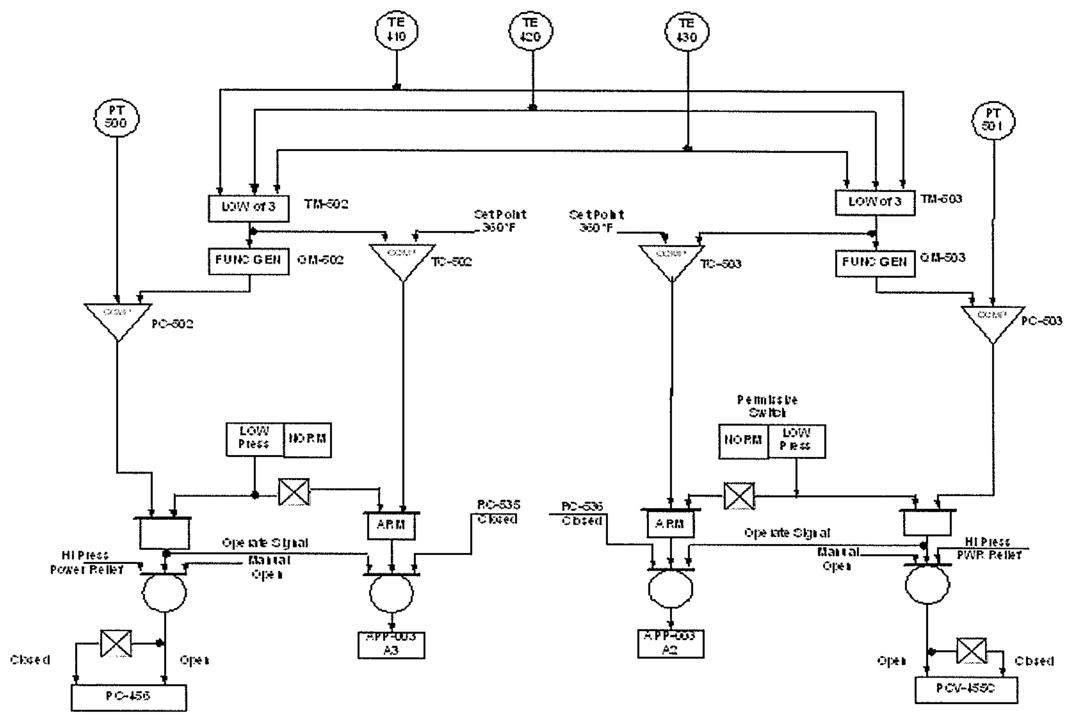
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify seal injection flow of ≥ 6 gpm to each RCP.	12 hours
SR 3.4.17.2	Verify seal injection flow of ≥ 6 gpm to each RCP from each Makeup Water Pathway from the RWST.	18 months
SR 3.4.17.3	For Makeup Water Pathways from the RWST to be OPERABLE, SR 3.5.4.2 is applicable.	In accordance with SR 3.5.4.2

Correct

There is one alarm associated with each channel of LTOPP. It actuates for 3 reasons: (1) RCS temperature is less than 365ºF and LTOPP is not selected on the key switch for OVERPRESSURE PROTECTION, (2) The PORV has received an actuation signal based upon current pressure and temperature or (3) the associated Block valve is shut.

Figure 22 LTOPP Logic



PZR Level Control

PZR level is controlled by controlling charging pump speed. The level is programmed to ramp up as Tavg increases by LC-459G. This maintains approximately constant mass in the RCS as Tavg is increased and the coolant in the RCS expands. Level program is 22.2% at Tavg of 547ºF and 53.3% at Tavg of 575.9ºF.

There are 3 PZR level channels LT-459, LT-460 and LT-461. LC-459G the PZR level controller is normally fed by level channel LT-459 but can be replaced by LT-461 with a selector switch on the RTGB. The output of LC-459G is then fed to the charging pump speed controllers to control speed of the charging pump if their controllers are selected to Auto.

If PZR level increases 5% above program LC-459D will turn on the backup heaters and sound an annunciator for High Level Heaters on.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION B

Pressurizer Level Transmitter Failure

(Page 2 of 5)

6. **Check RCP Seal Injection Flow - BETWEEN 8 GPM AND 13 GPM**

   Locally throttle RCP SEAL WATER FLOW CONTROL VALVE(s) to obtain flow to each RCP between 8 gpm and 13 gpm.

   - CVC-297A
   - CVC-297B
   - CVC-297C

   IF required to maintain 8 GPM flow, THEN throttle HIC-121, CHARGING FLOW Valve while maintaining Charging Pump Discharge pressure less than 2500 psig.

   IF the normal Seal Injection Range can NOT be maintained, THEN an expanded range of between 6 gpm and 20 gpm may be used.

   Check ITS LCO 3.4.17 for applicability.

7. **Check Number Of Operable PZR Level Channels - GREATER THAN ONE**

   Go To Step 15.

8. **Place LM-459, PZR LEVEL, In The Switch Position For The Alternate Channel Below:**

FAILED CHANNEL	SWITCH POSITION
LT-459	461 REPL 459
LT-460	461 REPL 460

20. 032 AA1.01 001

Given the following plant conditions:

-The plant is in Mode 3 preparing for startup.

-Several alarms are received and multiple controllers & indicators lose power.

-NI-32, Source Range Nuclear Instrument, is deenergized

-The crew has entered AOP-024, LOSS OF INSTRUMENT BUS

-An AO reports that an ILC student doing JPM walkdowns bumped into the supply breaker and the AO heard the breaker trip

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

In order to restore power to NI-32, the crew will need to reset and close the supply breaker to Instrument Bus ___(1)___ at ___(2)___.

A. (1) 1
(2) Inverter A

B.✔ (1) 2
(2) Inverter A

C. (1) 1
(2) MCC-5

D. (1) 2
(2) MCC-5

The correct answer is B

A) Incorrect. IB-1 is the power supply to NI-31. Plausible since NI-32 is powered from an Instrument Bus. Also, Inverter A is the correct supply to the instrument bus that supplies power to NI-32.

B) Correct. IB-2 powers NI-32. Since there are no faults and they know the reason for the breaker opening, they need to just shut the supply breaker on Inverter A per AOP-024.

C) Incorrect. IB-1 is the power supply to NI-31. Plausible since NI-32 is powered from an Instrument Bus. Second half of distractor is plausible since IB-1 is powered from MCC-5.

D) Incorrect. IB-2 is the correct power supply to NI-32. The second half of the distractor is plausible since MCC-5 is the power supply to one of the instrument busses.

Question: 20
Tier/Group: 1/2
K/A Importance Rating: RO 3.1 SRO 3.4
K/A: 032 Loss of Source Range Nuclear Instrumentation
AA1: Ability to operate and / or monitor the following as they apply to the Loss of Source Range Nuclear Instrumentation:
AA1.01: Manual restoration of power
References: Sim/Plant design, AOP-024
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 6 of NIS Lesson Plan, Objective 4 AOP-024 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR41.7 / 45.5 / 45.6)
Comment:

Meets the K/A because the student has to determine which Instrument Bus supply breaker to operate in order to restore power to NI-32.

It was discussed with the Chief Examiner that an appropriate way to meet the K/A was to ask what instrument bus would have to be manually restored to power the source range NI that lost power since we do not have a specific procedure to address a Loss of Source Range NI's.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- IF the cause for the loss of Instrument Bus is known and is not due to a fault, THEN one attempt to reset and close the open breaker in the Instrument Bus Normal Supply should be attempted.
- MCC-8 may only carry one Instrument Bus unless needed for nuclear safety concerns.

19. **Check Affected Instrument Bus - ENERGIZED**

Locally perform the applicable step below:

- IB-1 through IB-4
  a. IF the cause is known AND NOT a fault, THEN attempt to reset and close the open Instrument Bus normal supply breaker.
  b. IF MCC-8 is supplying an Instrument Bus, THEN Go To Step 74.
  c. Transfer the affected Instrument Bus to the alternate (MCC-8) power supply.

OR

- IB-6 through IB-9

  Attempt to reset AND reclose the appropriate tripped feeder breaker as determined below:

INSTRUMENT BUS LOST	FEEDER BREAKER
IB-6	Instrument Bus No. 1, Bkr 10
IB-7	Instrument Bus No. 2, Bkr 11
IB-8	Instrument Bus No. 3, Bkr 25
IB-9	Instrument Bus No. 4, Bkr 12

20. **Stop All Radioactive Batch Releases**

Correct

INSTRUMENT BUS NO. 2
Location: Safeguards Room, East Wall
Normal Power: Inverter "A" / Alternate Power: MCC-8 (2GL)

SPARE		Instrument Bus No.2 Power Supply (From INST BUS 2 PWR XFER SW)	
1	NIS Cabinet "B" (CWD-444)	2	NIS Cabinet "B" (CWD-444)
3	Hagan Rack 4 (CWD-476) Isolator Rack 29, Channel 2	4	Hagan Rack 7 (CWD-416)
5	Hagan Rack 9 (CWD-475)	6	Hagan Rack 11 (CWD-460)
7	Hagan Rack 12 (CWD-456); FR-113 rec (CWD 481, 964, 5379-3484)	8	Hagan Rack 13 (CWD-418)
9	RTGB A:TB-SM 27 and 28 RTGB C:TB-UE 45 and 46 (NOTE 1) TR-413 rec (CWD 468, 963, 5379-3502); NR- 47 rec (CWD 443, 963); FR-488 rec (CWD 964) FQ-124, FQ-127, FQ-130, FR-124 (CWD 963, 1830)	10	Quenching Valve Control
11	Instrument Bus No.7A and 7B	12	RMS console #1 receptacle (powers generator temp recorder) (CWD 877)
13	Spared by EC 79124	14	Lundell AC (NOTE 2)
15	Turning gear automatic control (CWD-731)	16	Waste Disposal System Panel Misc. Relays (CWD 367, 368)
17	MDAFW Pump "A" Flow control valve FCV-1424, FIC-1424 (CWD-657)	18	**(See EDP-005, 65VDC Dist Panel "B for individual loads from PNL Q-6)** PNL-Q-6-45V, (CWD-966) **(See EDP-005, Aux Panel GD for individual loads from PNL Q-4)** PNL-Q-4-65V (CWD-966); PI-1421A (CWD 602, 966); LI-1454B (CWD 601A, 966); LI-1941 (CWD 601, 966); PI-1911(CWD 594, 966)

NOTE 1: CWD's 433, 443, 468, 481, 963, and 964.

NOTE 2: FIC-154, FIC-155, and FIC-156 are powered by Lundell AC power.

Incorrect

MCC-5 POWER SUPPLY: 480V BUS E-1 (52/21A) LOCATION: AUX BLDG HALLWAY			
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
3J	RHR-744A, RHR LOOP TO RCS COLD LEG RHR-744A	220	52/MCC-5(3J)
3M	CC-749A, RHR HEAT EXCHANGER A COOLING WATER OUTLET CC-749A	218	52/MCC-5(3M)
4BL	FEED TO INSTRUMENT BUS 1 INST-1	N/A	52/MCC-5(4BL)
4BR	EDG A AUX PANEL DG-A-AUX-PNL	945	52/MCC-5(4BR)
4DL	0.30 KVA TRANSFORMER FOR HVS-6 CPT/HVS-6	561	N/A
4DR	0.30 KVA TRANSFORMER FOR HVE-18 CPT/HVE-18	563	N/A
4G	EDG A ROOM EXHAUST FAN, HVE-18 HVE-18	563	52/MCC-5(4G)
4M	AUX BUILDING EXHAUST FAN, HVE-2A HVE-2A	540	52/MCC-5(4M)
5B	BATTERY CHARGER A-1 BAT-CHRGR-A-1	955	52/MCC-5(5B)
5D	EDG A ROOM SUPPLY FAN, HVS-6 HVS-6	561	52/MCC-5(5D)
5G	BORIC ACID TRANSFER PUMP A BA-XFER-PMP-A	191	52/MCC-5(5G)
5M	CONTROL ROD DRIVE MECHANISM COOLING FAN, HVH-5A HVH-5A	516	52/MCC-5(5M)

21. 033 AG2.2.22 001

Given the following plant conditions:

-Power is above P-6

-Power is below P-10

-Power ascension is in progress

-NI-35, INTERMEDIATE RANGE NUCLEAR INSTRUMENT, fails as is

-NI-36, INTERMEDIATE RANGE NUCLEAR INSTRUMENT, fails as is

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

The maximum time the crew has to ___(1)___ is ___(2)___ IAW LCO 3.3.1, REACTOR PROTECTION SYSTEM (RPS) INSTRUMENTATION.

A. (1) suspend operations involving positive reactivity additions
(2) 15 minutes

B.✓ (1) suspend operations involving positive reactivity additions
(2) immediately

C. (1) increase thermal power to >P-10
(2) 15 minutes

D. (1) increase thermal power to >P-10
(2) immediately

The correct answer is B

A) Incorrect. This is the correct action to take. 15 minutes is incorrect. Plausible since this is a completion time associated with a Source Range Channel being inoperable in LCO 3.9.2.

B) Correct. IAW LCO 3.3.1 Condition G, they will suspend operations involving positive reactivity additions. This should be completed immediately.

C) Incorrect. This action is incorrect. Plausible because you would do this if under these power conditions if only one IR channel was inoperable. 15 minutes is incorrect. Plausible since this is a completion time associated with a Source Range Channel being inoperable in LCO 3.9.2.

D) Incorrect. This action is incorrect. Plausible because you would do this if under these power conditions if only one IR channel was inoperable. The time frame is correct.

Question: 21
Tier/Group: 1/2
K/A Importance Rating: RO 4.0 SRO 4.7
K/A: 033 Loss of Intermediate Range Nuclear Instrumentation
AG 2.2.22: Knowledge of limiting conditions for operations and safety limits.
References: Sim/Plant design, LCO 3.3.1
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 14 "A" of NIS Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.5 / 43.2 / 45.2)

Meets the K/A because the student must know the LCO for a loss/failure of the Intermediate Range NI's.

Correct

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 The RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 Open reactor trip breakers (RTBs).	55 hours

(continued)

Correct

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One channel inoperable.	E.1	Place channel in trip.	6 hours
	<u>OR</u>		
	E.2	Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	2 hours
	<u>OR</u>		
	F.2	Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1	--------NOTE-------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. -------------------- Suspend operations involving positive reactivity additions.	Immediately
	<u>AND</u>		
	G.2	Reduce THERMAL POWER to < P-6.	2 hours

(continued)

INCORRECT

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 The RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status. OR B.2.1 Be in MODE 3. AND B.2.2 Open reactor trip breakers (RTBs).	48 hours 54 hours 55 hours

(continued)

INCORRECT

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One channel inoperable.	E.1	Place channel in trip.	6 hours
	OR		
	E.2	Be in MODE 3.	12 hours
F. THERMAL POWER > P-6 and < P-10, one Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	2 hours
	OR		
	F.2	Increase THERMAL POWER to > P-10.	2 hours
G. THERMAL POWER > P-6 and < P-10, two Intermediate Range Neutron Flux channels inoperable.	G.1	--------NOTE-------- Limited boron concentration changes associated with RCS inventory control or limited plant temperature changes are allowed. --------------------	
		Suspend operations involving positive reactivity additions.	Immediately
	AND		
	G.2	Reduce THERMAL POWER to < P-6.	2 hours

(continued)

22. 060 AK3.02 001

Given the following plant conditions:

-High radiation alarms have just been received from R-20, FUEL HANDLING BUILDING LOWER LEVEL, and R-30, FUEL HANDLING BUILDING LOWER LEVEL HIGH RANGE

-The Inside Auxiliary Operator reports the following:
- -WGDT "A" is at 40 psig and is in service
- -WGDT "B" is at 20 psig and is in standby
- -WGDT "C" is at 80 psig and is on cover gas
- -WGDT "D" is at 60 psig with pressure slowly lowering

-AOP-009, ACCIDENTAL GAS RELEASE FROM A WGDT, has been entered

-The crew has started HVE-5A, AUX BLDG CHARCOAL EXH FAN

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

The basis for starting HVE-5A IAW AOP-009 is to ___(1)___ . WGDT "D" will be placed on cover gas and equalized with WGDT ___(2)___ .

A. (1) minimize any radiological release
(2) "A"

B.✓ (1) minimize any radiological release
(2) "B"

C. (1) limit the concentration of explosive gases
(2) "B"

D. (1) limit the concentration of explosive gases
(2) "A"

The correct answer is B

A) Incorrect. The basis for starting HVE-5A IAW AOP-009 is to minimize any release. WGDT "D" will be placed on cover gas, however, WGDT "B" is at a lower pressure and AOP-009 has you equalize pressure with a tank that is at a lower pressure by also opening its cover gas valve.

B) Correct. The basis for starting HVE-5A IAW AOP-009 is to minimize any radiological release. The charcoal filters will remove any airborne particulates. WGDT "D" will be placed on cover gas and then equalized with WGDT "B" since it is a non-leaking tank with the lowest pressure..

C) Incorrect. The first part of the distractor is plausible since the WGDTs do contain hydrogen and creating an explosive mixture is a concern when lowering the WGDT pressure. The second half of the distractor is correct. WGDT "D" will be placed on cover gas and then equalized with WGDT "B" since it is a non-leaking tank with the lowest pressure.

D) Incorrect. The first part of the distractor is plausible since the WGDTs do contain hydrogen and creating an explosive mixture is a concern when lowering the WGDT pressure. WGDT "D" will be placed on cover gas, however, WGDT "B" is at a lower pressure and AOP-009 has you equalize pressure with a tank that is at a lower pressure by also opening its cover gas valve.

Question: 22
Tier/Group: 1/2
K/A Importance Rating: RO 3.3 SRO 3.5
K/A: 060 Accidental Gaseous Radwaste Release
AK3: Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste:
AK3.02: Isolation of the auxiliary building ventilation
References: Sim/Plant design, AOP-009
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of AOP-009 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.5,41.10 / 45.6 / 45.13)
Comments: Per conversation with Chief Examiner on 5/22/2013, acceptable to write question for starting HVE-5A to minimize release vice isolating the Auxiliary Building.

Meets the K/A because the student must know the reason for starting HVE-5A IAW AOP-009. We do not isolate our auxiliary building ventilation during this AOP. We send the Aux Building air through charcoal filters prior to it exiting the Aux building. The normal path is isolated and forced through the charcoal filters when you start HVE-5A.

Correct

BASIS DOCUMENT, ACCIDENTAL GAS RELEASE FROM A WGDT

DISCUSSION:

It is expected that the operator will be alerted to this event due to radiation monitor alarms and/or verbal notification that a WGDT pressure is decreasing. The objective of this procedure is to reduce leaking tank pressure as rapidly as possible in order to terminate the release. This is accomplished by first ensuring non affected tanks are selected for in service and standby. The leaking tank is then aligned as the cover gas tank and equalized with a tank at lower pressure if possible. Further depressurization of the leaking tank is accomplished by manually pumping down the vent header to the in service tank.

INDIVIDUAL STEP DESCRIPTION:

Step	Description
1	This step directs making a PA announcement for procedure entry. The PA announcement should include the procedure being entered and the reason for entry. This will help to alert personnel that may be called upon for assistance for the event. Since this is an informational announcement that does not direct any local or plant evacuations or warn of any hazardous personnel situations, it is not necessary to use the VLC switch when making this announcement.
2	The first step is to evacuate all unnecessary personnel from the area. At a minimum, this is expected to be the WGDT Room and any other areas in which rad monitor indications dictate.
3	This step informs RC Personnel so that they can determine any other radiological concerns as may be appropriate.
4	This step checks that at least one Aux Bldg exhaust fan is running in order to start one of the HVE-5 fans in the next step.
5	This step places the charcoal bed in service in order to minimize any release.
6	This step checks to ensure that the R-14 noble gas channel is operable to monitor the release. If not, the R-11/R-12 channels are placed in service to monitor the vent stack. When this is done, the CV is evacuated due to the loss of monitoring capability.
7	This step monitors R-5 and R-21 to determine the need for placing the upper Fuel Handling Building Ventilation System in service in the fuel handling mode.

Correct

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

14. At The WDBRP, Place The Leaking WGDT On Cover As Follows:

 a. Close the COVER GAS valve on the tank aligned for cover gas:

  - WD-1629, WASTE GAS DECAY TANK A SUPPLY TO CVCS HUT

   OR

  - WD-1630, WASTE GAS DECAY TANK B SUPPLY TO CVCS HUT

   OR

  - WD-1631, WASTE GAS DECAY TANK C SUPPLY TO CVCS HUT

   OR

  - WD-1632, WASTE GAS DECAY TANK D SUPPLY TO CVCS HUT

 b. Open the COVER GAS valve on the leaking WGDT:

  - WD-1629, WASTE GAS DECAY TANK A SUPPLY TO CVCS HUT

   OR

  - WD-1630, WASTE GAS DECAY TANK B SUPPLY TO CVCS HUT

   OR

  - WD-1631, WASTE GAS DECAY TANK C SUPPLY TO CVCS HUT

   OR

  - WD-1632, WASTE GAS DECAY TANK D SUPPLY TO CVCS HUT

Correct

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
15.	At The WDBRP, Determine If The Leaking WGDT Can Be Equalized With A Tank At A Lower Pressure As Follows:	
	a. Check WGDT pressures - A NON-LEAKING TANK HAS THE LOWEST PRESSURE	a. Go To Step 16.
	b. Open the COVER GAS valve on a tank with a lower pressure: • WD-1629, WASTE GAS DECAY TANK A SUPPLY TO CVCS HUT <u>OR</u> • WD-1630, WASTE GAS DECAY TANK B SUPPLY TO CVCS HUT <u>OR</u> • WD-1631, WASTE GAS DECAY TANK C SUPPLY TO CVCS HUT <u>OR</u> • WD-1632, WASTE GAS DECAY TANK D SUPPLY TO CVCS HUT	

(CONTINUED NEXT PAGE)

incorrect

B 3.20 WASTE GAS DECAY TANKS - OXYGEN CONCENTRATION

BASES

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the Waste Gas Holdup System is maintained below the flammability limits of hydrogen and oxygen. This is accomplished by maintaining the oxygen concentration less than 4% through procedural controls. Maintaining the concentration of oxygen below the flammability limit provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

TR 3.20.1 specifies that monitoring be performed on the in-service tank, as that is the tank with the potential for changing gas concentrations.

Technical Specification Section 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," states that this program shall include limits for concentrations of hydrogen and oxygen in the Waste Gas Decay Tanks that are appropriate to the system's design criteria. By always requiring the oxygen concentration to be less than 4%, this TRM limit fully meets the requirement to establish appropriate limits for the concentration of the mixture of hydrogen and oxygen to preclude an explosive mixture.

Incorrect

4.6 Instrument Air is available for all air operated valves in accordance with OP-905.

4.7 Component Cooling Water is available for the Gas Compressor Heat Exchanger and make-up to the compressor in accordance with OP-306.

4.8 Service Water is being supplied to the HVAC systems in accordance with OP-903.

4.9 The HVAC systems in the Auxiliary and Fuel Handling buildings are in service and the Plant Vent Fan is in operation in accordance with OP-906.

4.10 The Radiation Monitoring System is in service in accordance with OP-920.

5.0 PRECAUTIONS AND LIMITATIONS

5.1 Maintain at least 0.5 psig on the Vent Header at all times to prevent air leakage into the system.

5.2 Due to the presence of hydrogen, oxygen must be kept out of the Waste Gas System. In the event that equipment normally vented to the vent header has been open to the atmosphere for any reason, equipment must be properly purged with Nitrogen prior to handling gases.

5.3 Ensure that adequate capacity is available (equivalent to one Gas Decay Tank) at all times to receive Gaseous Waste.

5.4 Ensure positive pressure addition to CVCS Holdup Tanks at all times by proper operation of PCV-1027 and PCV-1049, to prevent possible formation of vacuum in tanks. Prior to maintenance on one valve, ensure the other is operable.

5.5 If high Oxygen concentration is indicated for a Waste Gas Decay Tank, refer to OP-703.

5.6 When the selected Gas Decay Tank has been pressurized to 110 psig, flow will be automatically diverted to the backup tank. At this time select a new backup tank in accordance with the OP-702 section for Shifting of the Waste Gas Decay Tanks in Automatic.

Incorrect

5.7 Complete the OP-702 section for Shifting of the Waste Gas Decay Tanks Manually each time the Gas Decay Tank lineup is changed.

5.8 The quantity of radioactivity contained in each Waste Gas Decay Tank shall at all times be limited to 1.9 X 10^4 curies noble gases (considered as Xe-133).

5.9 When the quantity of radioactive materials in any Waste Gas Decay Tank exceeds the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit. Use caution as radiation levels in the gas decay tank room will be very high.

5.10 Before the pressure of a Gas Decay Tank drops to 5 psig while providing cover gas to the CVCS HUTs, another Gas Decay Tank should be placed on cover to ensure that a vacuum will not be formed in the CVCS HUTs and thus avoid possible tank collapse.

5.11 Ensure sampling of the Gas Decay Tank is performed for Oxygen concentration prior to placing it on CVCS HUT cover gas. To minimize a buildup of Oxygen in the Waste Gas System, Oxygen concentration should be less than or equal to 2%.

5.12 The Gas Analyzer provides an alarm if Oxygen concentration for the In Service Gas Decay Tank reaches 3%. (ESR 97-00446)

5.13 The oxygen concentration must be maintained within the limits of PLP-100, Technical Requirements Manual, TRMS 3.20. Oxygen concentration in the four Waste Gas Decay Tanks shall be less than or equal to 4% by volume.

23. 067 AK1.02 001

Given the following plant conditions:

-A fire breaks out on the RTGB

-The crew has entered AOP-041, RESPONSE TO FIRE EVENT, and subsequently, DSP-001, ALTERNATE SHUTDOWN DIAGNOSTIC

-The crew has tripped the reactor

CVC-200 A/B/C, LTDN ORIFICE
PCV-455C, PZR PORV
PCV-456, PZR PORV

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

Prior to leaving the control room, the OAC will __(1)__. This is performed to __(2)__ .

A. (1) Verify CVC-200 A/B/C closed
(2) prevent spurious operation

B✓ (1) Isolate PCV-456 & PCV-455C
(2) prevent spurious operation

C. (1) Verify CVC-200 A/B/C closed
(2) allow for remote operation of these valves

D. (1) Isolate PCV-456 & PCV-455C
(2) allow for remote operation of these valves

The correct answer is B

A) Incorrect. Plausible if the fire was not in the control room, DSP-001 would have you go to AOP-004. In AOP-004, you would verify closed the MSIV's. This is the reason why you isolate PCV-456 and PCV-455C. Plausible since a fire on the RTGB could affect the switches for these valves and cause them to open and close.

B) Correct. DSP-001 has you isolate your PORV's, PCV-456 and PCV-455C. The reason for this is to prevent spurious operation of the PORV's.

C) Incorrect. Plausible if the fire was not in the control room, DSP-001 would have you go to AOP-004. In AOP-004, you would verify closed CVC-200 A/B/C. Plausible because these valves can be operated remotely from the DS panel in the charging pump room.

D) Incorrect. You do isolate PCV-456 & PCV-455C. The reason is not to allow for remote operation of the PORV's. Plausible because you can remotely operate PCV-456 from the 4Kv room and this is done in DSP-012, PRESSURIZER PORV CONTROL/POWER REPAIR PROCEDURE.

Question: 23
Tier/Group: 1/2
K/A Importance Rating: RO 3.1 SRO 3.9
K/A: 067 Plant fire on site
AK1: Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site:
AK1.02: Fire fighting
Reference(s): Sim/Plant design, AOP-004, DSP-001, AOP-041 and their basis documents.
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 3 of DSP-001 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.8 / 41.10 / 45.3)
Comments:

This question meets the K/A because it asks the students for the operation implications(why we shut the PORV's) during a fire in the RTGB.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 3 and 4 are time critical and shall be performed within 10 minutes.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 3.	**Fire CONFIRMED In One Of The Following Areas:** • South Cable Vault (Fire Zone 10) • Unit 2 Cable Spread Room (Fire Zone 19) • Emergency Switchgear Room (Fire Zone 20) • Control Room (Fire Zone 22)	Go To Step 5
4.	**Perform the following:** a. Close PZR PORV Block Valves: • RC-535, PORV BLOCK • RC-536, PORV BLOCK b. At The Containment Fire Protection System Panel Place The Following Key-Operated Switches To The ISOLATED Position: • PCV-456 POWER ISOLATION • PCV-455C POWER ISOLATION	

Correct

INDIVIDUAL STEP DESCRIPTION:

RNP STEP	BASIS

MAIN BODY

1 BASIS

This step checks that the fire is inside the Fire Brigade Response Area. If the fire is outside the Fire Brigade Response Area, the RNO will transition the operator to near the end of the procedure to establish communication with the appropriate offsite agency and then exit this procedure.

N2 BASIS

Note allows the SM to determine if the procedure is needed based on event circumstances. For instance, if a fire is reported and already extinguished before performing Step 4 then the procedure may be exited

2 BASIS

The Fire Brigade is dispatched to the fire scene as soon as possible to mitigate the consequences of the fire effects. Communication information is included in the RNO to aid in notifications. Portable radios are credited for communications with the SRO in the event that the Public Address system is not available. Mini-cell phones, although not credited, would most likely be available through the Unit #1 repeater station since the cable for the repeater from the PBX Building to Unit #1 does not enter the fire area.

N3 BASIS

This note informs the operator that the following two steps must be performed within 10 minutes of the event initiation. This helps to raise the awareness of the action for the operator to ensure there are no delays in completing this action. Refer to calculation RNP-E-8.050, Appendix R Transient Analysis and Timeline Evaluation.

3 BASIS

This step checks if fire is in one of these areas which contain cables related to the PZR PORVs. If fire is not in one of these areas then the RNO bypasses isolating and defeating the PZR PORVs.

4 BASIS

This step closes the PZR PORV block valves as defense in depth based on the Revised Design Methodology that a concern remains involving the possibility of spurious operation of the PZR PORVs due to cable to cable hot shorts of the correct polarity. The key operating switches for PCV-456 and PCV-455C are operated to defeat these valves to prevent or mitigate the effects of spurious actuation of these components which could adversely affect the safe shutdown capability.

5 BASIS

This step performs an evacuation of the fire area using the local evacuation alarm or CV Evacuation alarm for personnel protection.

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

The EOP Network does NOT apply while in this procedure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Verify Reactor Trip As Follows:** • REACTOR TRIP MAIN AND BYPASS BREAKERS - OPEN	Manually trip the Reactor.
2.	**Verify Turbine Trip As Follows:** • All TURBINE STOP VALVES - CLOSED OR • All GOVERNOR VALVES - CLOSED	Manually trip the Turbine by simultaneously depressing the THINK and TURBINE TRIP Pushbutton.
3.	**Verify MSIVs AND MSIV BYPASSES - CLOSED**	
4.	**Place VLC Switch In The EMERG Position**	
5.	**At The RTGB, Verify LCV-115B, EMERG MU TO CHG SUCTION - OPEN**	IF LCV-115B will NOT open, THEN Go To Step 7.
6.	**At The RTGB, Verify LCV-115C, VCT OUTLET - CLOSED**	IF LCV-115C will NOT close, THEN close LCV-115B, EMERG MU TO CHG SUCTION.
7.	**Verify The Following Valves - CLOSED** • CVC-200A, LTDN ORIFICE • CVC-200B, LTDN ORIFICE • CVC-200C, LTDN ORIFICE	

Incorrect

CONTINUOUS USE

Purpose and Entry Conditions

(Page 1 of 1)

NOTE

MOTIVE FORCE as used in this procedure is the compressed gas used to provide opening force for the valves.

1. PURPOSE

   This procedure provides control/power connections to PCV-456, PZR PORV in the event that the normal control circuits are unavailable. This procedure also provides a backup motive force to PCV-455C and PCV-456, if necessary, in the event Instrument Air AND Nitrogen Supply Systems are unavailable to the PZR PORVs.

2. ENTRY CONDITIONS

   - RCS pressure control using normal OR auxiliary spray flow is unavailable AND RTGB control of PCV-455C and PCV-456 is unavailable. PCV-456 is undamaged AND DS power is available to connect repair cables to control pressurizer pressure.

   OR

   - Instrument Air AND Nitrogen Supply Systems are unavailable to supply a motive force for PZR PORVs.

\- END -

incorrect

Letdown Orifice Isolation Valves (CVC-200A, 200B, 200C)

4. CVC-200A, B, & C can be operated at the Dedicated Shutdown Panel (DSP) located in the charging pump room.

- Each valve has a LOCAL/REMOTE and a OPEN/CLOSE switch
- APP-036-J6, SHUTDOWN EQUIP IN LOCAL, will alarm.

24. 068 AG2.2.37 001

Given the following plant conditions:

-A fire has broke out in the Hagan Room

-It has spread to the Control Room

-The crew is currently in DSP-002, HOT SHUTDOWN USING THE DEDICATED/ALTERNATE SHUTDOWN SYSTEM

-Attachment 1, TURBINE BUILDING OPERATOR ACTIONS, are complete

-The DS Bus is energized

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

One of the operators will start "A" CCW pump from __(1)__. MDAFW pumps __(2)__ available.

A. (1) the Charging Pump room
(2) are

B.✓ (1) the Charging Pump room
(2) are **NOT**

C. (1) the Rod Control room
(2) are

D. (1) the Rod Control room
(2) are **NOT**

The correct answer is &ANSWER%

A) Incorrect. The charging pump room is correct. Components powered by E1/E2 are available is incorrect, therefore, the MDAFW pumps are not available. Plausible if the student does not know that DSP-002, attachment 1, has you de-energize all E1 and E2 busses.

B) Correct. The Auxiliary Building Operator will start the "A" CCW pump from the Charging Pump room on the DS Panel. E1 and E2 are de-energized in DSP-002, attachment 1, to prevent an operator from getting injured due to spurious operation. Therefore, the MDAFW pumps are not available.

C) Incorrect. The Rod Control room is incorrect. Plausible because SW pump D can be powered from the DS bus and operated from the Rod Control Room. Components powered by E1/E2 are available is incorrect, therefore, the MDAFW pumps are not available. Plausible if the student does not know that DSP-002, attachment 1, has you de-energize all E1 and E2 busses.

D) Incorrect. The Rod Control room is incorrect. Plausible because SW pump D can be powered from the DS bus and operated from the Rod Control Room. E1 and E2 are de-energized in DSP-002, attachement 1, to prevent an operator from getting injured due to spurious operation. Therefore, the MDAFW pumps are not available.

Question: 24
Tier/Group: 1/2
K/A Importance Rating: RO 3.6 SRO 4.6
K/A: 068 Control Room Evacuation
G 2.2.37: Ability to determine operability and/or availability of safety related equipment.
Reference(s): Sim/Plant design, DSP-002
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 5 of DSP-002 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.7 / 43.5 / 45.12)
Comments:

This question meets the K/A because the student must know whether or not the MDAFW pumps are available based on actions that were taken in DSP-002 after the Control Room has been evacuated.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 1

TURBINE BUILDING OPERATOR ACTIONS

(Page 1 of 4)

> NOTE
>
> Time Critical Action: Establish AFW to first S/G within 27.5 minutes of event initiation.

1. Obtain The Following Prior To Leaving The Old Fire Equipment Building:

   - Two-way radio
   - Flashlight
   - Locked valve keys (Keys 1, 1a, or 1b)
   - Locked high rad area key (in holder near door)
   - DSP-002 Attachment 4, Attachment 6, AND Attachment 7

2. At the West Side Of the Condenser, Close C-44A, LCV-1417A INLET

3. Check Fire Location In Any Of The Following: — RESPONSE NOT OBTAINED: Go To Step 5.

   - E-1/E-2 ROOM
   - HAGAN ROOM

CORRECT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 1

TURBINE BUILDING OPERATOR ACTIONS

(Page 2 of 4)

4. Perform The Following:

   a. Open 4.16KV SWITCH 2F, SST-2F DISCONNECT. (Located South of the Vacuum Pump Enclosure)

   b. Open 4.16KV SWITCH 2G, SST-2G DISCONNECT. (Located South of the Vacuum Pump Enclosure)

   c. Notify SM That The SST-2F AND SST-2G Have Been Deenergized

5. Obtain Verification From The SM That The Emergency Buses Are Deenergized

   RESPONSE NOT OBTAINED: WHEN verification received, THEN observe CAUTION prior to Step 6 and Go To Step 6.

****

CAUTION

The Emergency Buses are deenergized prior to any manual motor operated valve operation to prevent possible personnel injury.

****

NOTE

MS-V1-8A, B, & C are located in the Pipe Jungle, 2nd level.

6. Locally Open MS-V1-8A, SG "A" STM SUPPLY TO STM DRIVEN AFW PUMP

   RESPONSE NOT OBTAINED: Locally open one of the following valves:

   - MS-V1-8B, SG "B" STM SUPPLY TO STM DRIVEN AFW PUMP

   OR

   - MS-V1-8C, SG "C" STM SUPPLY TO STM DRIVEN AFW PUMP

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 3

AUXILIARY BUILDING OPERATOR ACTIONS

(Page 7 of 14)

11. Notify The SM That Charging Flow Has Been Established AND The Status Of Seal Injection

> **NOTE**
>
> Attachment 11, Credited Instrumentation Available, provides a list of the credited instrumentation not affected by the fire based on the fire location.

12. At the Charging Pump Control Panel Perform the following:

    a. Verify Transfer Switch for CCW PUMP A in LOCAL position.

    b. Check CCW PUMP A Status - STOPPED.

    Response Not Obtained:

    b. IF RCP Seal Injection has been established, THEN notify SM that CCW has been established AND Go To Step 15.

    IF RCP Seal Injection has NOT been established, THEN Locally Stop CCW Pump A.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CONTINUOUS USE

ATTACHMENT 3

AUXILIARY BUILDING OPERATOR ACTIONS

(Page 8 of 14)

> **NOTE**
>
> When dumping steam from the secondary a cooldown rate of 10°F/Hr is the maximum rate allowed.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	At the Charging Pump Control Panel Start CCW PUMP A As Follows:	
	a. Check RCP Seal Injection - ESTABLISHED	a. Perform the following: 1) Locally Raise speed on the Charging Pump to 100%. 2) IF required to maintain PZR level stable, THEN coordinate with the Turbine Building Operator to dump steam from the secondary. 3) WHEN FCV-626 OR CC-736 is closed, THEN Go To Step 14.
	b. Start CCW PUMP A	
	c. Locally Check CCW flow on FI-660, CHARGING PUMP OIL COOLER RETURN FLOW - FLOW INDICATED	c. Locally Verify Charging Pump A CCW alignment correct: • CC-825C, CC TO CHG PUMP "A" OIL COOL - OPEN • CC-825F, CC FROM CHG PUMP A OIL COOL - OPEN
	d. Notify the SM that CCW flow has been established	
	e. Go To Step 15	

RNP STEP	BASIS/DIFFERENCES

2 BASIS

This step limits water flow from CST to the condenser to ensure sufficient feedwater is available to S/Gs via SDAFW Pump. The Steam Driven Auxiliary Feedwater Pump suction is normally aligned to the Condensate Storage Tank (CST). However, if CST inventory is exhausted, then manual alignment of valves AFW-24 and SW-118 is credited to supply Service Water to the pump suction. Therefore, isolation of the CST from the condenser is a contingent 'optional' step not credited nor required to meet 10CFR50, Appendix R. This step is done to prolong the time until switchover to SW is required.

The credited egress route is from the Old Fire Equipment Building to the West side of the Condenser via the South side of the Auxiliary Building (grating pathway).

3 BASIS

This step provides transitional guidance based on fire location. If the fire is in the E-1/E-2 Room or Hagan Room, SST-2F and 2G disconnects are opened. If the fire is in the E-1/E-2 room the electrical operator will be required to use another access pathway to the battery room and will not be able to trip the Emergency Buses. The Turbine Building Operator (TBO) can not align steam to the SDAFW pump until the power supply to the valves is removed. The TBO will perform part of the actions to deenergize the Bus. The bus will be deenergized by removing the offsite power source via SSTs 2F and 2G and tripping the EDGs.

If the fire is in the Hagan Room, the E-1/E-2 Bus ductwork travels through the Hagan Room prior to entering the E-1/E-2 Room. As a defense in depth measure for a fire in the Hagan Room, power will also be removed at the disconnects for SST-2F and 2G.

4 BASIS

In the event of a fire in the E-1/E-2 room or Hagan Room, the TBO opens disconnects for the power supply to the Emergency Buses from SSTs 2F and 2G. This will remove the power source from offsite supplies. The Auxiliary Building Operator (ABO) will trip the EDGs for a fire in the E-1/E-2 room, thus with a combination of these actions the busses will be deenergized.

The credited egress route is from the West side of the Condenser, up the back stairwell of the Turbine Building, East along the second deck between the Feedwater Water Heaters to the disconnect switches behind the vacuum pumps. The stairwell has an exemption granted to use the Security High Mast lights instead of the 8 hour battery pack lights. One could also use a route that goes back to the front of the Turbine Building and up the stairs to the 2nd level since this is also a credited route (has lights). This route is longer than the route up the back stairwell.

5 BASIS

The TBO verifies with the SM (or EO) that the Emergency Buses are deenergized. Subsequent steps require the power to be removed. This step either verifies or acts as a hold point until the bus has been deenergized. This step holds the operator prior to local valve operation until the Emergency Buses are deenergized to prevent possible personnel injury. The post-fire safe shutdown capability as reflected in the Separation Analysis assumes power being disconnected manually to prevent spurious actuation of the components which could affect the safe shutdown process. In addition, the analysis/methodology reflected in the licensing basis assumes that the power and control circuits for all major systems other than safe shutdown systems have been destroyed.

C6 BASIS

Caution reminds operator that the Emergency Buses are deenergized prior to any manual valve operations to prevent personnel injury.

N6 BASIS

The note provides location information to the operator.

RNP STEP | BASIS/DIFFERENCES

5 BASIS

This step de-energizes selected breakers on 125V DC Distribution Panels A and B to remove power from E1/E2 Switchgear breaker control, Auxiliary Panels DC and GC, 4 KV Buses 1, 2, 3, and 4 breaker control, and PZR PORVs PCV-455C and PCV-456 if the fire is located in any of Zones 7, 15, 16, 17,or 36 (Auxiliary Bldg. Hallway (ground floor), Aux Bldg Second Level, Battery Room, HVAC Equip Room For Control Room, And CCW Surge Tank Room respectively.)

This step is performed to prevent or mitigate the effects of spurious actuation of components which could adversely affect the safe shutdown capability. A CP&L Licensing Submittal dated February 6, 1984, classified the spurious operations which adversely effect safe shutdown into two categories. As defined in this submittal, the first category includes those spurious operations which must be prevented from occurring. The second category includes those which may be allowed to occur for a limited time period, but must be prevented from continuing past the point of jeopardizing the safe shutdown capability.

The steps for deenergizing the DC power to the air operated components affected by a fire in these areas are credited to be performed in no more than 10 minutes.

If Distribution Panels A and B cannot be deenergized, placing the RTGB control switches for components listed in Attachment 5, into their Deenergized Position will help to prevent the spurious operation of the components. (Attachment 5 is Major Components Disabled By De-energizing Distribution Panels A and B). The RNO is needed in the event of a fire in the battery room which would prevent access to the DC breakers. If the fire is in the battery room the RTGB components are not affected. This step, although appropriate, is not credited in the Separation Analysis or the RNP Appendix R licensing basis. The Separation Analysis has evaluated the spurious actuation effects of fire induced faults on the cables associated with the components of concern, and the effects have been dispositioned in Document FPP-RNP-200. A review of the Separation Analysis and FPP-RNP-200 indicates that spurious actuation of the components of concern is not subject to occurrence under condition of a Battery Room fire. However, placing the appropriate control switches at the RTGB in the deenergized position provides an additional (conservative) level of assurance that a fire in the Battery Room will not induce unacceptable spurious actuation of plant equipment capable of impacting the safe shutdown process.

6-7 BASIS

Based on which breakers on DC Distribution Panels A & B has been opened based on fire location, the appropriate notification is made to the SM. The SM is tracking status so manual/local operation of safe shutdown components can be performed once the components are de-energized.

8-9 BASIS

This step provides transitional guidance based on fire location. If the fire is in the E-1/E-2 room the electrical operator will not be able to trip the Emergency Buses. Instead, the operator will go to the first level of the auxiliary building to deenergize MCC-5 and 10 loads. This transfer of actions will save overall time and assure that time critical steps are completed within the required times.

10 BASIS

This step deenergizes Emergency Buses E-1 and E-2. This prevents spurious actuations of components powered by E-1/E-2 and also allows operation of valves fed from MCC-5.

Performing this step will disconnect Offsite Power and Emergency Diesel Generators from 480V Buses E-1 and E-2. This action will prevent or terminate the spurious actuation of components powered by these power sources, and will minimize the electrical hazards of fighting the fire.

11 BASIS

This step notifies SM that the Emergency Buses are deenergized. SM is tracking status and Emergency Buses are to be deenergized. This will prevent spurious actuations of components powered by E-1/E-2, including MCC-5 and MCC-10.

Correct

8.0 480V-E1

480V-E1 POWER SUPPLY: NORMAL - 4160V BUS 2 (52/13)LOCATION: E-1/E-2 ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
17A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
17B	**EMERGENCY DIESEL GENERATOR A TO 480V BUS E-1** 480V-E1	890	52/17B
18A	**PT'S & METERING EQUIPMENT (**)** N/A	N/A	N/A
18B	**STATION SERVICE TRANSFORMER 2F TO 480V BUS E-1** 480V-E1	892	52/18B
19A	**CV SPRAY PUMP A** CV-SPRAY-PMP-A	287	52/19A
19B	**CV RECIRC FAN, HVH-1** HVH-1	511	52/19B
19C	**SERVICE WATER PUMP B** SW-PMP-B	832	52/19C
20A	**AUX FEEDWATER PUMP A** AFW-PMP-A	651	52/20A
20B	**SERVICE WATER PUMP A** SW-PMP-A	831	52/20B
20C	**CV RECIRC FAN, HVH-2** HVH-2	512	52/20C
21A	**FEED TO MCC-5 (NORM POWER) & MCC-16** MCC-5, MCC-16	1187	52/21A
21B	**CHARGING PUMP B** CHG-PMP-B	162B	52/21B
21C	**SAFETY INJECTION PUMP A** SI-PMP-A	237	52/21C
22A	**RESIDUAL HEAT REMOVAL PUMP A** RHR-PMP-A	214	52/22A
22B	**480V BUS E-1 SUPPLY TO SI PUMP B** 480V-E1, E2	891	52/22B
22C	**COMPONENT COOLING WATER PUMP B** CCW-PMP-B	205	52/22C

* Compartment 17A also contains two amp meters, two amp meter switches, one volt meter, one volt meter switch, two undervoltage relays, four overcurrent relays, and two auxiliary relays.

** Compartment 18A also contains eight run time meters, three degraded grid relays, one degraded grid trip signal, three test switches, and one degraded grid voltage switch.

Correct

9.0 480V-E2

480V-E2 POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: E-1/E-2 ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
23A	**CHARGING PUMP C** CHG-PMP-C	163B	52/23A
23B	**SAFETY INJECTION PUMP C** SI-PMP-C	239	52/23B
23C	**FEED TO MCC-6** MCC-6	1188	52/23C
24A	**SERVICE WATER PUMP C** SW-PMP-C	833	52/24A
24B	**CV RECIRC FAN, HVH-4** HVH-4	514	52/24B
24C	**AUX FEEDWATER PUMP B** AFW-PMP-B	655	52/24C
25A	**CV RECIRC FAN, HVH-3** HVH-3	513	52/25A
25B	**SERVICE WATER PUMP D (NORMAL SUPPLY)** SW-PMP-D	834B	52/25B
25C	**CV SPRAY PUMP B** CV-SPRAY-PMP-B	290	52/25C
26A	**FEED TO MCC-18** MCC-18	1189	52/26A
26B	**RESIDUAL HEAT REMOVAL PUMP B** RHR-PMP-B	216	52/26B
26C	**COMPONENT COOLING WATER PUMP C** CCW-PMP-C	209	52/26C
27A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
27B	**EMERGENCY DIESEL GENERATOR B TO 480V BUS E-2** 480V-E2	895	52/27B

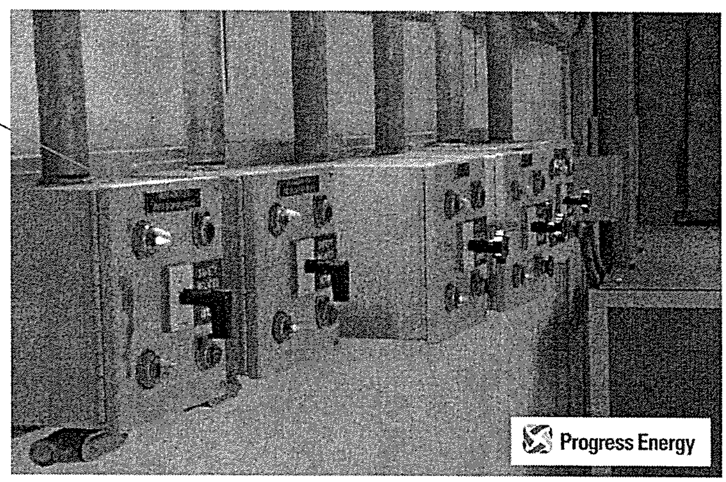
Incorrect

Service Water Pumps can be locally controlled from…

- Local Control Stations

- SW Pumps 'A' & 'B':
  - Behind Emergency Bus E-1 on the Battery Room wall

- SW Pumps 'C' & 'D':
  - Rod Control Room on the CV wall



SW Pump D ~~is~~ can be Powered from The DS Bus.

Incorrect

7.0 480V-DS

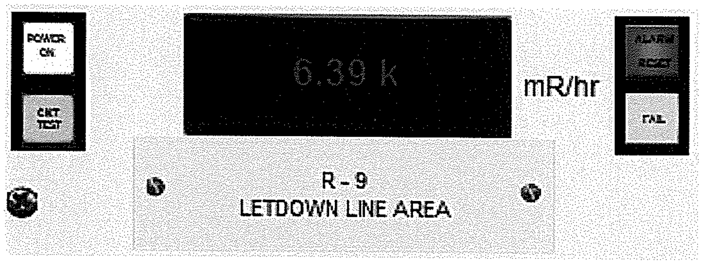
480V-DS POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15)LOCATION: 4160V SWITCHGEAR ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
32A	**FEED TO 480V BUS DS** 480V-DS	1015	52/32A
32B	**DEDICATED SHUTDOWN DIESEL GENERATOR TO 480V BUS DS (ALT POWER)** 480V-DS	1016	52/32B
33A	**CONTROL POWER TRANSFORMER (*)** CPT/480V-DS	N/A	N/A
33B	**SERVICE WATER PUMP D (ALT POWER)** SW-PMP-D	834C	52/33B
33C	**COMPONENT COOLING WATER PUMP A** CCW-PMP-A	201	52/33C
33D	**RESIDUAL HEAT REMOVAL PUMPS (ALT POWER)** RHR-PMP-A, B	1752	52/33D
34A	**POTENTIAL TRANSFORMER** PT/480V-DS	N/A	N/A
34B	**CHARGING PUMP A** CHG-PMP-A	161	52/34B
34C	**FEED TO MCC-5 (ALT POWER)** MCC-5	N/A	52/34C
34D	**FEED TO PP-51** PP-51	N/A	52/34D

* Compartment 33A also contains the Charging Pump A total run time meter and the Component Cooling Water Pump A total run time meter.

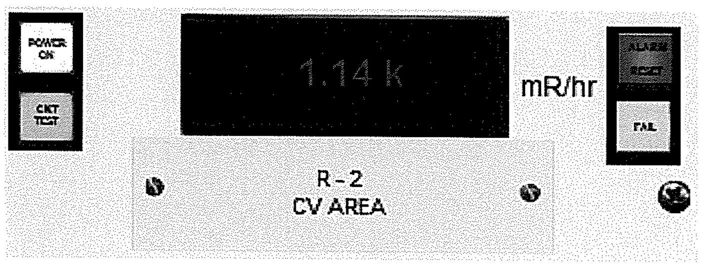
25. 076 AA1.04 001

The reactor is at 100% RTP. Which of the following would indicate failed fuel?

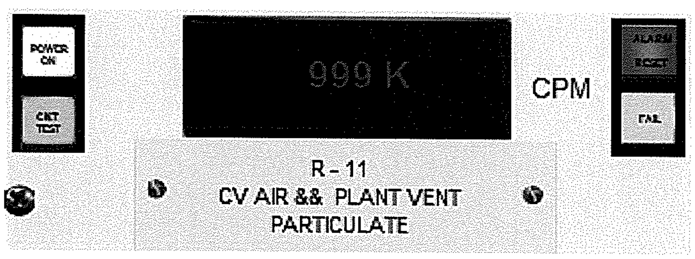
A.✓



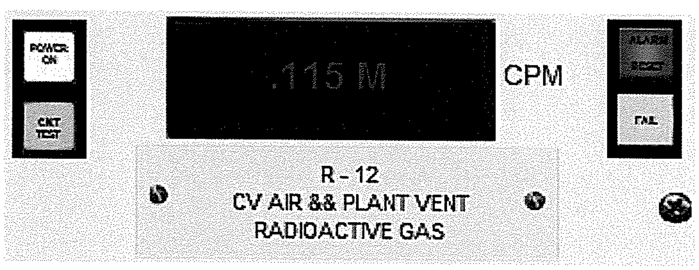
B.



C.



D.



The correct answer is A

A) Correct. R-9 is the correct rad monitor that would be used to indicate failed fuel.

B) Incorrect. Plausible if there was an RCS leak/LOCA. However, the stem says they are at 100% RTP.

C) Incorrect. Plausible if there was an RCS leak/LOCA. However, the stem says they are at 100% RTP.

D) Incorrect. Plausible if there was an RCS leak/LOCA. However, the stem says they are at 100% RTP.

Question: 25
Tier/Group: 1/2
K/A Importance Rating: RO 3.2 SRO 3.4
K/A: 076 High Reactor Coolant Activity
AA1: Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity:
AA1.04: Failed fuel-monitoring equipment
Reference(s): Sim/Plant design, AOP-005 Attachment 9
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 2 A9 of RMS lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.7 / 45.5 / 45.6)
Comments:

This question meets the K/A because the student has to determine which of the Rad Monitors they would monitor for failed fuel/high rcs activity.

ATTACHMENT 8, AREA MONITOR R-8 - DRUMMING STATION

1-5 These steps are standard instructions for performing a local area evacuation.

6 Although drumming operations are not normally performed at HBR, the ability to perform drumming operations still exists. Therefore, the possibility of drumming operations in progress is considered in this attachment. This step checks if drumming operations are in progress.

If drumming operations are not in progress, the RNO provides transition around steps securing drumming operations.

7 This step secures drumming operations until the cause of the abnormal radiation condition is determined and corrective actions have been completed.

8 Radiation Control personnel are contacted in response to all abnormal radiation conditions at the Plant. This step provides instruction to contact RC personnel and provides instruction on minimum radiological surveys to be performed.

9 This standard step provides transition back to the procedure body to address other Radiation Monitor alarms or to exit the procedure.

ATTACHMENT 9, AREA MONITOR R-9 - LETDOWN LINE AREA

1-5 These steps are standard instructions for performing a local area evacuation.

6 Radiation Control personnel are contacted in response to all abnormal radiation conditions at the Plant. This step provides instruction to contact RC personnel and provides instruction on minimum radiological surveys to be performed. The areas identified for survey were based on areas that could be adversely affected due to the letdown piping flowpath.

7 High Radiation in the letdown line in most cases are due to high activity levels in RCS. By Reducing the letdown flow, the total amount of radionuclides recirced throughout the auxiliary building are also reduced.

8 If letdown flow rate was reduced as a result of performance of Step 7, charging flow may need to be adjusted in order to control PZR level. This step provides instruction to control charging flow in order to maintain PZR level.

26. W/E08 EK1.2 001

Given the following plant conditions:

-A reactor trip has occurred

-The turbine failed to trip

-The MSIV's have failed to close

-The crew is in FRP-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK

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

IAW FRP-P.1, the crew will maintain __(1)__ . The crew will be required to perform a soak for __(2)__ hour(s).

A.✓ (1) between 80-90 gpm feed flow to each S/G
(2) 1

B. (1) between 80-90 gpm feed flow to each S/G
(2) 29

C. (1) total feed flow > 300 gpm
(2) 1

D. (1) total feed flow > 300 gpm
(2) 29

The correct answer is A

A) Correct. Because the MSIV's are stuck open, there are no S/G's that are intact. FRP-P.1 will have you maintain feed flow to each S/G between 80-90 gpm. FRP-P.1 requires a 1 hour soak once RCS temperature is stable.

B) Incorrect. Because the MSIV's are stuck open, there are no S/G's that are intact. FRP-P.1 will have you maintain feed flow to each S/G between 80-90 gpm. 29 hour soak time is incorrect. Plausible because this is the soak time from EPP-5, Natural Circulation Cooldown.

C) Incorrect. Total feed flow > 300 gpm is incorrect. Plausible because FRP-P.1 has you maintain total feed flow > 300 gpm if S/G's are intact but level is <8%. FRP-P.1 requires a 1 hour soak once RCS temperature is stable.

D) Incorrect. Total feed flow > 300 gpm is incorrect. Plausible because FRP-P.1 has you maintain total feed flow > 300 gpm if S/G's are intact but level is <8%. 29 hour soak time is incorrect. Plausible because this is the soak time from EPP-5, Natural Circulation Cooldown.

Question: 26
Tier/Group: 1/2
K/A Importance Rating: RO 3.4 SRO 4.0
K/A: WE/08 Pressurized Thermal Shock
EK1: Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock)
EK1.2: Normal, abnormal and emergency operating procedures associated with (Pressurized Thermal Shock).
References: Sim/Plant design, FRP-P.1
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of FRP-P.1 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.8 /41.10/45.3)

Meets the K/A because it asks the student what actions(Operational Implications) per FRP-P.1 will be taken given a set of parameters.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Check S/G Status - ANY INTACT**	Perform the following: a. Verify CLOSED all MSIVs AND MSIV BYPs. b. IF the MDAFW Pumps are NOT available, THEN maintain steam supply to the SDAFW Pump from at least one S/G. c. Verify CLOSED the following STEAM SHUTOFFs to the SDAFW PUMP: • V1-8A • V1-8B • V1-8C d. Maintain feed flow to each S/G between 80 gpm and 90 gpm. e. Go To Step 11.
* 7.	**Check Level In At Least One Intact S/G - GREATER THAN 8% [18%]**	Maintain total feed flow greater than 300 gpm OR 0.2×10^6 pph until level greater than 8% [18%] in at least one intact S/G. WHEN level in at least one intact S/G is greater than 8% [18%], THEN perform Step 8. Go To Step 9.
8.	**Control Feed Flow To Intact S/G(s) To Stop RCS Cooldown**	

CORRECT

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
41.	**Check Adequate RCS Depressurization:** • Check RCS Subcooling - GREATER THAN <u>OR</u> EQUAL TO 45°F [65°F] <u>AND</u> • RCS Pressure - GREATER THAN 200 PSIG	Go To Step 44.

****

<u>CAUTION</u>

The upper head region may void during RCS depressurization if RCPs are not running. This will result in a rapidly rising PZR level.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
42.	**Check Normal PZR Spray - AVAILABLE**	<u>IF</u> normal letdown is in service, <u>THEN</u> observe the <u>NOTE</u> prior to Step 33 and Go To Step 33. <u>IF</u> normal letdown is <u>NOT</u> in service, <u>THEN</u> Go To Step 30.
43.	**Go To Step 29**	
44.	**Check Cooldown Rate In RCS Cold Legs - GREATER THAN 100°F IN ANY 60 MINUTE PERIOD**	Go To Step 47.
45.	**Check RCS Temperature - HAS BEEN STABLE FOR ONE HOUR**	Perform the following: a. Do <u>NOT</u> cooldown the RCS. b. Do <u>NOT</u> raise RCS pressure. c. Perform actions of any other procedures in effect which do <u>NOT</u> cooldown the RCS <u>OR</u> raise RCS pressure. d. <u>WHEN</u> RCS temperature has been stable for one hour, <u>THEN</u> Go To Step 46.

incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
6.	**Check S/G Status - ANY INTACT**	Perform the following: a. Verify CLOSED all MSIVs AND MSIV BYPs. b. IF the MDAFW Pumps are NOT available, THEN maintain steam supply to the SDAFW Pump from at least one S/G. c. Verify CLOSED the following STEAM SHUTOFFs to the SDAFW PUMP: • V1-8A • V1-8B • V1-8C d. Maintain feed flow to each S/G between 80 gpm and 90 gpm. e. Go To Step 11.
* 7.	**Check Level In At Least One Intact S/G - GREATER THAN 8%** [18%]	Maintain total feed flow greater than 300 gpm OR 0.2×10^6 pph until level greater than 8% [18%] in at least one intact S/G. WHEN level in at least one intact S/G is greater than 8% [18%], THEN perform Step 8. Go To Step 9.
8.	**Control Feed Flow To Intact S/G(s) To Stop RCS Cooldown**	

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
40.	**Determine If RHR System Can Be Placed In Service:**	
	a. Check RCS temperature - LESS THAN 350°F	a. <u>WHEN</u> RCS temperature is less than 350°F, <u>THEN</u> Go To Step 40.b.
	b. Check RCS pressure - LESS THAN 375 PSIG	b. <u>WHEN</u> RCS pressure is less than 375 psig, <u>THEN</u> Go To Step 40.c.
	c. Place RHR system in service using Supplement I	
41.	**Continue RCS Cooldown To Cold Shutdown**	
42.	**Continue Cooldown Of Inactive Portions of RCS As Follows:**	
	a. Verify both CRDM Cooling Fans - RUNNING SINCE DURATION AT STEP 24 • HVH-5A • HVH-5B	a. Perform the following: 1) Maintain RCS temperature less than 212°F for 29 hours. 2) Go To Step 42.c.
	b. Cool upper head region using Both CRDM Cooling Fans • HVH-5A • HVH-5B	
	c. Cool S/G U-tubes by dumping steam from all S/Gs until the S/Gs have stopped steaming	
43.	**Check Cooldown Status - ALL REQUIREMENTS OF STEP 42 SATISFIED**	<u>WHEN</u> all requirements met, <u>THEN</u> observe <u>CAUTION</u> prior to Step 44 and Go To Step 44.

27. W/E10 EK2.2 001

Given the following plant conditions:

-The crew has experienced a Loss of All A/C Power

-They have transitioned to EPP-6, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL

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

IAW EPP-6, the maximum RCS temperature at which RHR can be placed in service is LESS THAN __(1)__. Per EPP-6, the maximum cooldown rate is LESS THAN __(2)__ in the last 60 minutes.

A.✓ (1) 350°F
(2) 100°F

B. (1) 250°F
(2) 100°F

C. (1) 350°F
(2) 25°F

D. (1) 250°F
(2) 25°F

The correct answer is A

A) Correct. IAW EPP-6, RHR will be placed in service when the RCS temperature is < 350°F. EPP-6 has you cooldown at a rate of < 100°F in the last 60 minutes.

B) Incorrect. 250°F is incorrect. Plausible because OP-201, RHR System, RHR system cannot be placed in service unless RCS temperature is <250°F. The cooldown rate is correct.

C) Incorrect. IAW EPP-6, RHR will be placed in service when the RCS temperature is < 350°F. 24°F is incorrect. Plausible because in EPP-5, NATURAL CIRCULATION COOLDOWN, your allowed cooldown rate is <25°F.

D) Incorrect. 250°F is incorrect. Plausible because OP-201, RHR System, RHR system cannot be placed in service unless RCS temperature is <250°F. 25°F is incorrect. Plausible because in EPP-5, NATURAL CIRCULATION COOLDOWN, your allowed cooldown rate is <25°F.

Question: 27
Tier/Group: 1/2
K/A Importance Rating: RO 3.6 SRO 3.9
K/A: WE/10 Natural Circulation with Steam Void in Vessel with/without RVLIS
EK2: Knowledge of the interrelations between the (Natural Circulation with Steam Void in Vessel with/without RVLIS) and the following:
EK2.2: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.
References: Sim/Plant design, EPP-6
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.7 / 45.7)

Meets the K/A because the students must have knowledge of when the RHR system can be placed in service in EPP-6, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Supplement K is available for optimizing Auxiliary Spray below.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
13.	**Continue RCS Cooldown And Depressurization As Follows:**	
	a. Maintain cooldown rate in RCS cold legs less than 100°F in the last 60 minute	
	b. Maintain RCS temperature and pressure within limits of curve 3.4, Reactor Coolant System pressure - temperature limitations for cooldown	
	c. Maintain RCS subcooling greater than 55°F	c. Stop depressurization <u>AND</u> establish subcooling.
	d. Check steam dump to Condenser - AVAILABLE	d. Dump steam using STEAM LINE PORVs. Go To Step 13.f.
	e. Dump steam to Condenser	
	f. Control feed flow to maintain S/G levels - BETWEEN 39% AND 50%	
	g. Control depressurization using auxiliary spray as follows:	g. Use one PZR PORV.
	1) Establish letdown using OP-301, Chemical and Volume Control System (CVCS)	
	2) Use CVC-311, AUX PZR SPRAY	

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*24.	**Determine If The SI System Should Be Removed From Service As Follows:**	
	a. Check RCS Temperature - LESS THAN 350°F	a. Continue the RCS cooldown. <u>WHEN</u> RCS temperature is less than 350°F, <u>THEN</u> perform Steps 24.b and 24.c. Go To Step 25
	b. Verify SI System status - MAXIMUM OF ONE SI PUMP CAPABLE OF INJECTING INTO THE RCS	
	c. Open the following breakers: • SI-865C, ACCUMULATOR C DISCHARGE (MCC-5, CMPT 9F) • SI-865A, ACCUMULATOR A DISCHARGE (MCC-5, CMPT 14F) • SI-865B, ACCUMULATOR B DISCHARGE (MCC-6, CMPT 10J)	
25.	**Determine If RHR System Can Be Placed In Service:**	
	a. Check RCS temperature - LESS THAN 350°F	a. <u>WHEN</u> RCS temperature is less than 350°F, <u>THEN</u> Go To Step 25.b.
	b. Check RCS pressure - LESS THAN 375 PSIG	b. <u>WHEN</u> RCS pressure is less than 375 psig, <u>THEN</u> Go To Step 25.c.
	c. Place RHR system in service using Supplement I	
26.	**Continue RCS Cooldown To Cold Shutdown**	

incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

****

CAUTION

Excessive steam dump using the steam line PORVs may initiate a high steam line ΔP SI.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
14.	**Initiate RCS Cooldown To Cold Shutdown As Follows:**	
	a. Maintain cooldown rate in RCS cold legs - LESS THAN 25°F IN THE LAST 60 MINUTE	
	b. Maintain RCS temperature and pressure - WITHIN LIMITS OF CURVE 3.4, REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITATIONS FOR COOLDOWN	
	c. Check steam dump to Condenser - AVAILABLE	c. Dump steam using STEAM LINE PORVs. Go To Step 14.e.
	d. Dump steam to Condenser	
	e. Control feed flow to maintain S/G levels - BETWEEN 39% AND 50%	
15.	**Check RCS Low Tavg Bistables - AT LEAST TWO ILLUMINATED**	WHEN at least two RCS Low Tavg bistables are ILLUMINATED, THEN Go To Step 16.

Incorrect

4.0 PREREQUISITES

4.1 Plant Electrical Distribution is in service IAW OP-603.

4.2 Instrument Air is available to HCV-758, FCV-605, and HCV-142 IAW OP-905.

4.3 The Radiation Monitoring system is in service IAW OP-920.

4.4 The Heating, Cooling, and Ventilating system is in service IAW OP-906.

4.5 Component Cooling Water System is in normal operation IAW OP-306.

4.6 The Pressurizer Relief Tank is available, IAW OP-103, to receive discharge from Residual Heat Removal System relief valves should the system be overpressurized.

4.7 The Residual Heat Removal Pump Pit Sump Pumps are operable to remove liquid in the event a Residual Heat Removal Pump seal should fail.

4.8 The Waste Holdup Tank level is below 85% in order to provide holdup capacity for 1,500 gallons of waste which may be received should a Residual Heat Removal Pump seal fail.

5.0 PRECAUTIONS AND LIMITATIONS

5.1 Reactor Coolant System temperature and pressure shall be less than 250•F and 375 psig before the Residual Heat Removal System is put in service, and the RHR system will be removed from service before RCS pressure and temperature are raised above these values.

5.2 To prevent boiling the CCW liquid contained in an RHR HX, CCW flow should **NOT** be isolated to an RHR HX when the temperature of the RHR System is greater than 200•F. (CR 95-00565)

5.3 This procedure has been reviewed per OPS-NGGC-1306 and has been determined to contain an R3 Reactivity Evolution when placing RHR in service for Core Cooling (Section 8.1.1).

28. 003 A1.09 001

Given the following plant conditions:

- The Plant is at 100% RTP
- APP-001-D2, RCP #1 SEAL LEAKOFF HI FLOW is in alarm
- RCP #1 Seal Leakoff Flows and Thermal Barrier D/Ps are as indicated **(SEE REFERENCE ON NEXT PAGE)**

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

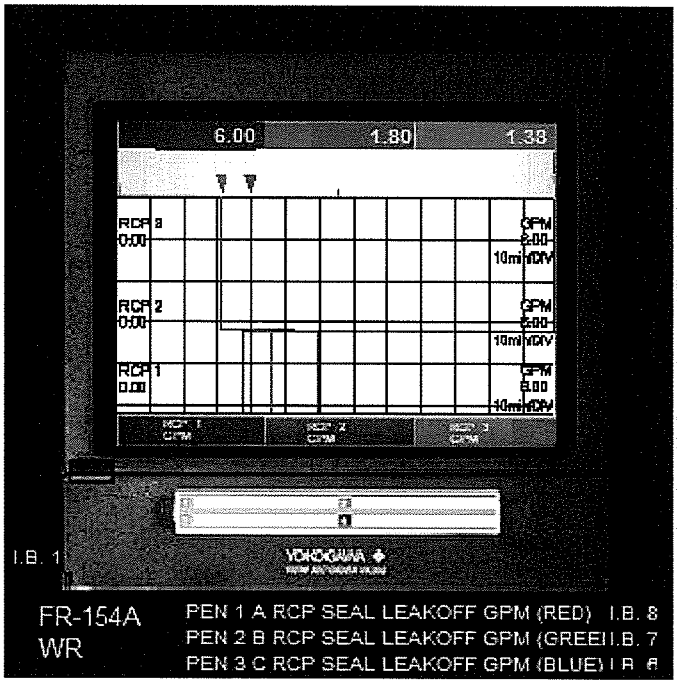
RCP A ___(1)___ seal is failed . To prevent damage to the RCP A Seal Stack, the RCP A SEAL LEAKOFF Valve, CVC-303A should be closed _____(2)_____ after tripping RCP A.

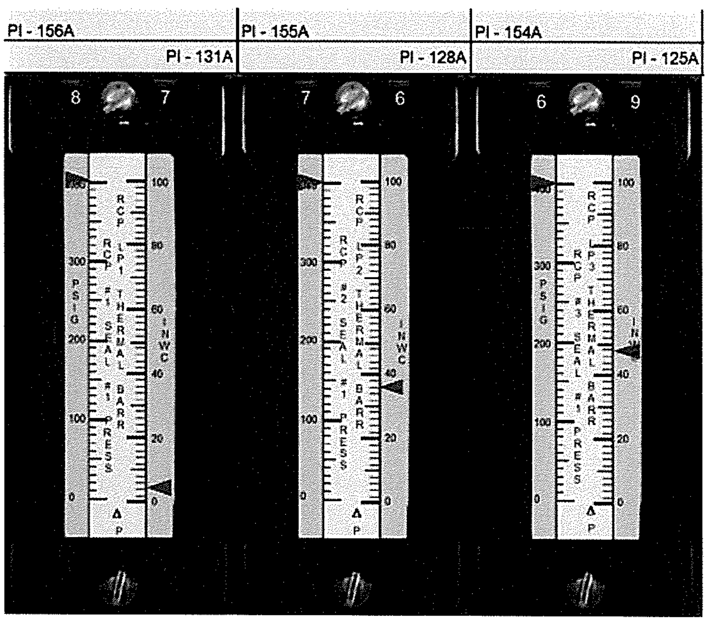
A. (1) #1
(2) immediately

B. (1) #2
(2) immediately

C✓ (1) #1
(2) between 3 and 5 minutes

D. (1) #2
(2) between 3 and 5 minutes


6.00
1.80
1.38
RCP 3
0.00
GPM
8.00
10min/DIV
RCP 2
0.00
GPM
8.00
10min/DIV
RCP 1
0.00
GPM
8.00
10min/DIV
YOKOGAWA
I.B. 1
FR-154A
WR
PEN 1 A RCP SEAL LEAKOFF GPM (RED) I.B. 8
PEN 2 B RCP SEAL LEAKOFF GPM (GREEI I.B. 7
PEN 3 C RCP SEAL LEAKOFF GPM (BLUE) I.B. 6


PI - 156A
PI - 131A
PI - 155A
PI - 128A
PI - 154A
PI - 125A
8
7
7
6
6
9
PSIG
IN WC
RCP 1 THERMAL BARR
RCP #1 SEAL #1 PRESS
RCP 2 THERMAL BARR
RCP #2 SEAL #1 PRESS
RCP 3 THERMAL BARR
RCP #3 SEAL #1 PRESS
ΔP

The correct answer is C.

A. Incorrect. Plausible because candidate could believe that this needs to be performed expeditiously, however AOP-018-BD states that the valve should not be closed until the pump is stopped, this is the basis for the 3 minute wait.

B. Incorrect. If #2 seal were failed the leakoff flow on the affected pump would bypass the flow indication and flow up through the #2 seal. Plausible because parameter diagnosis of seal failures are commonly misunderstood by initial license candidates. Second part is plausible because candidate could believe that this needs to be performed expeditiously, however AOP-018-BD states that the valve should not be closed until the pump is stopped, this is the basis for the 3 minute wait.

C. Correct. #1 Seal leakage rising on the affected pump with low flow on the unaffected pumps provides indication that #1 seal on the affected pump is failed. AOP-018-BD states that the valve should not be closed until the pump is stopped, this is the basis for the 3 minute wait. The step requires closure between 3 and 5 minutes.

D. Incorrect. If #2 seal were failed the leakoff flow on the affected pump would bypass the flow indication and flow up through the #2 seal. Plausible because parameter diagnosis of seal failures are commonly misunderstood by initial license candidates.

Question: 28
Tier/Group: 2/1
K/A Importance Rating: RO 2.8 SRO 2.8

K/A: 003 Reactor Coolant Pumps (RCPS)
A1: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including:
A1.09: Seal Flow and D/P

Reference(s): APP-001, AOP-018, AOP-018-BD
Proposed References to be provided to applicants during examination: None
Learning Objective: RCS009, AOP-018 2a
Question Source: New
Question History:
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 41.5, 10
Comments: This question meets the K/A because the candidate must be able to monitor RCP Seal parameters to determine action to prevent operating outside of RCP seal design.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 1 of 13)

* 1.	**Check Any RCP #1 Seal Leakoff Flow - GREATER THAN 5.7 GPM**	IF seal leakoff exceeds 5.7 gpm, THEN Go To Step 2. Go To Step 10.
2.	**Check Either Of The Following Conditions Exist:** • RCP #1 Seal Leakoff Flow On Unaffected RCP(s) - REDUCED OR • RCP Thermal Barrier ΔP On Affected RCP(s) - REDUCED	Perform the following: a. Perform cross-check of all RCP parameters to determine cause of indicated high leakoff flow. b. Observe The NOTE Prior To Step 2 and Go To The Main Body, Step 2 Of This Procedure

****

CAUTION

To prevent damage to the RCP Seal Stack, the affected RCP Seal Leakoff Isolation valve must be closed between 3 minutes and 5 minutes of stopping the RCP.

****

3.	**Check Plant Status - MODE 1 OR MODE 2**	Stop the affected RCP(s) Observe the CAUTION prior to Step 5 and Go To Step 5.
4.	**Perform The Following:** a. Trip the reactor b. Trip the affected RCP(s) c. Go To EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure.	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 2 of 13)

****

CAUTION

Restart of a RCP that has been stopped due to a seal malfunction, prior to the cause of the seal malfunction being determined and corrected, could cause catastrophic failure of all 3 RCP Seals.

****

5. **Check Time Elapsed Since Stopping The Affected RCP(s) - GREATER THAN 3 MINUTES**

WHEN at least 3 minutes have elapsed since tripping the affected RCP(s), THEN Go To Step 6.

6. **Close Seal Leakoff Valve(s) For Affected RCP(s):**

RCP	VALVE
A	CVC-303A
B	CVC-303B
C	CVC-303C

7. **Check SI - ACTUATED**

Go To Step 30

Step	Description
	SECTION A
1	This step determines if a RCP trip is required. Flow above 5.7 gpm requires an immediate RCP trip. The actual Westinghouse recommendations are: – Immediate trip > 8 gpm or > 6 gpm with rising seal temperatures – 8 hour shutdown between 6 and 8 gpm and stable seal temperatures Since RNP indication only goes to 6 gpm, the allowable range for RCP trip is not possible. It must be assumed that once flow has reached the end of the scale, that flow is greater than 8 gpm. Since the flow meter only goes to 6 gpm, the setpoint has been further reduced to 5.7 gpm in order to maintain a readable amount on the instrument scale. This reduction, in the conservative direction, is based solely on Management discretion.
2	Verification of a #1 Seal failure is made in this step after high seal flow of greater than 5.7 gpm is determined. If other (unaffected) RCP Seal leakoff flows lower, or affected RCP Thermal Barrier • P has lowered, a RCP #1 Seal failure is verified and the operator is transitioned to steps to remove the RCP from service. If no other indications of RCP #1 Seal failure are indicated, the RNO provides instruction to check alternate indications. If the only indication is high flow on a single pump, the most likely scenario is a failed instrument. Since this is a non-redundant indication, verification is made by check of diverse indication.
C3	The caution warns the operator of possible damage to the RCP seal stack should actions not be completed within the time listed in the caution. In this case Westinghouse Technical Bulletin ESBU-TB-93-01, Rev. 1, recommends that the isolation valve be closed within a time "window" of 3 to 5 minutes of stopping the RCP.
3	This step is reached in the event a malfunction of the Number 1 seal has been diagnosed which requires an RCP trip. It checks the plant at Mode 1 or Mode 2 to determine if a reactor trip is required for subsequent steps which will trip the affected RCP.

Step	Description
	SECTION A
4	This step performs the required series of steps to trip the affected RCP from a Mode 1 or Mode 2 condition. The reactor is tripped first to prevent •challenging a safety function•. The affected RCP is tripped to provide rapid response to a serious malfunction, then EOP-E-0, Reactor Trip or Safety Injection, is entered in parallel with this procedure. This is the common series of steps described in OMM-022 for tripping RCPs. No checks of power level interlocks are made for tripping the Reactor and RCP even though this is possible. These steps had been implemented in the past, but through experience on the simulator were found to be difficult to implement in a timely manner with an excessive burden of operator actions to perform during time challenging event (e.g. reduce reactor power rapidly, take manual control of S/Gs, trip affected RCP while maintaining unit on line during resulting shrink and swell. Not only does the 5 minute time limitation make this type of step series difficult, but conservative decision making philosophy warrants that the Reactor be tripped.
C5	This caution is derived from the Westinghouse step Do NOT restart the RCP until the cause of the seal malfunction has been determined and corrected. Since a "non-action" step is difficult to incorporate in procedures, the step has been converted into a caution that warns of the consequences should a pump that has been tripped, be restarted. The Westinghouse step and this caution are based on RNP experience in restarting a RCP following shutdown for a seal problem. Following restart of a RCP, catastrophic failure of all 3 Seals in the affected RCP occurred. Subsequent investigation showed that the RCP had a warped shaft which had caused the seal problem. Restart of the pump with the warped shaft destroyed the remaining seals. The caution has been placed after the RCP trip step and prior to the isolation step for two reasons: 1. It applies to all steps after stopping the RCP. 2. Although placing it in front of the step for tripping the RCP is possible, it would hinder action during a time critical portion of the procedure. The isolation portion of the procedure contains 3 minutes of hold point, allowing ample time to read the caution without hindering progress through the procedure.
5	Westinghouse requires closing #1 Seal Leakoff Valve if the RCP was tripped due to RCP #1 Seal failure. However, the valve should not be closed until the RCP has stopped rotating. This step acts as a hold point in the procedure until the RCP has been stopped for at least 3 minutes which assures that the pump will be stopped.

Step	Description **SECTION A**
6	This step closes the Seal Leakoff Valve for the affected RCP. The step is constructed in table format to ease selection of the applicable valve. Closure of the affected Seal Leakoff valve isolates the failed RCP #1 Seal and establishes the pressure boundary at the #2 Seal. This also prevents excessive back pressure from hindering operation of the intact seals and ingress of trash and debris from the failed component to the still intact RCP Seals via excessive back pressure.
7	This step checks for the actuation of Safety Injection, which would result in a Phase A. The Phase A signal isolates Instrument Air to the CV.
8	This step establishes Instrument Air to the CV to maintain the CVC-303A/B/C valve closed. This is a C/A step for the availability of an air supply.
9	This step provides transitional direction to steps at the end of the section for restoration of near normal conditions once the failed RCP Seal has been isolated.
10	This is a C/A step. This step is reached if #1 Seal Leakoff is <u>NOT</u> greater than 5.7 gpm. It is the first of a series of steps to diagnose low flow problems in the #1 Seal or high flow problems in the #2 Seal. Actual flow less than 0.8 gpm indicates a potential problem with the #1 Seal which could allow face contact, thus wiping the seal. If this is noted, subsequent steps will determine which seal has the problem.
N11	This note states the assumption made by the calculations in the subsequent step. The procedure is written from an "initial event" standpoint, meaning that if leakoff was already high on a #2 Seal, the calculation will not be valid. During normal operation #2 Seal Leakoff is negligible, falling in a range that does not manifest a level rise during a one hour period for all three RCPs. After the first event involving raised RCP #2 Seal leakage, it is expected that special instructions will be provided for alteration of this calculation or that a procedure change will be affected. This calculation also does not apply for a sudden catastrophic failure of the number 2 seal in range greater than number 1 leakoff.
11	This step determines the actual #1 Seal Leakoff flow for the affected RCP. As discussed earlier, if the #2 Seal opens, indicated flow will lower while actual flow will either remain the same or rise slightly for the expected slow failure. Note that a rapid catastrophic failure of a number 2 seal at a value greater than number 1 seal leakoff could cause backflow from the unaffected pumps to the affected pump. See discussion at N11.
	The first part of the step directs completion of an attachment to determine #2 Seal Leakoff flow. This calculation was included in an attachment so that it could be performed locally as a discrete function if desired.

ALARM

RCP #1 SEAL LEAKOFF HI FLOW *** WILL REFLASH ***

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Failure of RCP Number 1 Seal

OBSERVATIONS

1. Labyrinth Seal ΔP (PI-125A, PI-128A, PI-131A)
2. RCP Seal Leakoff Temperatures **AND** flows (FR-154, RCP Temperature Recorder, and Computer)
3. RCP Number 1 Seal ΔP (PI-154A, PI-155A, PI-156A)
4. ERFIS Group Display "RCP LOG"

ACTIONS	CK (✓)
1. **IF** failure of a RCP Number 1 Seal has occurred, **THEN REFER TO** AOP-018.	_____

DEVICE/SETPOINTS

1. FC-154A, FC-155A, FC-156A / 5 gpm

POSSIBLE PLANT EFFECTS

1. Loss of RCS inventory

REFERENCES

1. ITS LCO 3.4.4, LCO 3.4.5, LCO 3.4.6 and LCO 3.4.13
2. AOP-018, Reactor Coolant Pump Abnormal Condition
3. CWD B-190628, Sheet 477, Cables K, L, and M

29. 003 K4.04 001

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

When Seal Injection is unavailable, CCW flow to the ___(1)___ ensures adequate cooling of the Reactor Coolant Pump seals. This cooling flow path remains in service until isolated by a Phase ___(2)___ actuation.

A. (1) Thermal Barrier
   (2) "A"

B.✓ (1) Thermal Barrier
   (2) "B"

C. (1) Seal Water HX
   (2) "A"

D. (1) Seal Water HX
   (2) "B"

The correct answer is B.

A. Incorrect. CCW flow is isolated to the thermal barrier when containment pressure exceeds 10 psig generating a Phase B actuation. Plausible because some other valves associated with CCW, such as CC-739, isolate on Phase A actuation.

B. Correct. Normally RCP seal cooling is provided by seal injection flow. When seal injection flow is unavailable water flows into the seals through the thermal barrier and the RCS water is cooled by CCW. CCW flow is isolated to the thermal barrier when containment pressure exceeds 10 psig generating a Phase B actuation.

C. Incorrect. Normally RCP seal cooling is provided by seal injection flow. When seal injection flow is unavailable water flows into the seals through the thermal barrier and the RCS water is cooled by CCW. Plausible because the heat exchanger does cool seal water however its function is to cool the common seal return flow prior to entering the VCT. CCW flow is isolated to the thermal barrier when containment pressure exceeds 10 psig generating a Phase B actuation. Plausible because some other valves associated with CCW, such as CC-739, isolate on Phase A actuation.

D. Incorrect. Normally RCP seal cooling is provided by seal injection flow. When seal injection flow is unavailable water flows into the seals through the thermal barrier and the RCS water is cooled by CCW. Plausible because the heat exchanger does cool seal water however its function is to cool the common seal return flow prior to entering the VCT.

Question: 29
Tier/Group: 2/1
K/A Importance Rating: RO 2.8 SRO 3.1

K/A: 003 Reactor Coolant Pumps (RCPS)
K4: Knowledge RCPS design feature(s) and/or interlock(s) which provide for the following:
K4.04: Adequate cooling of RCP motor and seals

Reference(s): ST-013 CCW, OP-101
Proposed References to be provided to applicants during examination: None
Learning Objective: RCS010
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.3, 7
Comments: This question meets the K/A because the candidate needs to know design features that provide cooling to the seals and interlocks that affect the design feature.

5.2 Reactor Coolant Pumps

5.2.1 General

1. Prior to RCP operation, the RCP standpipe level should be in the normal band to ensure that the No. 3 Seal does **NOT** operate dry.

2. Maximum allowable RCP vibration at the installed detector location is 5 mils at the frame or 20 mils at the shaft. AOP-018 provides guidance for operating an RCP with greater than 3 mils vibration at the frame or greater than 15 mils vibration at the shaft.

3. To avoid an overpressure transient, an RCP shall **NOT** be started when the RCS is water solid and any steam generator temperature is 50• F greater than the temperature of the RCS (Reference: ITS LCO 3.4.6).

4. Never start an RCP unless the Oil Lift Pump has been delivering oil to the Upper Thrust Shoes for at least 2 minutes. The Oil Lift Pump shall run for at least 50 seconds after the RCP is started. The Oil Lift Pump indicating lights for motor operation and oil pressure should be monitored.

5. An RCP should **NOT** be operated beyond the low and high level alarms in the upper and lower oil pots.

6. Seal Injection Water should be flowing at all times when the RCS is filled and pressurized. The minimum seal injection flow rate to each pump is 6 gpm.

7. The RCPs may be operated without seal water provided either of the following criteria is met:

   a. Reactor Coolant temperature is less than 150• F.

   b. RCP seal leakage rate is 5 gpm or less and at least 25 gpm of Component Cooling Water at an inlet temperature less than 115• F is flowing through the Thermal Barrier Cooling Coil.

CC-730, Bearing Outlet Isolation - MOV

(CWD-B-190628 Sh00233)

Valve CC-730 is operated by a two position (OPEN/CLOSE), spring return to center, switch located on the RTGB. The valve also automatically closes on a "P" signal providing containment isolation of this potential leak path. The valve is located outside containment downstream of CV Penetration P-19 in the Auxiliary Building pipe alley. The valve is powered from MCC-6. Valve position indication is provided on the RTGB at the control switch and as part of the Containment Phase B isolation indications.

FCV-626, Thermal Barrier Outlet Isolation Flow Control - MOV

(CWD- B-190628 Sh00234)

Valve FCV-626 is operated by a two position (OPEN/CLOSE), spring return to center, switch located on the RTGB. The valve will automatically close on a high flow (100 gpm) as monitored by FIC-626. The valve also automatically closes on a "P" signal providing containment isolation of this potential leak path. The valve is located outside containment downstream of CV Penetration P-20 in the Auxiliary Building pipe alley. The valve is powered from MCC-6. Valve position indication is provided on the RTGB at the control switch and as part of the Containment Phase B isolation indications.

At one time this valve would close when power was lost to instrument Bus #4 due to power loss to a closure relay. To correct this problem, a new time delay relay has be installed in place of the closure relay. This new time delay relay is supplied with electrical power from Aux. Panel GC which is ultimately supplied from the "B" Station Safety Related battery.

The new relay has a short time delay feature (3 seconds) and is powered from Aux. Panel GC. This will prevent FCV-626 closure on a loss of electrical power to IB #4, MCC-6, Emergency Bus #2 or a Loss of off-site power event. This will also prevent closure of FCV-626 in the event of a spurious high flow condition. (EC 76896)

CC-735, Thermal Barrier Outlet Isolation - MOV

(CWD-B-190628 Sh00230)

Valve CC-735 is operated from the two position (OPEN/CLOSED), spring return to the center, switch located on the RTGB. The valve is located outside containment downstream of FCV-626 in the Auxiliary Building pipe alley. The motor operator for the valve is powered from MCC-5. Upon receipt of a "P" signal, the valve will close.

CC-739, Excess Letdown HX. Outlet Isolation Valve, Air Operated Valve

(CWD-B-190628 Sh00229)

Valve CC-739 is operated from the RTGB using a two position (OPEN/CLOSED), spring return to center, switch. This valve is located in the Auxiliary Building pipe alley and provides CV isolation downstream of CV penetration P-22. CC-739 is an Air Operated Valve that receives operating air from the instrument air system through 125V DC solenoid valves. The solenoid valves receive power from the 125V DC Auxiliary Panel GC CKT#29. A safeguards actuation signal, "T" signal, will de-energize the solenoid valves causing CC-739 to close. Valve position indication is provided at the RTGB control switch and at the Containment Phase "A" Isolation indication on the vertical section of the RTGB.

RCV-609, Component Cooling Surge Tank Vent Isolation Valve, Air Operated Valve

(CWD-B-190628 Sh00204)

RCV-609 is currently gagged open to ensure CCW System does not over pressurize. The control switch for this valve is located on the RTGB. It is a two position (OPEN/CLOSED) spring return to center switch. The solenoid valves that control the air to the valve receive power from 125V Auxiliary DC Panel CKT#4. When Radiation Monitor RE-17 reaches its setpoint, the solenoid valves are deenergized to remove air pressure to RCV-609. When the activity in the CCW surge tank vent line is reduced to below the setpoint of RE-17, the solenoid valves for RCV-609 energize, opening the air supply to the air operator, and RCV-609 then strokes open. (Original design)

CC-832, CCW Makeup From Primary Water, MOV

(CWD-B-190628 Sh00203)

Valve CC-832 is operated from the RTGB using a two position (OPEN/CLOSED) switch. To open the valve the switch must be held in the OPEN position until the valve has reached its full stroke. To close the valve the switch must be held in the CLOSED position until the valve has reached the full closed position. It has throttle capability and will allow primary

30. 004 K3.08 001

Given the following plant conditions:

- The plant is operating at 100% RTP.
- Seal Injection flow to each RCP is 8.5 gpm
- "C" Charging Pump is currently running in Manual at minimum speed.
- "A" Charging Pump is running in MANUAL at 35% demand during the performance of OP-301-1, Section 8.4.7, Charging Pump Break-In After Maintenance.
- CVC-283C, Charging Pump "A" Discharge Relief, has lifted and has not reseated.

Which ONE (1) of the following identifies the impact on Seal Injection flow(s) AND how this malfunction would be addressed by AOP-018, Reactor Coolant Pump Abnormal Conditions?

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

Seal Injection flows will lower to ___(1)___ and AOP-018 directs the operators to ___(2)___.

A.✓ (1) ZERO flow
(2) Isolate letdown, secure all Charging Pumps, manually isolate "A" Charging pump.

B. (1) MINIMUM flow (~ 6 gpm each)
(2) Isolate letdown, secure all Charging Pumps, manually isolate "A" Charging pump.

C. (1) ZERO flow
(2) Stop "A" Charging Pump and adjust the speed of "C" Charging Pump to restore normal seal injection flows.

D. (1) MINIMUM flow (~ 6 gpm each)
(2) Stop "A" Charging Pump and adjust the speed of "C" Charging Pump to restore normal seal injection flows.

sw
The correct answer is A

A. Correct. The seal injection flows will lower to zero due to all charging flow being diverted through CVC-283C to the drain header. AOP-018 will direct the Operators to isolate letdown, secure all charging pumps, manually isolate "A" Charging pump. Ultimately the Operators would start an available Charging Pump, restore letdown and establish normal seal injection.

B. Incorrect. See above discussion. Plausible if the candidate incorrectly thinks that CVC-283C lifting will only affect "A" Charging Pump discharge flow or if the candidate incorrectly thinks that a check valve would prevent flow from "C" Charging Pump to CVC-283C.

C. Incorrect. First part of distractor is correct. The second half is incorrect but plausible if the candidate incorrectly thinks that CVC-283C lifting will only affect flow from "A" and "C" Charging pump as long as "A" Charging Pump is running and a check valve is in the "A" Charging pump discharge line.

D. Incorrect. Plausible if the candidate incorrectly thinks that CVC-283C lifting will only affect "A" Charging Pump discharge flow or if the candidate incorrectly thinks that a check valve would prevent flow from "C" Charging Pump to CVC-283C.

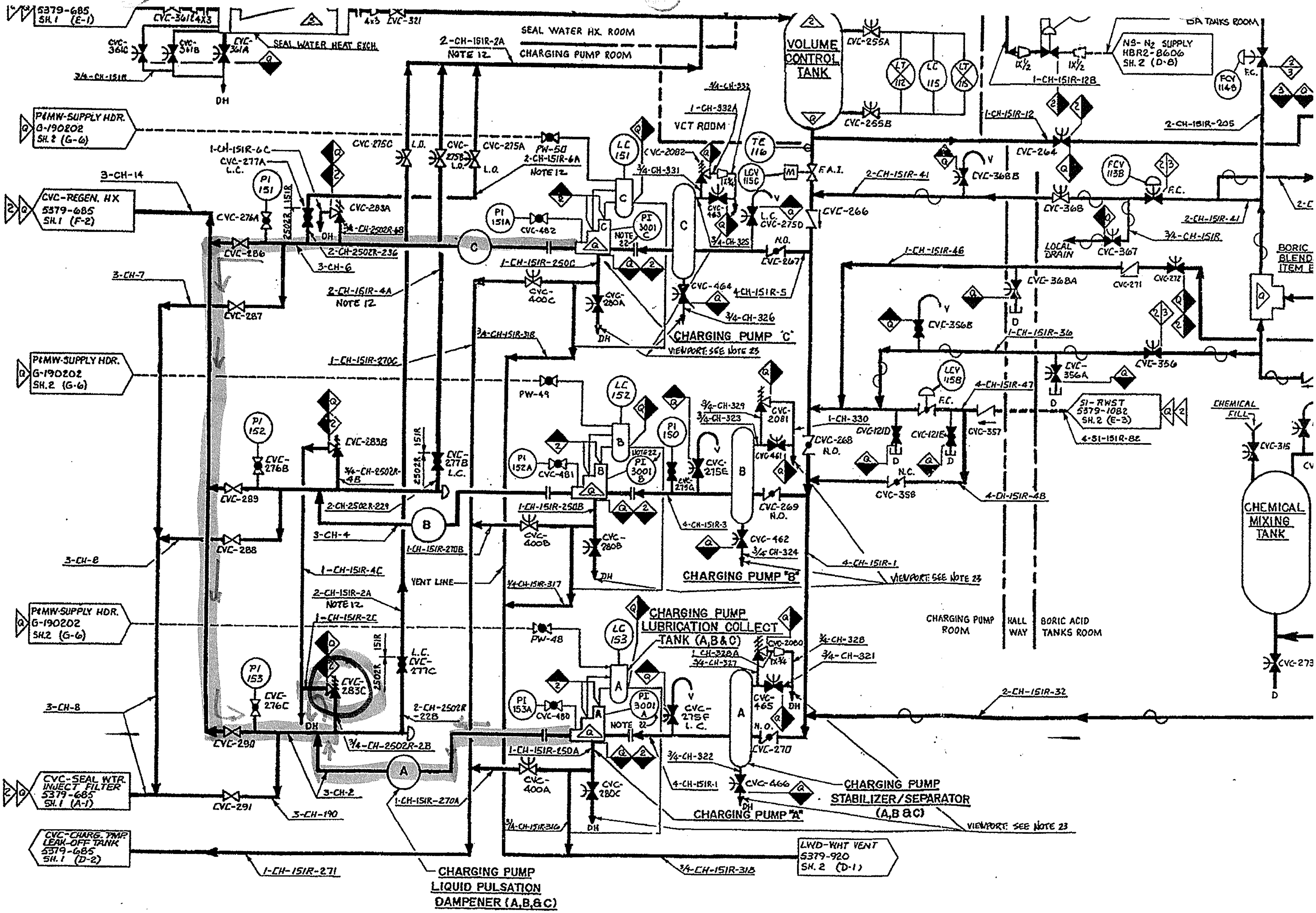
Question: 30
Tier/Group: 2/1
K/A Importance Rating: RO 3.6 SRO 3.8

K/A: 004 Chemcal and Volume Control System(CVCS)
K3: Knowledge of the effect that a loss or malfunction of the CVCS will have on the following:
K3.08: RCP seal injection

Reference(s): Sim/Plant design, System Description, AOP-018, Drawing 5379-00685, Sheet 2
Proposed References to be provided to applicants during examination: None
Learning Objective: CVCS009
Question Source: RNP Bank
Question History: NRC 11-1
Question Cognitive Level: H
10 CFR Part 55 Content: 41.7
Comments:

K/A match because candidate must understand how a charging pump discharge relief valve failure will impact RCP seal injection.

This is OE from RNP.



5379-685 SH. 1 (E-1)
SEAL WATER HEAT EXCH.
SEAL WATER HX. ROOM
CHARGING PUMP ROOM
2-CH-161R-2A NOTE 12
VOLUME CONTROL TANK
NS-N2 SUPPLY HBR2-8606 SH. 2 (D-8)
P&MW-SUPPLY HDR. G-190202 SH. 2 (G-6)
VCT ROOM
CVC-REGEN. HX 5379-685 SH. 1 (F-2)
3-CH-14
3-CH-7
3-CH-6
3-CH-4
3-CH-8
3-CH-2
3-CH-190
CHARGING PUMP 'C'
VIEWPORT: SEE NOTE 23
CHARGING PUMP "B"
CHARGING PUMP "A"
CHARGING PUMP LUBRICATION COLLECT TANK (A,B&C)
CHARGING PUMP STABILIZER/SEPARATOR (A,B&C)
CHARGING PUMP LIQUID PULSATION DAMPENER (A,B,&C)
CHARGING PUMP ROOM
HALL WAY
BORIC ACID TANKS ROOM
CHEMICAL MIXING TANK
CHEMICAL FILL
SI-RWST 5379-1082 SH. 2 (E-3)
CVC-SEAL WTR. INJECT FILTER 5379-685 SH. 1 (A-1)
CVC-CHARG. PMP. LEAK-OFF TANK 5379-685 SH. 1 (D-2)
LWD-WHT VENT 5379-920 SH. 2 (D-1)
1-CH-151R-271
3/4-CH-151R-318
YENT LINE
LOCAL DRAIN
CVC-283C

Purpose and Entry Conditions

(Page 1 of 1)

1. **PURPOSE**

   This procedure provides instructions to protect the Reactor Coolant Pumps after a component failure or support system malfunction.

2. **ENTRY CONDITIONS**

   This procedure is entered for any of the following events:

   - Loss of Seal Injection flow to any RCP
   - High Vibration on an RCP during steady state conditions (Confirmed valid IAW APP-001-B5)
   - Indication of any RCP Seal malfunction
     - RCP #1 Seal leakoff flow greater than 5 gpm
     - RCP #1 Seal leakoff flow less than 1 gpm
     - RCP #2 Seal leakoff flow greater than 0.5 gpm
     - Indication of RCP #3 Seal malfunction (repeat low standpipe level at normal pressure)

   - As directed by AOP-016, Excessive Primary Plant Leakage

- END -

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Make PA Announcement For Procedure Entry.**	

NOTE

- The RCP malfunctions in the Table below are listed in order of priority.
- This procedure is NOT applicable during implementation of EPP-1, Loss Of All AC OR any of its recovery procedures.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
2.	**Evaluate Plant Conditions AND Go To The Appropriate Section For RCP Malfunction Not Yet Addressed:**	Return to procedure and step in affect.

MALFUNCTION	SECTION
Reactor Coolant Pump Seal Failure	Section A
High Reactor Coolant Pump Vibration	Section B
Loss of Seal Injection	Section C

- END -

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 1 of 16)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 1.	**Check APP-001-D1, RCP THERM BAR COOL WTR LO FLOW alarm - ILLUMINATED**	IF APP-001-D1 ILLUMINATES, THEN Go To Step 2. Observe the NOTE prior to Step 11 and Go To Step 11.
2.	**Check Plant Status - MODE 1 OR MODE 2**	Stop the affected RCP(s) Observe the CAUTION prior to Step 4 and Go To Step 4.
3.	**Perform The Following:** a. Trip the reactor b. Trip the affected RCP(s) c. Go To EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure	

****

CAUTION

IF more than 15 minutes elapses without RCP Seal Cooling, THEN Seal Cooling (CCW AND Seal Injection) MUST be isolated before starting CCW OR Charging or Seal Damage could occur.

****

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
* 4.	**Check Elapsed Time Since All RCP Seal Cooling Was Lost - GREATER THAN 15 MINUTES**	IF RCP Seal Cooling is NOT OR can NOT be restored in less than 15 minutes, THEN Go To Step 5. Go To Step 10.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 6 of 16)

NOTE

- A rupture is a leak of sufficient magnitude to require stopping the Charging Pumps or reduces Charging Pump Discharge Pressure to less than RCS Pressure.
- Charging System piping is any piping where a leak prevents the Charging Pumps from delivering flow to the Charging Line OR Seal Injection Line.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**Determine If A Charging Pump Can Be Started:**	
	a. Check Charging System Piping - RUPTURED	a. Go To Step 12
	b. Go To Step 22.	
12.	**Check SI - INITIATED**	Go To Step 14
13.	**RESET SI**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

18.	**Check Seal Injection To RCPs:** • ANY Seal Injection flow - LESS THAN 6 GPM AND • ANY Thermal Barrier ΔP - LESS THAN 5 inches	Go To Step 47.
19.	**Check PI-121, CHARGING PUMPS DISCH PRESS Indicator - LESS THAN RCS PRESSURE**	Go To Step 41.
20.	**Check APP-001-B6, LP LTDN LN HI TEMP Alarm - EXTINGUISHED**	Close LCV-460 A&B, LTDN LINE STOP Valves.

NOTE

- A rupture is a leak of sufficient magnitude to require stopping the Charging Pumps or reduces Charging Pump Discharge Pressure to less than RCS Pressure.
- Charging System piping is any piping where a leak prevents the Charging Pumps from delivering flow to the Charging Line OR Seal Injection Line.

21.	**Check Charging System Piping - RUPTURED**	Go To Step 45.
22.	**Verify LCV-460 A&B, LTDN LINE STOP Valves - CLOSED**	
23.	**Stop The Running Charging Pump(s)**	
24.	**Momentarily Place RCS MAKEUP SYSTEM Switch To STOP**	
25.	**Dispatch An Operator To Locate And Isolate The Charging System Leak**	

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 9 of 16)

NOTE

"Charging Line" in the subsequent step is defined as the Charging Line downstream of HCV-121, CHARGING FLOW valve.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
*26.	**Check Location Of Break - IN CHARGING LINE**	IF break is found in the Charging Line, THEN Observe the NOTE prior to Step 27 and Go To Step 27. Go To Step 33.

NOTE

Throttling any RCP SEAL WATER FLOW CONTROL VALVE with Charging isolated will only shift flow to the other RCPs AND raise Charging Pump discharge pressure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
27.	**Perform The Following:** a. Close HIC-121, CHARGING FLOW b. Fully Open the RCP SEAL WATER FLOW CONTROL VALVEs • CVC-297A • CVC-297B • CVC-297C	
28.	**Check RCP SEAL WATER FLOW CONTROL VALVEs - FULLY OPEN**	WHEN the RCP SEAL WATER FLOW CONTROL VALVEs are fully open, THEN Go To Step 29.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION C

LOSS OF SEAL INJECTION

(Page 11 of 16)

33.	**Perform The Following:**	
	a. Check leak location - IDENTIFIED	a. Go to ONE of the below steps: • IF the leak can NOT be readily identified, THEN assume the leak is unisolable AND Go To Step 33.b OR • IF the leak is identified downstream of HCV-121, CHARGING FLOW, THEN Go To Step 27. OR • IF the leak is identified anywhere but downstream of HCV-121, CHARGING FLOW, THEN Go To Step 33.b.
	b. Check Charging System Leak - ISOLABLE	b. Perform the following: 1) Notify Operations Staff and Engineering to obtain further guidance. 2) Implement ITS 3.4.17. 3) Go To Step 50.
34.	**Check Charging System Leak - ISOLATED**	When the charging system leak is isolated, THEN Observe the NOTE prior to Step 35 and Go To Step 35.

31. 004 K6.26 001

Given the following plant conditions:

-The RCS is on RHR and **SOLID**

-RCS pressure is 340 psig

-PC-145, PRESSURE, in AUTO

-HIC-142, PURIFICATION FLOW, controller setting is at 55% demand

Subsequently, an RCS leak occurs

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

PC-145 controller output ____(1)____ and PCV-145 throttles ___(2)___.

A. (1) lowers
(2) closed

B. (1) lowers
(2) open

C✓ (1) rises
(2) closed

D. (1) rises
(2) open

The correct answer is C

A. Incorrect. PCV-145 controller output rises. Plausible if the candidate believes that PC-145 is a direct acting controller.

B. Incorrect. PCV-145 controller output rises. Plausible if the candidate believes that PC-145 is a direct acting controller. To balance the inventory losses from the RCS, PCV-145 must throttle closed. Plausible if a candidate believes that pressure is sensed downstream of PCV-145.

C. Correct. Pressure upstream of the PCV-145 will lower as inventory from the RCS is lost. PCV-145 must throttle closed and lower letdown flow to raise pressure back to setpoint. PC-145 is a reverse acting controller. 0% output correlates to PCV-145 being fully open and 100% output correlates with PCV-145 being fully closed. Therefore, to throttle PCV-145 closed PC-145 controller output must rise.

D. Incorrect. To balance the inventory losses from the RCS, PCV-145 must throttle closed. Plausible if a candidate believes that pressure is sensed downstream of PCV-145.

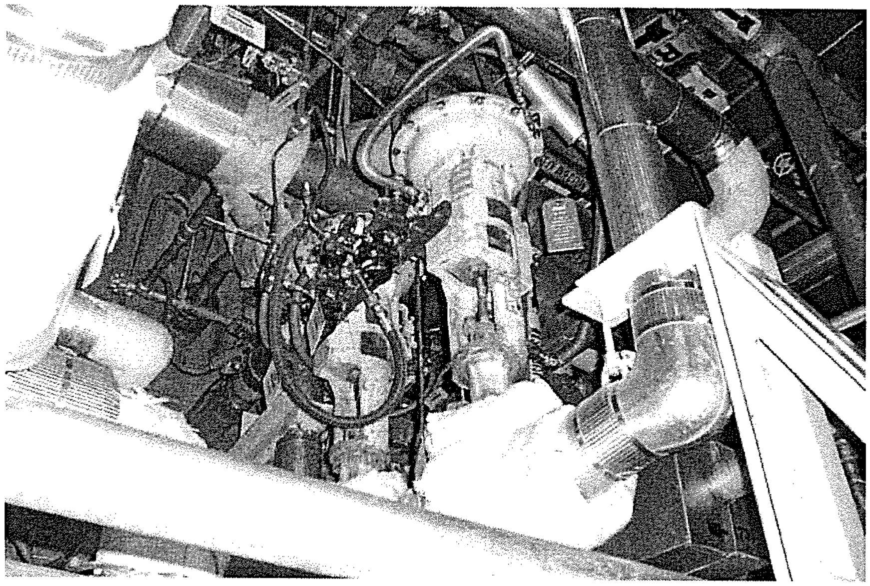
Question: 31
Tier/Group: 2/1
K/A Importance Rating: RO 3.8 SRO 4.1

K/A: 004 Chemical and Volume Control System (CVCS)
K6: Knowledge of the effect of a loss or malfunction on the following CVCS components:
K6.26: Methods of pressure control of solid plant (PZR relief and water inventory)

Reference(s): ST-021 CVCS
Proposed References to be provided to applicants during examination: None
Learning Objective: CVCS007d
Question Source: New
Question History:
Question Cognitive Level: H
10 CFR Part 55 Content: 41.7
Comments: K/A match because candidate must understand how PC-145 and PCV-145 work to compensate for a loss of water inventory.

or when removing letdown from service. The cross connect from the RHR System, HCV-142, taps in just downstream of CVC-204B.

Figure 45 CVC-204A & B in PA



Low Pressure Letdown Valve (PCV-145) and Low Pressure Letdown Relief Valve (CVC-209)

This valve is controlled from the RTGB in either automatic or manual to keep the water in the piping downstream of the orifices to PCV-145 from flashing to steam. During normal plant operation this valve is adjusted to maintain approximately 300 psig upstream of the valve. For plant heatups and cooldowns the valve is adjusted to maintain the proper system pressure for evolutions in progress; i.e. running RCPs, etc. PCV-145 is air operated and fails open.

RCS leak causes pressure to lower, so valve must close

The pressure signal used to operate this valve in automatic comes from PT-145, located upstream of the valve. PT-145 drives pressure indicator PI-145 located on the RTGB. The controller for PCV-145, located on the RTGB, consists of a dial potentiometer for automatic setpoint control, pushbuttons for manual control and a controller output demand indication. Opening the valve results in a lower backpressure on PT-145 and closing the valve results in a higher backpressure. PT-145 alarms APP-001-D6, LP LTDN LN HI PRESS, at 400 psig.

Low Pressure Letdown Relief Valve, CVC-209, is located just downstream of PCV-145. CVC-209 has a setpoint of 200 psig and relieves to the VCT.

Figure 46 CVCS Letdown Pressure Controller PC 145 & PCV 145

Output must rise to close valve

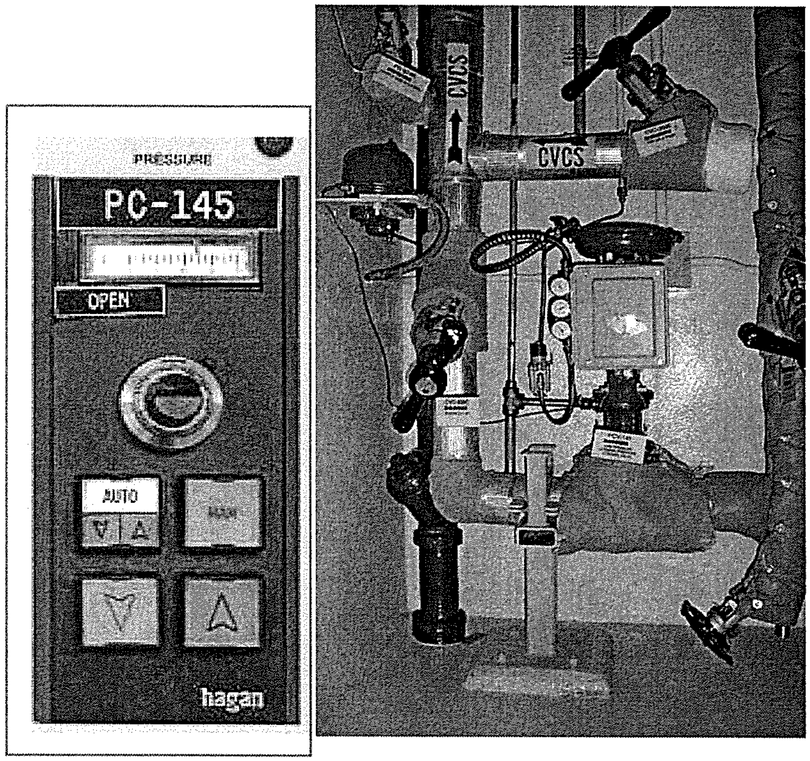
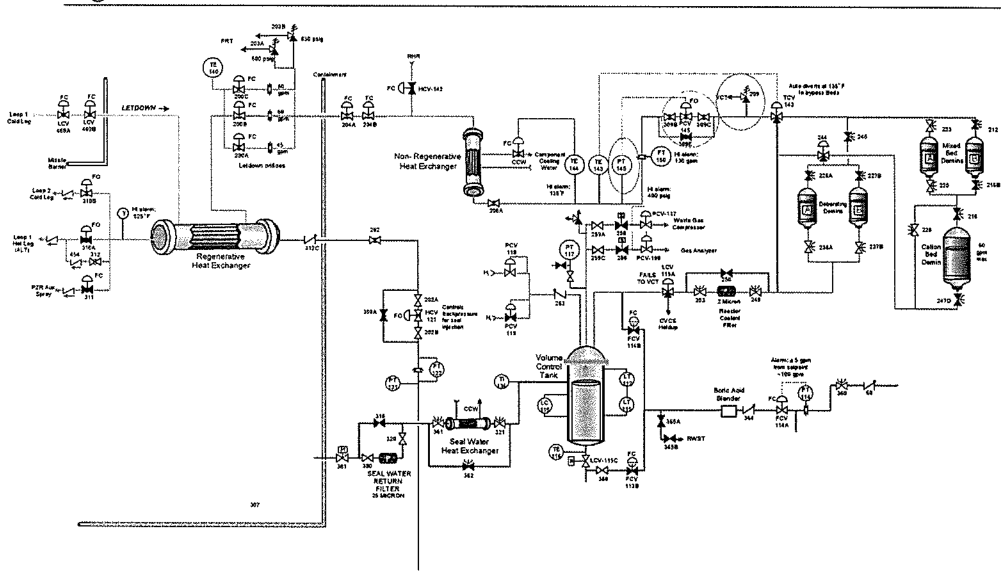


Figure 47 Letdown Pressure Control



32. 005 A2.04 001

Given the following plant conditions:

- The plant is in Mode 5.
- The air supply line to HCV-758, RHR HX OUTLET FLOW TO COLD LEGS, breaks, causing a complete loss of Instrument Air to the valve.

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

Based off the conditions above, total RHR flow initially ___(1)___. IAW AOP-020, Loss of Residual Heat Removal (Shutdown Cooling), the crew will **MANUALLY** throttle ____(2)____.

A. (1) rises
   (2) FCV-605, RHR HX BYPASS closed

B. (1) rises
   (2) RHR-764, HCV-758 BYPASS closed

C. (1) lowers
   (2) FCV-605, RHR HX BYPASS open

D✓ (1) lowers
   (2) RHR-764, HCV-758 BYPASS open

The correct answer is D.

A. Incorrect. HCV-758 fails closed on a loss of air which causes total RHR flow to lower initially. Plausible assuming the valve failed closed. Throttling FCV-605 would have no mitigating effect since HCV-758 failed closed vice open. Plausible if HCV-758 failed open, since this would lower total flow back to setpoint.

B. Incorrect. HCV-758 fails closed on a loss of air which causes total RHR flow to lower initially. Plausible assuming the valve failed closed.

C. Incorrect. Opening FCV-605 will return total flow to normal value, however it would not stabilize RCS temperature -- an RCS heatup would continue. Plausible if RHR-764 was open to begin with, however RHR-764 was closed as part of the initial lineup for core cooling. RHR-764 is normally full open when aligned for injection.

D. Correct. HCV-758 fails closed on a loss of air which causes total RHR flow to lower initially. This causes an RCS heatup to commence since initially temperature was stable and heat removal has been reduced. RHR-764 must be throttled open to stabilize RCS temperature.

Question: 32
Tier/Group: 2/1
K/A Importance Rating: RO 2.9 SRO 2.9

K/A 005 Residual Heat Removal System (RHRS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.04: RHR valve malfunction.

Reference(s): ST-003 RHR, AOP-020
Proposed References to be provided to applicants during examination: None
Learning Objective: RHR009
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7,10
Comments: This question meets the K/A because the candidate must know how the RHR valve failure affects the RHR system and must determine the proper method per AOP-020 to mitigate the failure.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Section E

Loss Of RHR Flow Or Temperature Control

NOTE

RHR Pump Starting Duty requirements are contained in ATTACHMENT 7, RHR PUMP STARTING DUTY REQUIREMENTS.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
31.	**Maintain RCS Temperature as follows:**	
	a. Check RCS Temperature - RISING	a. Perform the following: 1) IF temperature is lowering, THEN perform the following to stabilize temperature: • Adjust HIC-758, RHR HX DISCH FLOW OR o Throttle RHR-764, HCV-758 BYPASS 2) IF RHR flow can NOT be maintained greater than 500 gpm, THEN control RCS temperature by starting and stopping the RHR pump at desired intervals. 3) Return to procedure and step in effect.
	b. Maintain RCS Temperature using one of the following: • Adjusting HIC-758, RHR HX DISCH FLOW OR o Throttle RHR-764, HCV-758 BYPASS	Won't work with valve failed closed
	c. Return to procedure and step in effect	

(862A and 863A -PC-601A, 862B and 863B -PC-600B). Keyed switches located behind the RTGB remove the control power from these valves during normal operation.

FCV-605, RHR HX Bypass Flow

FCV-605 will automatically maintain a preset flowrate through the operating RHR loop (set by operator). It is an air operated, fail closed valve. If FCV-605 did fail closed, all the flow would be directed through the RHR heat exchanger. This may result in Cooldown rate being higher than desired until valve control was obtained. This problem is addressed in AOP-020, Loss of RHR Cooling.

FCV-605 works in conjunction with hand control valve HCV-758 and FT-605. HCV-758 is adjusted to raise or lower flow through the RHR Heat Exchangers to change the Heat up rate (60°F/hr, **Admin 50°F/hr**) or Cooldown rate (100°F/hr, **Admin 80°F/hr**). This causes total system flow to be effected and is sensed by FT-605. The flow loop circuitry provides a control signal to FCV-605 which maintains a constant total system flow. The Hagen control potentiometer on the RTGB for this valve will adjust the RHR total system flow when in automatic. To close the valve the Hagan controller is placed in manual and the down arrow is depressed.

At power, Instrument Air is isolated to FCV-605 (Required by Tech. Specs. when in Mode 3, 2 or 1). A portable skid mounted controller is available for use during Post Fire Repairs if FCV-605 control circuits are damaged or inoperable. This procedure would use the Nitrogen system for motive force and for valve control.

HCV-758, RHR HX Discharge Flow

HCV-758 is throttled from RTGB to control Cooldown or Heat up rate by controlling RHR flow through the heat exchanger. It is an air operated valve that fails closed.

At power, Instrument Air is isolated to HCV-758 (Required by Tech. Specs. when in Mode 3, 2 or 1). A portable skid mounted controller is available for use during Post Fire Repairs if HCV-758 control circuits are damaged or inoperable. This procedure also allows the use of Nitrogen as a backup for motive force and for valve control.

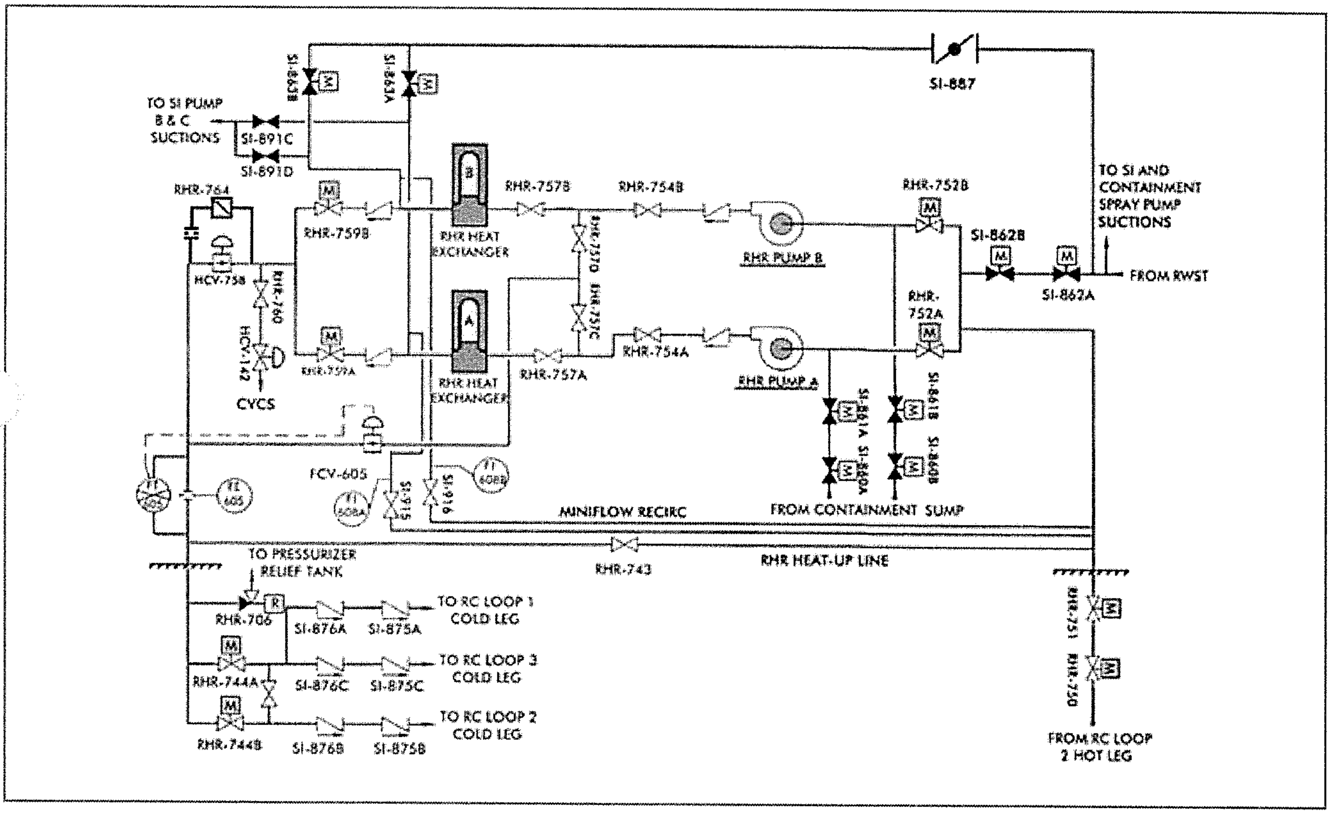
RHR-744 A & B, RHR to RCS Cold Legs

These motor operated valves in parallel provide a redundant path to the RCS during RHR operation and safeguard actuation. **The valves open automatically on a SI signal.**

RHR discharge to the RCS Cold legs passes through CV penetration #17. CV isolation for this penetration is said to exist when the RHR system is isolated (outside containment) and RHR-744A & B, RHR to RCS Cold Legs and RHR-706, RHR Relief valve are closed (inside containment).

RHR System Isolation Valves	
Design pressure	2485 psig
Design temperature	650°F

Figure 2 Normal Core Cooling



System Description

Normal Core Cooling Operation

Reactor coolant flows from "B" RCS hot leg to the RHR Pumps, through the tube side of the RHR Heat Exchangers where the heat is transferred to the Component Cooling Water (CCW) system. The flow then returns to all three RCS cold legs. A relief valve is provided for over pressure protection and relieves to the Pressurizer Relief Tank (PRT). A portion of the RHR loop flow can be directed to the CVC System for purification. The Cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of

33. 006 A4.02 002

Given the following plant conditions:

-The reactor is at 100% RTP

-A Safety Injection signal is received

SI-870 A/B, BIT INJECTION TANK OUTLETS
RHR-744 A/B, LOOP DISCHARGE TO RCS ISOLATION VALVES
SI-865 A/B/C, SI ACCUMULATOR DISCHARGE VALVES
SI-867 A/B, BIT INJECTION TANK INLETS

Which of the following valves will reposition because of the Safety Injection signal?

A.✓ SI-870 A/B & RHR-744 A/B

B. SI-865 A/B/C & SI-867 A/B

C. SI-870 A/B & SI-865 A/B/C

D. RHR-744 A/B & SI-867 A/B

The correct answer is A

A) Correct. SI-870's and RHR 744's will repostion during an SI signal.

B) Incorrect. Neither SI-865's nor SI-867's will reposition during an SI Signal. Plausible since these valves do receive SI signals, however, they do not reposition.

C) Incorrect. SI-870's will reposition. SI-865's will not reposition. Plausible because SI-865's do receive an SI signal, however, they do not reposition.

D) Incorrect. RHR 744's will repostion during an SI signal. SI-867's will not reposition during an SI Signal. Plausible because SI-867's do receive an SI signal, however, they do not reposition.

Question: 33
Tier/Group: 2/1
K/A Importance Rating: RO 4.0 SRO 3.8

K/A: 006 Emergency Core Cooling System (ECCS)
A4: Ability to manually operate and/or monitor in the control room:
A4.02: Valves

Reference(s): SIM/Plant Design, Logics
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 9 of the Safety Injecfion lesson plan
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.10
Comments:

K/A met because candidate must know what valves will reposition for a Safety Injection Signal so they can monitor on the RTGB.

Emergency Diesel Generator Startup

The safeguard trains will start their respective diesel generator by de-energizing their redundant solenoid operated air start valves. The diesel generator will come up to speed and voltage but will not close its output breaker unless there is a loss of voltage to their respective emergency busses (E1 and E2).

Feedwater Isolation

When this signal is received it will shut all of the feedwater regulating valves, feedwater bypass valves, feedwater block valves, and trip the Main Feedwater Pumps and the Turbine.

Safeguard Sequence Actuation

There are several different functions provided by this actuation.

Safety Injection - This lines up the Emergency Core Cooling System (ECCS) for the injection phase by operating the following valves:

HIGH HEAD SAFETY INJECTION

VALVE	TRAIN	SAFEGUARD POSITION
*SI-867A	A	Open
*SI-867B	B	Open
SI-870A	A	Open
SI-870B	B	Open

SAFETY INJECTION ACCUMULATORS

VALVE	TRAIN	SAFEGUARD POSITION
*SI-865A	A	Open
*SI-865B	B	Open

VALVE	TRAIN	SAFEGUARD POSITION
***SI-865C**	A	Open

NOTE

*These valves should already be open, but do receive an open signal.

LOW HEAD SAFETY INJECTION"A

RHR-744A	A	Open
RHR-744B	B	Open

The Status Light Panels on the RTGB will indicate at a glance if all the safeguard valves are in their proper position. Lights will be pink when in the proper position.

Starting of safeguard pumps and containment recirculation units. This equipment is timed onto their emergency busses to prevent overloading the diesel generator and the emergency bus. If equipment was already running, it will continue.

1. "A" -TRAIN
    a. 5 Seconds - "A" SI Pump starts.
    b. 5 Seconds - "B" SI Pump starts if Breaker 52/22B is racked in.
    c. 15 Seconds - "A" RHR Pump starts.
    d. 20 Seconds - "A" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.
    e. 25 Seconds - "B" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.
    f. 30 Seconds - No. 1 HVH Unit starts.
    g. 35 Seconds - No. 2 HVH Unit starts.
    h. 39.5 Seconds - "A" Auxiliary Feed Pump starts, FCV-1424 modulation is enabled, V2-16A/B/C open, and SGBD isolation valves for all SG's close.
2. "B"- TRAIN
    a. 5 Seconds - "C" SI Pump starts.
    b. 5 Seconds - "B" SI Pump starts if Breaker 52/29B is racked in.
    c. 15 Seconds - "B" RHR Pump starts.
    d. 20 Seconds - "C" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.

34. 006 K5.10 001

Given the following plant conditions:

-The reactor was at 100% RTP when a LOCA occurs

-A Safety Injection has been initiated and is injecting into the core

Immediately upon the Safety Injection Flow going to the core, the **INNER WALL** of the SI piping that connects to the RCS undergoes __(1)__ stress. If one of the SI pipes that connects to the RCS breaks from this stress, __(2)__ pipe(s) would still be available to supply SI flow to the core.

A. (1) tensile
(2) one

B.✓ (1) tensile
(2) two

C. (1) compressive
(2) one

D. (1) compressive
(2) two

The correct answer is B

A) Incorrect. Tensile stress is correct. One pipe still carrying flow is incorrect. Plausible if the student is unaware of how many RCS loops have tap ins for SI flow. There are three total, if you lose one, you have two left.

B) Correct. The inner wall of the piping undergoes tensile stress. This is due to the large Temperature difference between the inner wall and the outer wall. The cold fluid is cooling down the inner wall while the outer wall is still at normal RCS temperature. There are still two pipes carrying flow to the RCS.

C) Incorrect. Compressive stress is incorrect. Plausible since the outer wall of the piping is undergoing compressive stress. One pipe still carrying flow is incorrect. Plausible if the student is unaware of how many RCS loops have tap ins for SI flow. There are three total, if you lose one, you have two left.

D) Incorrect. Compressive stress is incorrect. Plausible since the outer wall of the piping is undergoing compressive stress. There are still two pipes carrying flow to the RCS.

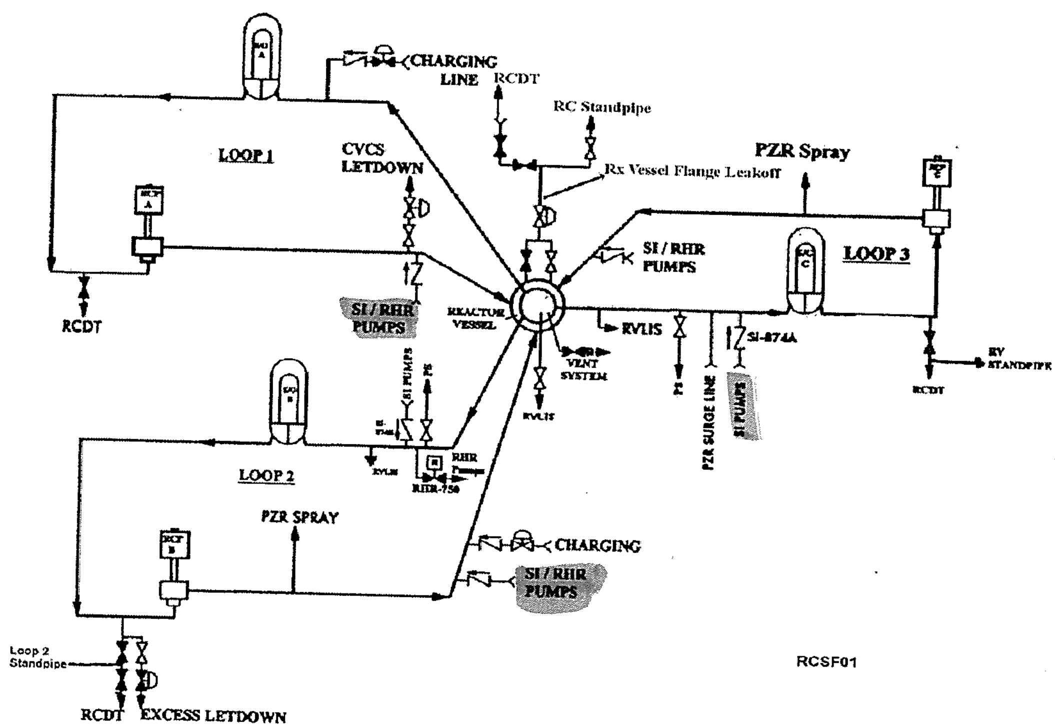
Question: 34
Tier/Group: 2/1
K/A Importance Rating: RO 2.5 SRO 2.9

K/A: 006 Emergency Core Cooling System (ECCS)
K5: Knowledge of the operational implications the following concepts as they apply to PRTS:
K5.10: Theory of thermal stress

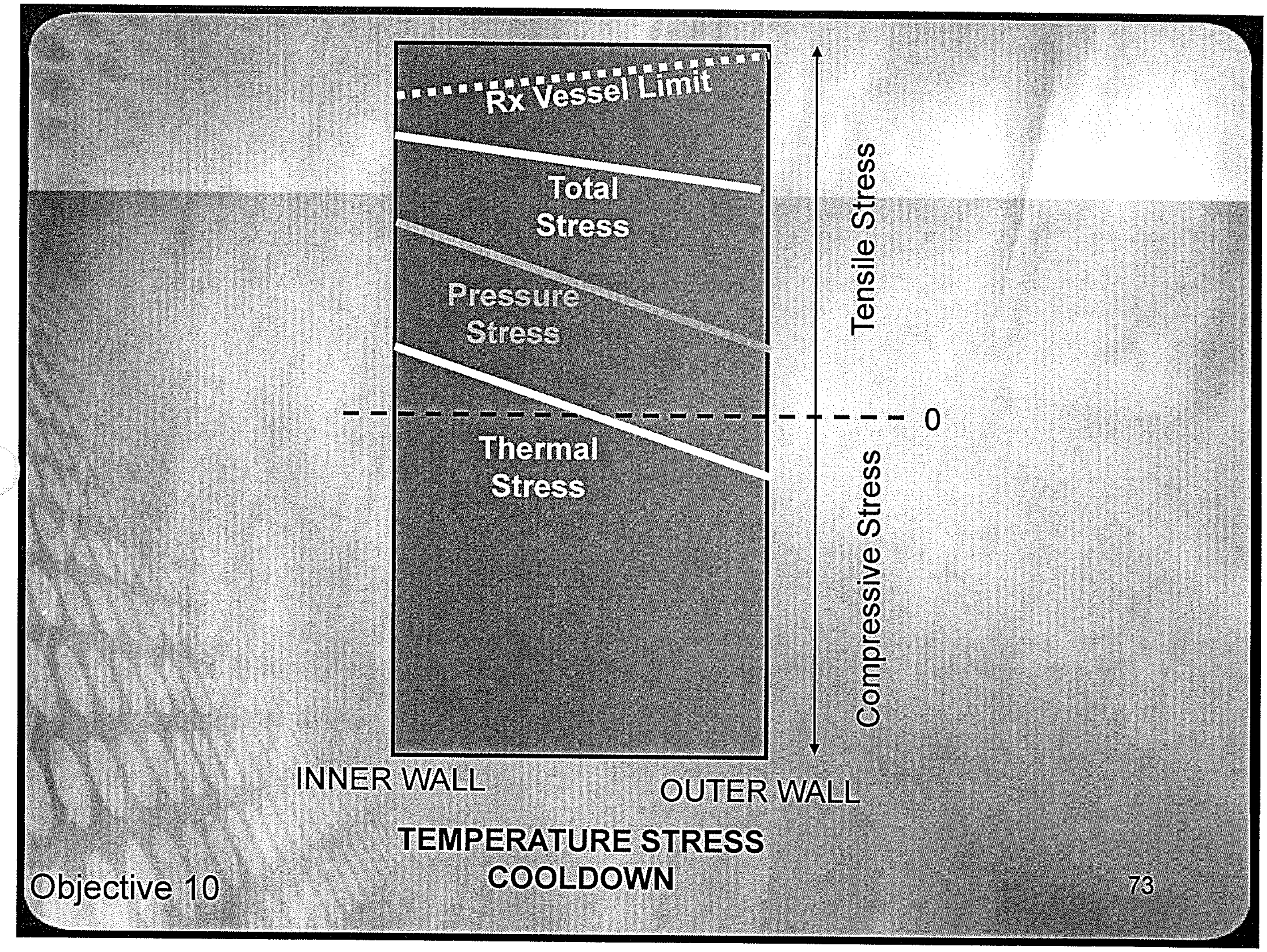
Reference(s): GFES, Brittle fracture, RCS drawing
Proposed References to be provided to applicants during examination - None
Learning Objective: 10 of Brittle Fracture lesson plan
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content - 41.5 / 45.7
Comments -

This question meets the K/A because the student must understand what stress the inner piping of the SI system goes through upon injection, then based off that piping failing, determine how many pipes are still carrying SI flow to the core.

INTRODUCTION to the RCS



RCSF01


Rx Vessel Limit
Total Stress
Pressure Stress
Thermal Stress
0
Tensile Stress
Compressive Stress
INNER WALL
OUTER WALL
TEMPERATURE STRESS
COOLDOWN
Objective 10
73

35. 007 K5.02 001

Given the following plant conditions:

- A bubble is being drawn in the Pressurizer IAW, OP-104, Pressurizer Operations
- RCS pressure is 335 psig
- TI-454, PRZR VAPOR TEMP is 432°F
- LI-462, COLD PRESSURIZER, Level is 80% and slowly lowering
- TI-463, PZR PWR RELIEF LINE TEMP, reads 320°F

A PZR PORV ____(1)____ leaking. Confirmatory indications would be ____(2)____ pressure, level, or temperature.

A.✓ (1) is
(2) PRT

B. (1) is
(2) RCDT

C. (1) is NOT
(2) PRT

D. (1) is NOT
(2) RCDT

The correct answer is A.

A. Correct. The high temp on the PWR Relief line is indicative of a leaking PORV. The PORV is leaking into the PRT.

B. Incorrect. The PORV would leak into the PRT. Plausible because the RCDT accepts drains from the RCS and PZR.

C. Incorrect. The high temp on the PWR Relief line is indicative of a leaking PORV. Plausible because PZR Vapor Temp is much higher than relief line temp.

D. Incorrect. The high temp on the PWR Relief line is indicative of a leaking PORV. Plausible because PZR Vapor Temp is much higher than relief line temp. The PORV would leak into the PRT. Plausible because the RCDT accepts drains from the RCS and PZR.

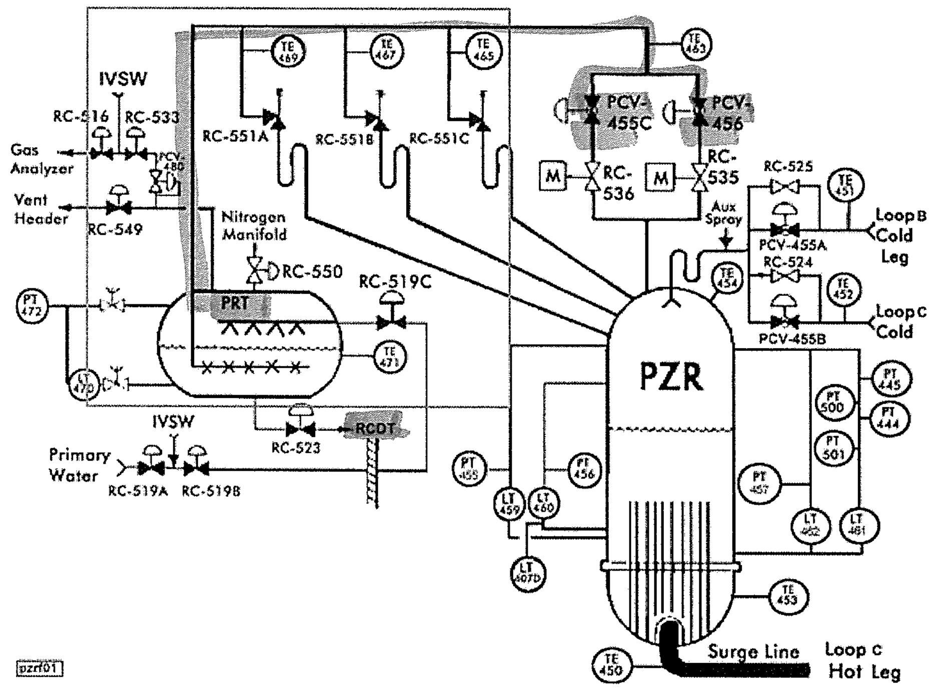
Question: 35
Tier/Group: 2/1
K/A Importance Rating: RO 3.1 SRO 3.4

K/A: 007 Pressurizer Relief Tank/Quench Tank System (PRTS)
K5: Knowledge of the operational implications the following concepts as they apply to PRTS:
K5.02: Method of forming a steam bubble in the PZR

Reference(s): OP-104
Proposed References to be provided to applicants during examination - Steam Tables
Learning Objective: Objective 2 of PZR/PRT Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content - 41.5
Comments -

This question meets the K/A because the candidate must know the operational implication of the indications required to be monitored while drawing a bubble in the pressurizer.

Figure 1 Pressurizer and PRT Simplified Drawing



8.1.1.2 (Continued) INIT

n. **IF** Plant conditions dictate aligning Hydrogen to the VCT, **THEN VERIFY** PCV-118, HYDROGEN SUPPLY REGULATOR TO VCT, regulates VCT pressure at 20 to 30 psig. ____

NOTE:	PZR PORV and Safety Valve leakage data is to be monitored until the RCS is at normal operating temperature and pressure, except as directed in this procedure.

o. **WHEN** PZR water temperature is greater than 200• F, **THEN PERFORM** one of the following:

1) **IF** ERFIS is available, **THEN USE** the following ERFIS points to monitor for PZR PORV and Safety Valve leakage. ____

- CHT0194A, TE-141 Letdown Line Relief Temp
- RCT0485A, TE-471 PRT Temp
- RCP0485A, PT-472 PRT Pressure
- RCL0485A, LI-470 PRT Level
- RCT0494A, TE-465 PZR Safety RLF Line Temp
- RCT0495A, TE-467 PZR Safety RLF Line Temp
- RCT0498A, TE-469 PRZ Safety RLF Line Temp
- RCT0481A, TM-454 Pressurizer Steam Temp
- RCT0493A, TE-463 PZR PORV Relief Line Temp
- RCP0493A, PT-500 PZR Pressure

36. 008 A2.03 001

Given the following plant conditions:

- The reactor is at 100% RTP
- CCW Heat Exchanger outlet temperature is 106°F and rising
- All Service Water Pumps are running
- The highest RCP Motor bearing temperature is 190°F and rising
- Service Water pressure is 46 psig and stable on both headers

AOP-018, REACTOR COOLANT PUMP ABNORMAL CONDITIONS
AOP-014, COMPONENT COOLING WATER SYSTEM MALFUNCTION

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

Based off the conditions above, the __(1)__ . The crew is required to __(2)__ .

A. (1) RCP operating limits have been exceeded
(2) monitor all RCP parameters IAW AOP-018.

B. (1) RCP operating limits have been exceeded
(2) Lower heat load on CCW system IAW AOP-014.

C.✓ (1) CCW system cooling capacity is degraded
(2) lower heat load on CCW system IAW AOP-014

D. (1) CCW system cooling capacity is degraded
(2) monitor all RCP parameters IAW AOP-018.

The correct answer is C.

A. Incorrect. No RCP motor operating limits have been exceeded yet. Plausible because RCP Bearing High Temperature alarm would be actuated which sends you to AOP-014. AOP-018 is plausible because high temperatures on RCP bearings is an RCP abnormal condition.

B. Incorrect. No RCP motor operating limits have been exceeded yet. Plausible because RCP Bearing High Temperature alarm would be actuated which sends you to AOP-014. Lower Heat load on CCW system IAW AOP-014 is correct.

C. Correct. CCW Heat Exchanger Outlet temperature is exceeding 105°F which means it's cooling capacity is degraded. Proper use of AOP-014 with the conditions noted sends the operator to step 11 of Section D to reduce loads on the CCW system to restore CCW outlet temperature < 105°F.

D. Incorrect. CCW Heat Exchanger Outlet temperature is exceeding 105°F which means it's cooling capacity is degraded. You will monitor RCP paramaters, however, it is in accordance with AOP-014, not AOP-018 for the given conditions.

Question: 36
Tier/Group: 2/1
K/A Importance Rating: RO 3.0 SRO 3.2

K/A: 008 Component cooling Water System (CCWS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the Component Cooling water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.03: High/low CCW temperature

Reference(s): AOP-014, APP-001
Proposed References to be provided to applicants during examination:
Learning Objective: AOP-014-004
Question Source: RNP Bank
Question History: 2004 NRC exam
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7, 10
Comments:

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 1 of 7)

1. **Monitor RCP Temperatures Using One Of The Following:**

   - ERFIS group display RCP LOG

     OR

   - RCP temperature recorder, TR-448

MOTOR BEARING	RCP A		RCP B		RCP C	
UPPER THRUST	PT.2	TE-417A	PT.9	TE-427A	PT.16	TE-437A
LOWER THRUST	PT.3	TE-417B	PT.10	TE-427B	PT.17	TE-437B
UPPER GUIDE	PT.4	TE-418A	PT.11	TE-428A	PT.18	TE-438A
LOWER GUIDE	PT.5	TE-419	PT.12	TE-429	PT.19	TE-439

2. **Check APP-001-B1, RCP BRG COOL WTR LO FLOW - EXTINGUISHED**

   Verify the following CCW Valves open:

   - CC-716A, CCW TO RCP ISO
   - CC-716B, CCW TO RCP ISO
   - CC-730, BRG OUTLET ISO

   IF CCW to the RCP(s) can NOT be restored, THEN Go To Step 4.

* 3. **Check ANY RCP Motor Bearing Temperature - GREATER THAN 200°F**

   IF any RCP Motor Bearing temperature exceeds 200°F, THEN perform Step 4

   Go To Step 5

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 2 of 7)

4.	**Determine If Reactor Trip Is Required As Follows:**	
	a. Check Reactor - CRITICAL	a. Perform the following: 1) Verify Control Rods - TRIPPED 2) Stop ALL Affected RCPs. 3) IF Control Rods were inserted on the trip, THEN perform the following: a) IF RCS temperature is greater than OR equal to 350°F, THEN Go To Step 4.d. b) IF RCS temperature is less than 350°F, THEN Go To Step 5. 4) IF Control Rods were already inserted, THEN Go To Step 5.
	b. Verify Reactor - TRIPPED	
	c. Stop ALL Affected RCPs	
	d. Go To EOP-E-0, Reactor Trip or Safety Injection, While Continuing With This Procedure	
5.	**Check CCW HX OUTLET Temperature - GREATER THAN 105°F**	Go To Step 17.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

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****

CAUTION

If only one SW Pump is running, it is subject to runout until the following step is completed.

****

6. **Check SW Header Pressure AND Transition To Steps Indicated By The Table Below:**

SW PRESSURE CONDITION	STEP
LESS THAN 40 PSIG	7
GREATER THAN 50 PSIG	8
BETWEEN 40 PSIG AND 50 PSIG	11

7. **Raise SW Pressure As Follows:**

a. Start additional SW Pumps as necessary to obtain at least 40 psig SW Header pressure

RESPONSE NOT OBTAINED: a. IF ALL available SW Pumps are running AND at least 40 psig can NOT be obtained, THEN isolate SW to the Turbine Building by closing:

- V6-16C, SW TURB BLDG ISO

OR

- V6-16A AND V6-16B, SW TURB BLDG SUPPLY

b. Check SW Header pressure - GREATER THAN 50 PSIG

RESPONSE NOT OBTAINED: b. Go To Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 4 of 7)

8.	**Reduce SW Pressure As Follows:**	
	a. Check number of SW Pumps Running - GREATER THAN 2	a. WHEN personnel are available, THEN locally perform Attachment 10, Throttling CCW Heat Exchanger SW Valves, while continuing with this procedure. Go To Step 11.
	b. Stop 1 Pump	
	c. Check SW Header Pressure - GREATER THAN 50 PSIG	c. Go To Step 8.e.
	d. Go To Step 8.a	
	e. Check SW Header pressure - GREATER THAN 40 PSIG	e. WHEN personnel are available, THEN locally perform Attachment 10, Throttling CCW Heat Exchanger SW Valves, while continuing with this procedure.
9.	**Check SW To Turbine Building Status - ISOLATED**	Go To Step 11.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 5 of 7)

10. **Shutdown Secondary Systems As Follows:**

    a. Close all MSIVs AND MSIV BYPASS Valves

    b. Break vacuum to the Condenser as follows:

       1) Depress AND hold the THINK Button

       2) Open VACUUM BREAKER VALVES:

          - MS-70A
          - MS-70B

       3) WHEN Vacuum Breaker Valves are Open, THEN Release the THINK Button

    c. Verify The Following Equipment - STOPPED:

       - FW PUMP A AND B
       - COND PUMP A AND B
       - HEATER DRAIN PUMP A AND B
       - GOV FLUID PUMP A AND B
       - VACUUM PUMP A AND B

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION D

CCW SYSTEM LOW FLOW OR HIGH TEMPERATURE

(Page 6 of 7)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
11.	**Reduce Heat Loads On The CCW System As Necessary To Maintain Temperature**	
	a. Stop Waste Gas Compressor(s)	
	b. Secure excess letdown	
	c. Check RHR - IN CORE COOLING MODE	c. Go To Step 11.e.
	d. Minimize RCS cooldown rate	
	e. Reduce number of Charging Pumps in service	
	f. Throttle CC-775, CC FROM SPENT FUEL PIT HX BUTTERFLY, to maintain SFP temperature between 115°F and 120°F (located East of Heat Exchanger 9 foot above floor)	
12.	**Check CCW Temperature -** • LESS THAN 105°F AND • STABLE OR LOWERING	Go To Step 14.
13.	**Go To The Main Body, Step 4 Of This Procedure**	

ALARM

RCP A BEARING HI TEMP

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Loss of Component Cooling Water
2. Failure of Bearing
3. Inadequate Seal Leakoff flow at reduced RCS pressure (Pump bearing)

OBSERVATIONS

1. Component Cooling Water Flow
2. RCP Bearing Temperatures (RCP Temperature Recorder and Computer)
3. RCP Vibration Monitor

ACTIONS

CK (✓)

> **NOTE:** If more than 15 minutes elapses without RCP Seal Cooling, then Seal Cooling must be isolated before starting CCW OR Charging to prevent Seal damage.

1. **IF** CCW AND Seal Injection are lost to any RCP, **THEN REFER TO** AOP-018. _____
2. **IF** CCW is lost to the motor bearing oil coolers, **THEN REFER TO** AOP-014. _____
3. **IF** any RCP bearing vibration is high, **THEN REFER TO** AOP-018. _____
4. **IF** at reduced RCS pressure **AND** OP-101 requirements are met, **THEN VERIFY OPEN** CVC-307, PRI SEAL BYP ISO. _____
5. **IF** due to extended loss of RCP Seal Injection Flow, **THEN** refer to AOP-014. _____

DEVICE/SETPOINTS

1. TE-417A, TE-417B, TE-418A, TE-419 / 185°F
2. TE-131 / 175°F

POSSIBLE PLANT EFFECTS

1. RCP Bearing Failure/trip

REFERENCES

1. ITS LCO 3.4.4, LCO 3.4.5, LCO 3.4.6 and LCO 3.7.6
2. AOP-014, Component Cooling Water System Malfunction
3. AOP-018, Reactor Coolant Pump Abnormal Conditions
4. OP-101, Reactor Coolant System and Reactor Coolant Pump Startup and Operation
5. CWD B-190628, Sheet 114, Cable T

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

SECTION A

REACTOR COOLANT PUMP SEAL FAILURE

(Page 6 of 13)

*14. **Check ANY Of The Following Conditions Met:**

- RCP Bearing Temperature - GREATER THAN 225°F
- RCP Bearing Temperature - RISING
- RCP #1 Seal Leakoff Temperature - GREATER THAN 235°F
- RCP #1 Seal Leakoff Temperature - RISING

Perform the following:

a. Continue to monitor RCP Bearing Temperature, RCP Seal Leakoff Temperature, AND RCP Seal Leakoff Flow.

b. Place the Plant in Mode 3 within 8 hours using GP-006-1, Normal Plant Shutdown From Power Operation To Hot Shutdown.

c. IF RCP Bearing OR Seal Leakoff Temperature begin to rise, THEN Observe the CAUTION prior to Step 3 and Go To Step 3.

d. WHEN the plant has been placed in Mode 3, THEN stop the affected RCP(s)

e. Go To Step 30.

15. **Observe The CAUTION Prior To Step 3 And Go To Step 3**

37. 010 K1.08 002

Given the following plant conditions:

-RCS Tavg is 560°F

-LT-459, PRZR LEVEL, has failed high

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

Prior to the failure, pressurizer level was __(1)__ . Pressurizer heaters will all turn on when LT-459 reaches __(2)__ .

A. (1) 36.1%
(2) 46.1%

B. (1) 53.3%
(2) 63.3%

C. (1) 53.3%
(2) 58.3%

D.✓ (1) 36.1%
(2) 41.1%

The correct answer is D

A) Incorrect. 36.1% is the correct level is the correct pressurizer level prior to the failure. 46.1% is incorrect. This is 10% above reference level. Plausible because 10% is the trip criteria for pressurizer level, therefore, it is a number that is associated with the pressurizer.

B) Incorrect. 53.3% is incorrect. Plausible because this is the max level in which the pressurizer level is maintained. Also, this level was used by using the S/G level control system. S/G level is ramped up from 0-20%(power). Then from 20%-100%(power), level is stable. On the Pressurizer, level is ramped up from 0%-100%(power), at 100% power, level is stable at 53.3%. 63.3% is incorrect. This is 10% above reference level. Plausible because 10% is the trip criteria for pressurizer level, therefore, it is a number that is associated with the pressurizer.

C) Incorrect. 53.3% is incorrect. Plausible because this is the max level in which the pressurizer level is maintained. Also, this level was used by using the S/G level control system. S/G level is ramped up from 0-20%(power). Then from 20%-100%(power), level is stable. On the Pressurizer, level is ramped up from 0%-100%(power), at 100% power, level is stable at 53.3%. 58.3% is incorrect. This value is 5% above the reference level. 5% above reference level is when pressurizer heaters turn on.

D) Correct. 36.1% is the correct level. By using RCS Tavg and converting that to a pressurizer level, you get 36.1% Pressurizer level ramps linearly from 547°F to 575.9°F. So at 547°F, pressurizer level is 22% and at 575.9°F, pressurizer level is 53.3%. 41.1% is correct. This value is 5% above the reference level and this is when pressurizer heaters will turn on.

$$\frac{31.1\%}{28.9°F} = \frac{X}{13°F}$$

X=36.1%

Question: 37
Tier/Group: 2/1
K/A Importance Rating: RO 3.2 SRO 3.5

K/A: 010 Pressurizer Pressure Control System (PZR PCS)
K1: Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems:
K1.08: PZR LCS

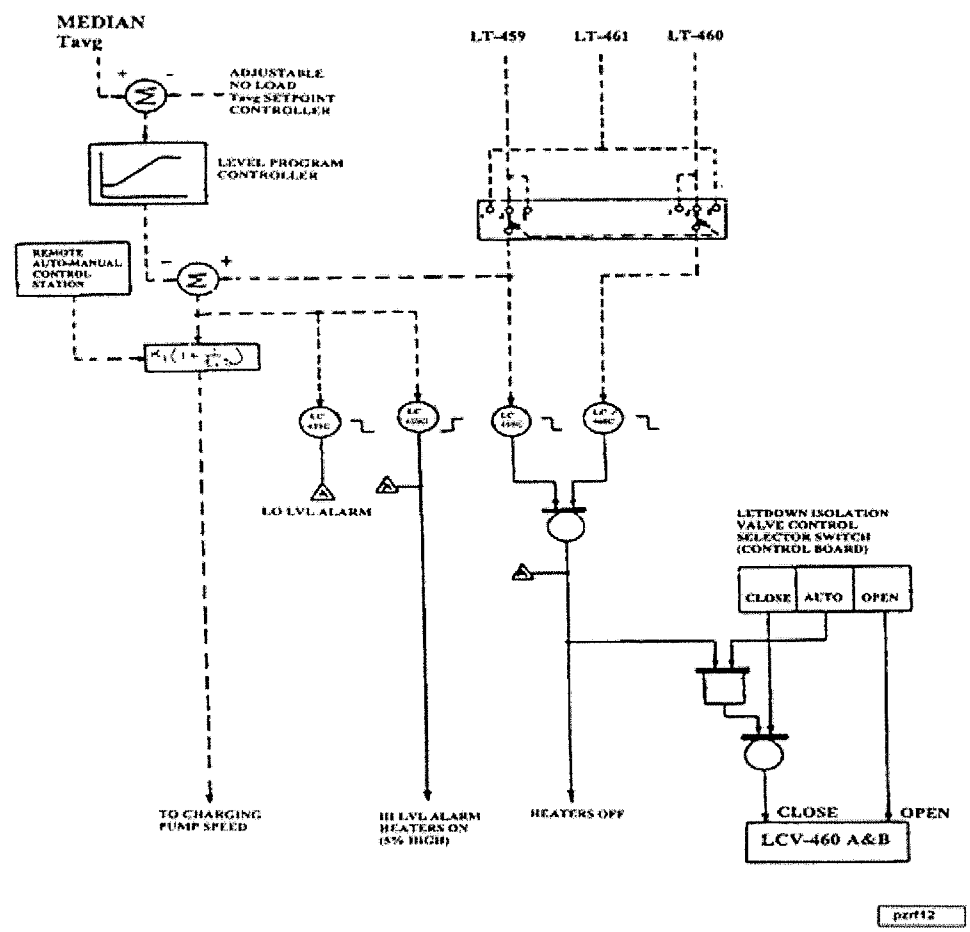
Reference(s): OP-104, ST-059 PZR, ST-021 CVCS
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 41.7
Comments:

This question meets the K/A because the candidate must know how changes in pressure caused by the PZR PCS affect the operation of the PZR LCS.

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

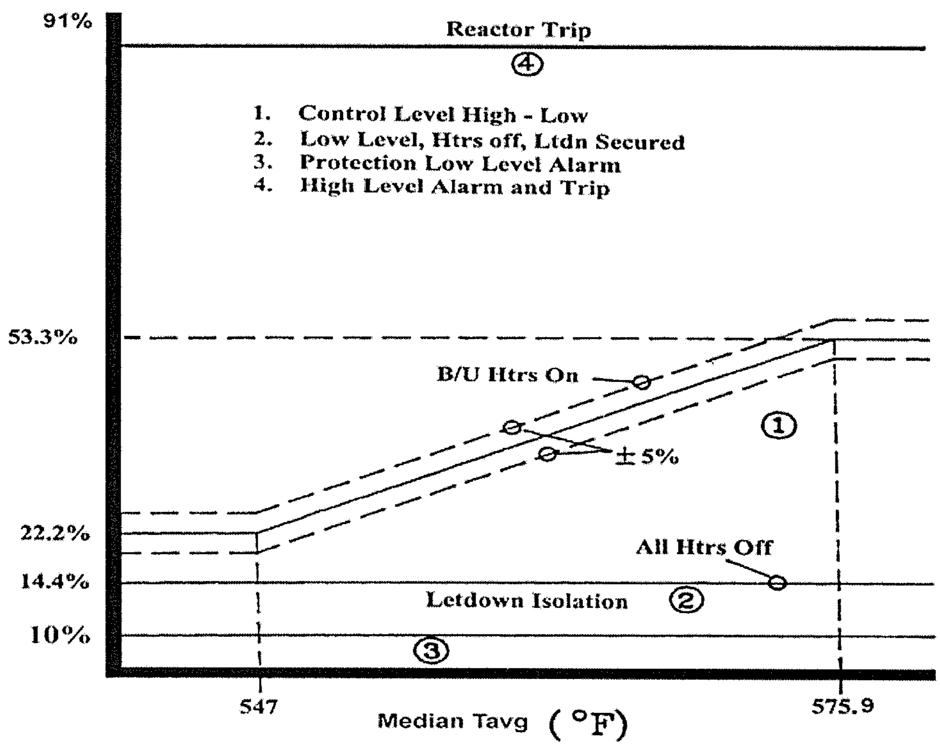
LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would de-energize and any backup heater in manual would remain energized.

Figure 23 Level Control Logic



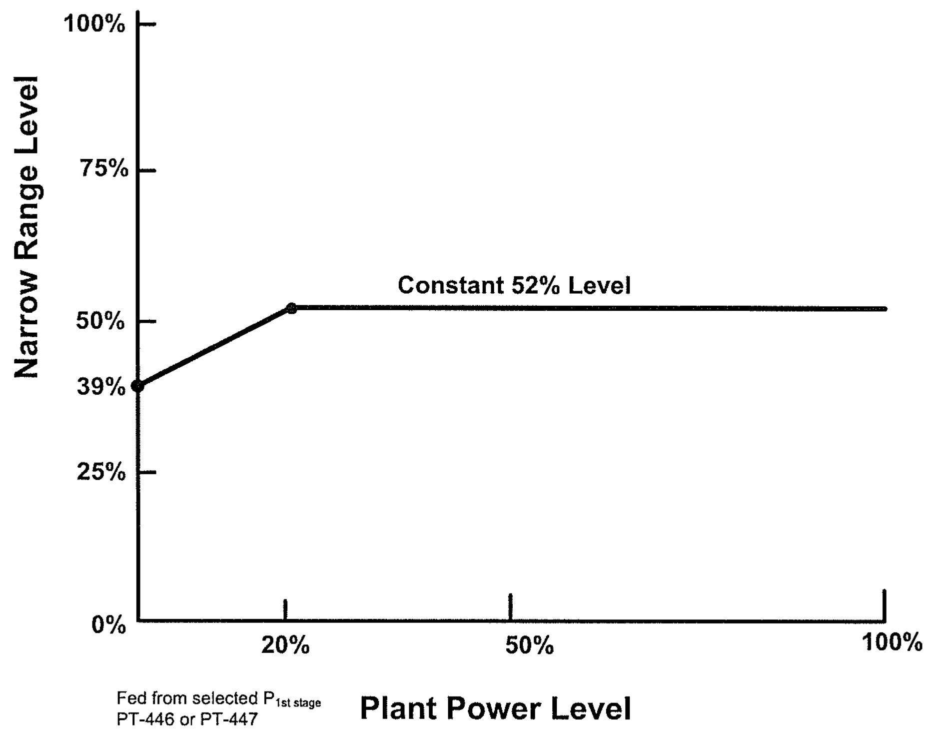
CorrecT

Figure 24 Pressurizer Level Control Setpoints



InCorrect

Figure 18 Ramped Steam Generator Level



S/G Water Level

Each S/G has three (3) transmitters for level protection.

Each transmitter is a differential pressure detector that measures the differential pressure between the reference leg (outside S/G) and the variable leg (water level inside S/G downcomer).

AMSAC Initiation Logic

The signals are used for protection, alarms, indication, AMSAC, and control of S/G level during power operation.

In addition, each S/G has one (1) transmitter for wide range level indication and one (1) transmitter for providing level indication to the Dedicated Shutdown Panel.

38. 012 A3.04 001

Given the following plant conditions:

-The reactor is at 37% RTP

-RCP breaker A trips

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

Reactor Trip Breakers RTA and RTB ___(1)___ lights will be illuminated. The conditions for P-8 ___(2)___ satisfied.

A. (1) red
   (2) are

B. (1) green
   (2) are

C.✓ (1) red
   (2) are **NOT**

D. (1) green
   (2) are **NOT**

The correct answer is C.

A. Incorrect. P-8 is not satisfied. Plausible because at 35% turbine power another protective feature (AMSAC) is armed.

B. Incorrect. The green lights for RTA and RTB will not illuminate because. P-8 is not satisfied. Plausible because at 35% turbine power another protective feature (AMSAC) is armed.

C. Correct. RTA and RTB remain closed (red light illuminated) because the P-8 logic is not satisfied. P-8 is satisfied when 2/4 Power Ranges exceed 40% RTP. The conditions given in the stem is 37%. When power is between 10% and 40% two RCPs would have to trip to get an automatic reactor trip.

D. Incorrect. The green lights for RTA and RTB will not illuminate because. P-8 is not satisfied. because at 35% turbine power another protective feature (AMSAC) is armed.

Question: 38
Tier/Group: 2/1
K/A Importance Rating: RO 2.8 SRO 2.9

K/A: 012 Reactor Protection System
A3: Ability to monitor automatic operation of the RPS, including:
A3.04: Circuit breaker

Reference(s): ST-011
Proposed References to be provided to applicants during examination: None
Learning Objective: LP-11 Obj. 7d, 10
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.6, 7
Comments: This question meets the K/A because the candidate must be able to determine if automatic operation of the RPS is expected due to circuit breaker operation.

8. Reactor Coolant Pump (RCP) Circuit Breaker Open Trip

    a. The RCP Circuit Breaker Open Trip provides protection for the Reactor against DNB following a loss of coolant flow accident. This trip will occur when:

        i. **Above 10% (P-7) 2 out of 3 RCP Breakers Open**

        ii. **Above 40% (P-8) 1 out of 3 RCP Breakers Open**

        iii. **No trip occurs below 10% (P-7) for any Loss of Flow**

9. RCP Bus UV Trip

    a. The RCP Bus UV Trip provides protection for the Reactor against DNB as a result of a loss of voltage to more than one RCP. This trip occurs when an UV condition exists on 2 out of 3 RCP Buses when above 10% (P-7). This trip is automatically blocked below 10% (P-7).

       This trip assures a Reactor Trip Signal is generated before the Low Flow Trip Setpoint is reached.

    b. Setpoint - **75% of nominal Bus Voltage or 3120 V**

10. RCP Bus Underfrequency Trip (Not a direct reactor trip)

    a. The RCP Bus Underfrequency Trip trips the RCP breakers. RCP breaker open signal trips the reactor which provides protection for the Reactor against DNB as a result of an underfrequency on more than one RCP Bus. This trip occurs when an underfrequency condition exists on 2 out of 3 RCP Buses when above 10% (P-7). This trip is automatically blocked below 10% (P-7).

       This trip assures a Reactor Trip Signal is generated before the Low Flow Trip Setpoint is reached.

    b. Setpoint - **58.2 Hertz**

11. High PZR Pressure Trip

    a. The High PZR Pressure Trip provides protection for the Reactor Coolant System (RCS) against over pressurization. The setpoint is selected to be below the PZR safety valve actuation pressure and above the PORV setting. This setting minimizes challenges to safety valves while avoiding

PERMISSIVE NUMBER	DERIVATION	FUNCTION
	3/4 Power Ranges below setpoint (10%) from P-10 AND 2/2 Turbine First Stage Pressure below setpoint (10%)	Blocks the following reactor trips: • RCS Low Flow • RCP Breakers Open • UV • PZR Low Pressure • PZR High Level
P-8	2/4 Power Ranges above setpoint (40%) 3/4 Power Ranges below setpoint (40%)	Enables Reactor Trip on low flow in a single loop Enables Turbine Trip Blocks Reactor Trip on low flow in a single loop Blocks Turbine Trip

39. 013 K2.01 001

Given the following conditions:

- Plant is at 100% RTP
- All equipment is in normal alignment
- An electrical fault results in a loss of power to Instrument Bus 3

Which ONE (1) of the following describes SI Pump response if an AUTO SI actuation occurs before Instrument Bus 3 is recovered?

A. SI Pumps A and B start.

B. SI Pumps B and C start.

C✓ Only SI Pump A starts.

D. Only SI Pump C starts.

The correct answer is C.

A. Incorrect. Plausible because candidate could believe that the "B" breaker is normally racked in.

B. Incorrect. Plausible because candidate could believe that the "B" breaker is normally racked in. The candidate could believe Instrument Bus 3 powers the "A" Sequencer.

C. Correct. "A" SI Pump is started by the "A" Sequencer which still has power from Instrument Bus 7A. The "B" SI Pump does not start because its breaker is not normally racked in. "C" SI Pump does not start because it is started by the "B" train sequencer which lost power with Instrument Bus 3.

D. Incorrect. "C" SI Pump is started by the "B" Sequencer, which will not have power. The candidate could believe Instrument Bus 3 powers the "A" Sequencer.

Question: 39
Tier/Group: 2/1
K/A Importance Rating: RO 3.6 SRO 3.8

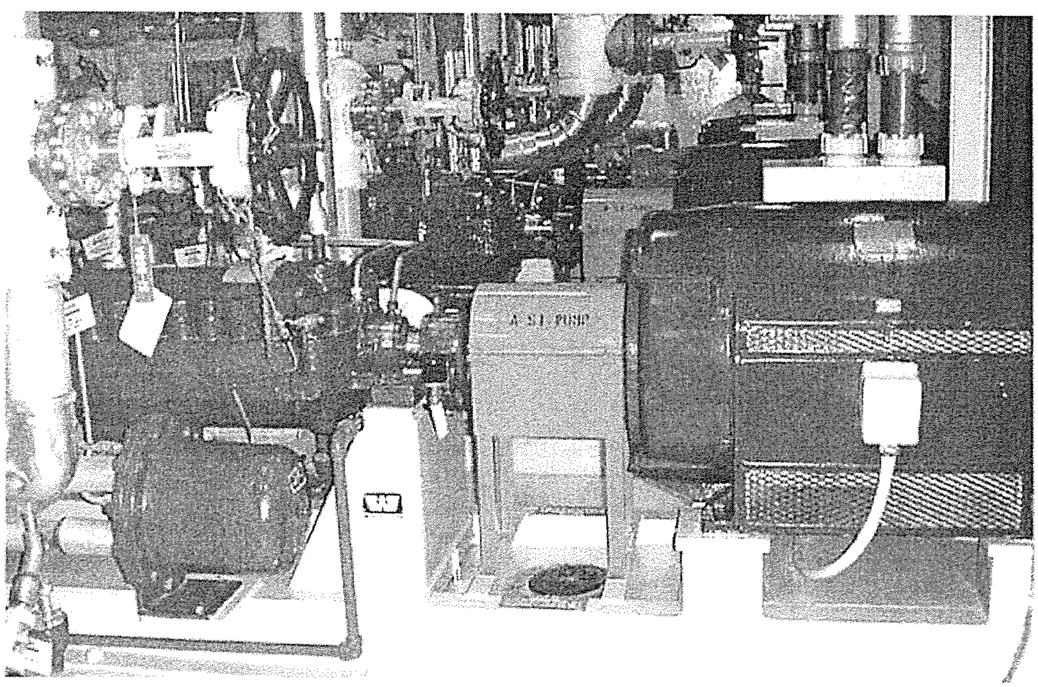
K/A: 013 Engineered Safety Features Actuation System (ESFAS)
K2: Knowledge of the power supplies to the following:
K2.01: ESFAS/Safeguards Equipment Control

Reference(s): EDP-008, ST-002 SI
Proposed References to be provided to applicants during examination: None
Learning Objective: ESF004a, ESF008j
Question Source: RNP Bank
Question History:
Question Cognitive Level: Comprehension/Analysis
10 CFR Part 55 Content: 41.7
Comments: This question meets the K/A because the candidate must know the power supply to ESFAS/Safeguards Equipment Control (power to SI Sequencers) to answer the question.

Material	Martensitic stainless steel
Power	350 HP

"A" and "C" SI Pumps are normally lined up for auto start upon safeguard actuation. The supply breakers for "B" SI pump are normally racked out. "B" SI Pump is available to replace "A" or "C" pumps. It is a good operating practice to vent SI pump "B" prior to operation if it has been idle for more than 3 months. Instructions for filling and venting portions of the Safety Injection system and individual SI components are located in OP-202-1, SI System Venting. The SI pumps are used to fill the SI Accumulators and can be used to fill the Refueling Cavity (slow fill). They are motor driven, multi-stage, centrifugal pumps. The seals are cooled by Component Cooling Water in the seal water heat exchanger. The thrust bearing is cooled by Service Water (SW) which discharges to the storm drains. If SW cooling for the thrust bearing is not available, emergency cooling alignment can be from Fire Water or the Primary Water pumps in accordance with AOP-022, Attachment 1.

Figure 10 SI Pump



INSTRUMENT BUS NO. 3

Location: Safeguards Room, East Wall

Normal Power: Inverter "B" / Alternate Power: MCC-8 (2GL)

	SPARE		**Instrument Bus No.3 Power Supply (From INST BUS 3 PWR XFER SW)**
1	SPARE	2	Hagan Rack 8 (CWD 417)
3	Hagan Rack 14 Isolator Rack 30, Channel 3	4	Hagan Rack 15 (CWD 457)
5	Hagan Rack 16 (CWD 418)	6	Hagan Rack 17 (CWD 421); LI-970 (CWD 495); LI-969 (CWD 494A)
7	Hagan Rack 18 (CWD 422)	8	RTGB "C" TB-UE 42 and 43 RTGB "A" TB-SM 41 and 42 (CWD's 434, 963 and 964); FR-498 rec (CWD 964)
9	AFW Pump "B" Flow Control Valve FCV-1425, FIC-1425 (CWD 658)	10	NIS Cabinet "C", CWD 446
11	Safeguards Rack 63 (CWD 420)	12	Safeguards Rack Status Lights (CWD 397)
13	Spared by EC 79124	14	FIC-638, FIC-657 (CWD-489)
15	Exhaust hood spray valves (CWD 761)	16	Turbine Supervisory Recorder, Net Generation Recorder, Turbine MSR Temperature Recorder (CWD 791, 792, 793)
17	SPARE	18	Load, Frequency control panel (CWD 722)
19	NIS Cabinet "C" (2 Pole Bkr) (CWD 446)	20	Safeguards sequencing relays Train "B"
21		22	BLANK
23	Power Panel No. 26 Alternate Supply	24	BLANK
25	Instrument Bus No. 8	26	SPARE (3 Pole Bkr)
27	(3-pole breaker)	28	
29		30	

40. 022 A3.01 001

Given the following plant conditions:

-Tavg is 547°F

-RCS Pressure is 2235 psig

-A loss of offsite power occurs

-One minute later a large break LOCA occurs

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

The **FINAL** CV Recirculation fans start ___(1)___ seconds after both trains of the ___(2)___ sequencers start.

A. (1) 30
(2) Blackout

B. (1) 35
(2) Blackout

C. (1) 30
(2) Safety Injection

D.✓ (1) 35
(2) Safety Injection

The correct answer is D.

A. Incorrect. 30 seconds is incorrect. Plausible because this is the time that the first HVH fan starts on both sequencer trains for a Safety Injection. The Blackout sequencer is incorrect. Plausible since the blackout sequencer did start, however, HVH units do not start on the blackout sequencer.

B. Incorrect. 35 seconds is correct. The Blackout sequencer is incorrect. Plausible since the blackout sequencer did start, however, HVH units do not start on the blackout sequencer.

C. Incorrect. 30 seconds is incorrect. Plausible because this is the time that the first HVH fan starts on both sequencer trains for a Safety Injection. The question is asking for the time the FINAL HVH fan starts. Safety Injection sequencer is correct since they have experienced a LOCA.

D. Correct. 35 seconds is correct. This is the time the Safety Injection sequencer starts the FINAL HVH fan. Safety Injection sequencer is correct since they have experienced a LOCA.

Question: 40
Tier/Group: 2/1
K/A Importance Rating: RO 4.1 SRO 4.3

K/A: 022 Containment Cooling System (CCS)
A3: Ability to monitor automatic operation of the CCS, including:
A3.01: Initiation of safeguards mode of operation

Reference(s): ST-006 ESF
Proposed References to be provided to applicants during examination: None
Learning Objective: CVHVAC005b
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7
Comments: This question meets the K/A because the candidate must be able to determine when the CV Recirculation fans will be started for the safeguards mode of operation.

CHAPTER 6

SYSTEM OPERATION

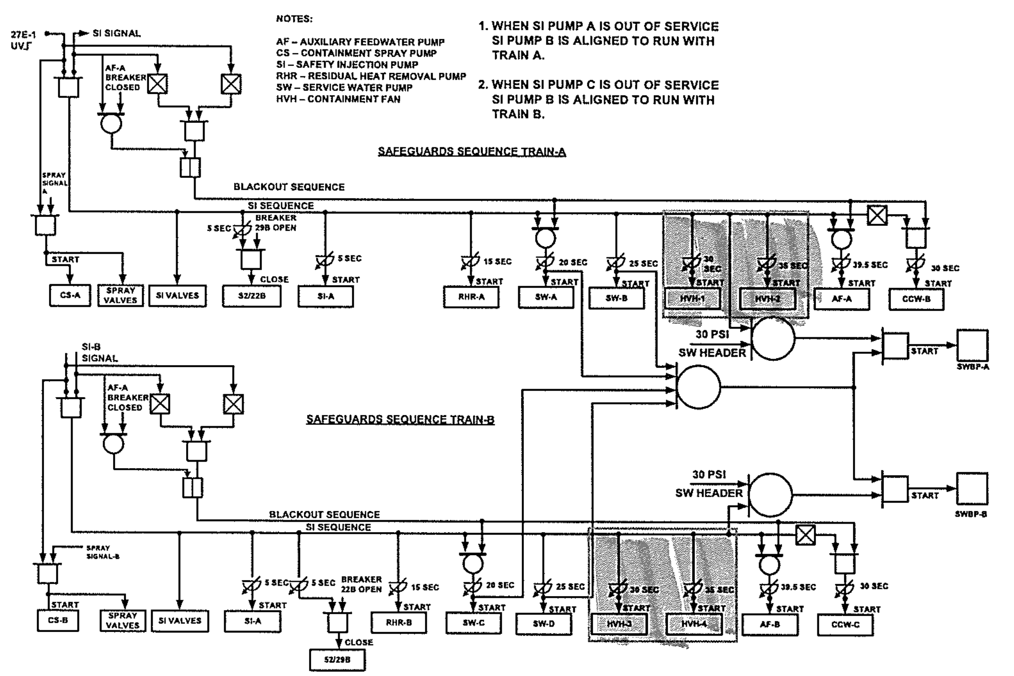
NOTE

During normal operation, this system stays in a standby status. Therefore, the only operations covered here will be emergencies.

Safety Injection Actuation

After an SI signal (also called "S" signal) is received, the signal is automatically locked in until it is reset. Then these evolutions will follow: (Note that the signal cannot be reset until at least two minutes have elapsed from the time of receipt.)

Figure 18 Safety Injection Sequence



REACTOR TRIP

The under voltage coil on the reactor trip breaker will de-energize and the shunt trip coils will be energized causing the breakers to open.

LOW HEAD SAFETY INJECTION"A

RHR-744A	A	Open
RHR-744B	B	Open

The Status Light Panels on the RTGB will indicate at a glance if all the safeguard valves are in their proper position. Lights will be pink when in the proper position.

Starting of safeguard pumps and containment recirculation units. This equipment is timed onto their emergency busses to prevent overloading the diesel generator and the emergency bus. If equipment was already running, it will continue.

1. "A" -TRAIN

    a. 5 Seconds - "A" SI Pump starts.

    b. 5 Seconds - "B" SI Pump starts if Breaker 52/22B is racked in.

    c. 15 Seconds - "A" RHR Pump starts.

    d. 20 Seconds - "A" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.

    e. 25 Seconds - "B" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.

    f. 30 Seconds - No. 1 HVH Unit starts.

    g. 35 Seconds - No. 2 HVH Unit starts.

    h. 39.5 Seconds - "A" Auxiliary Feed Pump starts, FCV-1424 modulation is enabled, V2-16A/B/C open, and SGBD isolation valves for all SG's close.

2. "B"- TRAIN

    a. 5 Seconds - "C" SI Pump starts.

    b. 5 Seconds - "B" SI Pump starts if Breaker 52/29B is racked in.

    c. 15 Seconds - "B" RHR Pump starts.

    d. 20 Seconds - "C" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.

e. 25 Seconds - "D" Service Water Pump starts and a start signal is supplied to both Service Water Booster Pumps.

f. 30 Seconds - No. 3 HVH Unit starts.

g. 35 Seconds - No. 4 HVH Unit starts.

h. 39.5 Seconds - "B" Auxiliary feed pump starts, FCV-1425 modulation is enabled, V2-16A/B/C open, and SGBD isolation valves for all SG's close.

During SI sequencing, WCCU-1A and 1B are inhibited and are not available until 10 seconds after the AFW Pumps receive their start signal. This ensures the AFW Pumps are at their normal running current before the inhibit is removed.

These actions have completed the requirements for injection from high head and low head Safety Injection and have ensured a path is available from the accumulators. Injection will now commence dependent upon Reactor Coolant System pressure. "B" SI Pump is designated as a maintenance pump and will only be used in the event either "A" or "C" SI Pump is declared out of service.

Operation of the Service Water Booster Pumps and the Containment air recirculation units are now cooling and depressurizing the CV if the accident was a loss-of-coolant or steam line break inside containment.

The operation of the Auxiliary Feed System ensures the availability of the steam generators for decay heat removal.

PHASE A CV ISOLATION AND ISOLATION VALVE SEAL WATER SYSTEM ACTUATION

Containment isolation valves in non-essential process lines are shut to minimize the leakage from containment. To ensure that the valve seats do not allow leakage from containment, water is injected between the isolation valves at a pressure higher than CV design pressure. For a list of valves actuated during Phase "A" isolation, see below, and/or refer to EPP Supplement A, Volume 3, Part 4 of POM.

Placing the "Instrument Air Isolation to Containment" switch on the RTGB in "Override" allows the instrument air supply valve to the Containment (PCV-1716) to be opened with a T-signal (Phase A Containment Isolation) present.

The signal causing this alarm will also CLOSE the following valves;

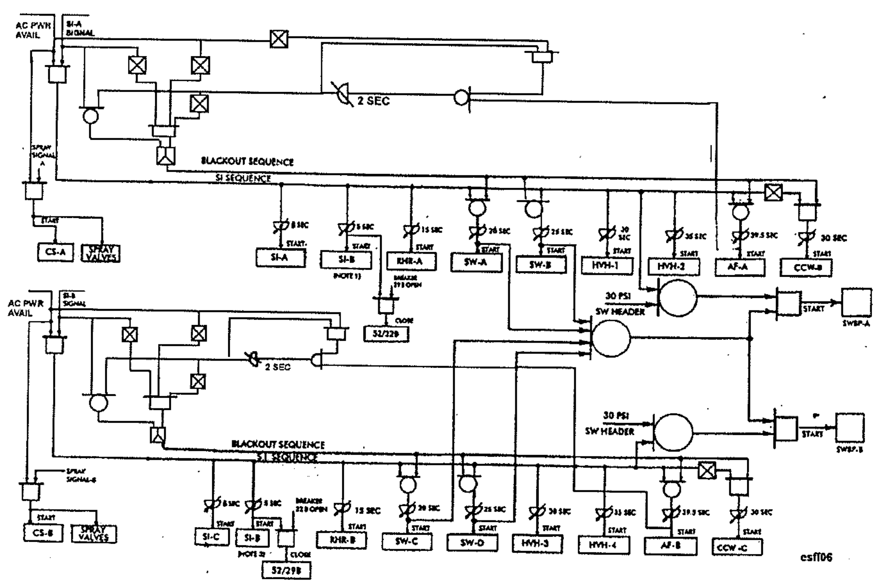
1. PS-956A & PS-956B Pressurizer Steam Space Valves
2. PS-956C & PS-956D Pressurizer Liquid Space Valves

- CC-730 - CC from R.C.P. "A", "B", "C" Oil Cool Isol.
- FCV-626 - CC From R.C.P. "A", "B", "C" Thermal Barrier Isol.

Station Blackout

Occurs on a sensed loss of AC power (<328V or degraded grid setpoint of<430V for ten seconds) to the emergency busses (E1 and/or E2). Degraded Grid protection is to open the normal supply breaker to the emergency bus. The UV relays (2) will now see a complete loss of voltage and start the blackout sequence. Note that 125VDC power must be present for the blackout sequencer to operate.

Figure 19 Blackout Sequence



Loss of Power to E1

Isolates E1 from rest of electrical plant by opening normal supply breaker (52/18B). It also strips all the breakers on E1 except for the MCC5/16 supply breaker. A block signal is placed on "A" Auxiliary Feed Pump to prevent its breaker from closing due to a loss of feed pumps, low-low level in a steam generator or AMSAC. The diesel generator is started and its output breaker closes when output voltage builds up (within 10 seconds). There is a two-second time delay that ensures that a voltage dip does not result in closure unless E1 is de-energized. The closing of the output breaker renews the block to ensure that "A" Auxiliary Feed Pump starts at the proper time on the sequence. When the loss of voltage on the E1 Bus is sensed with no SI signal present, the timing for the blackout sequence starts. After twenty (20) seconds "A" Service Water Pump starts, a start signal is sent to both Service Water Booster Pumps, and the booster pumps will start when the

Service Water Header pressure builds up to 30 psig. After twenty (25) seconds "B" Service Water Pump starts and a start signal is sent to both Service Water Booster Pumps. Both Service Water pumps are started to ensure sufficient flow exist to the EDG if one of the pumps is inoperable. After thirty (30) seconds "B" Component Cooling Pump starts. After thirty-nine and one-half (39.5) seconds "A" Auxiliary Feed Pump will start and reset the blackout sequence after a 2 second delay.

If sufficient voltage does not exist on E1 at the end of one cycle, the control circuit will lock-out the loss of voltage signal and the blackout sequence will stop. In this case, the E1 motor loads must be manually started upon voltage restoration. This lock-out feature is installed to prevent excessive cycling of equipment and to remove additional battery loading during a sustained loss of power to E1.

BLACKOUT AND SI SEQUENCER FACTS

(Reference drawings 5379-3238 and 5379-3367)

3. Train "A" of Safeguards used as example:

4. The Blackout and SI sequencers are nothing more than initiating and timing relays. Once an initiating relay (E1VP for SI or E1LV for Blackout) has been energized, it closes contacts that energize digital timing relays and they begin timing. Once they have timed out (i.e. 20 seconds for Service Water Pump "A" for E-1, they close a contact that energizes an AC powered interposing relay. Interposing relays were added when the old sequence timing relays were replaced with the new digital timing relays. These new digital timing relays did not have the contact current carrying capability necessary to start loads. The AC powered interposing relays are powered from IB#7A for Train "A" and IB#3 for Train "B" of safeguards. When the interposing relay is energized it closes a contact in the control circuitry for the applicable load and causes it's breaker to close (i.e. SW Pump, etc.)

5. You must have DC power available for either the Blackout or SI sequences to start.

6. Blackout sequence starts as soon as power is lost to the emergency bus. By design, the EDG will get a start signal as soon as the emergency bus loses power and will energize the emergency bus in 10 seconds. The first blackout load is a Service Water pump at 20 seconds. Thus, the emergency bus should be energized by the EDG for 10 seconds before the SW pump gets a start signal. However, It doesn't matter if the emergency bus ever gets power back. The sequencer will run until it has completed one complete sequence and reset 2 seconds after the

AFW pump has been given a start signal (thus, 39.5 + 2 + 41.5 seconds). If there is no power on the bus, the blackout load breakers will close in and then trip back open because of no power.

7. Only two things will stop the blackout sequence:

    a. It has gone through one complete cycle.

    b. An SI signal occurs. If an SI signal occurs after the blackout sequencer has started, the blackout sequence stops wherever it is, and the SI sequence starts at the beginning.

8. For an SI sequence to start you must have an SI signal AND the emergency bus must have AC power. There is a difference between an SI signal and the SI Sequence. You can have an SI signal without the SI sequence starting because there is no power (e.g. EDG failed to start following a LOOP).

9. Miscellaneous facts:

    a. Anytime the EDG output breaker is closed, there is no under-voltage protection for the bus.

    b. There is a 2 second time delay for closing the EDG output breaker to allow for bus stripping prior to energizing the bus. This would come in to play if the EDG were sitting there at 480V waiting for paralleling operations (OST) and power was lost to the Emergency Bus. The output breaker would not close for 2 seconds to allow the bus to strip.

    c. Degraded Grid protection consists of 3 electronic relays arranged in a 2/3 logic. If the setpoint is reached for Degraded Grid (430VAC) the relays start timing. After 10 seconds the degraded grid protection trips the normal supply breaker to the emergency bus. Then, the two undervoltage relays would energize (setpoint 328VAC with no time delay). These two undervoltage relays energize other auxiliary relays that cause EDG start, bus stripping and start of the blackout sequence, etc.

Loss of Power to E2

Isolates E2 from rest of electrical plant by opening normal supply breaker (52/28B). It also strips all the breakers on E2 except for the MCC6 and MCC18 supply breakers. A block signal is placed on "B" auxiliary feed pump to prevent its breaker from closing due to a loss of feed pumps, low-low level in a steam generator, or AMSAC. The diesel generator is started and its output breaker closes when output voltage builds up (within

10 seconds). There is a two-second time delay that ensures that a voltage dip does not result in closure unless E2 is de-energized. The closing of the output breaker renews the block to ensure that "B" auxiliary feed pump starts at the proper time. When the loss of voltage on the E2 Bus is sensed, the timing for the blackout sequence starts. After twenty (20) seconds "C" Service Water Pump starts, a start signal is sent to both Service Water Booster Pumps, and both Service Water Booster Pumps start when the Service Water Header pressure builds up to 30 PSIG. After twenty (25) seconds "D" Service Water Pump starts and a start signal is sent to both Service Water Booster Pumps. Both Service Water pumps are started to ensure sufficient flow exist to the EDG if one of the pumps is inoperable. After thirty (30) seconds "C" Component Cooling Pump will start. After thirty-nine and one-half (39.5) seconds "B" Auxiliary Feed Pump will start and reset the blackout sequence after a 2 second time delay.

If sufficient voltage does not exist on E2 at the end of one cycle, the control circuit will lock-out the loss of voltage signal and the blackout sequence will stop. In this case, the E2 motor loads must be manually started upon voltage restoration. This lock-out feature is installed to prevent excessive cycling of equipment and to remove additional battery loading during a sustained loss of power to E2.

LOSS OF POWER TO THE EMERGENCY BUSES WITH AN SI SIGNAL PRESENT

The same action that occurred in Item 6.3.1 and 6.3.2 will take place except the blackout sequence loads will be blocked by the S-Signal. During the bus clearing 52/1B and 52/2B will also be opened on the E1 side and 52/15B and 52/32A and 52/16B will be open on the E2 side. This is a redundant measure to ensure the Emergency Busses are isolated from the rest of the plant electrical system. After the diesel generator output breakers are closed the safeguard sequence will take place. If a Containment Spray signal (P-signal) is present, the Containment Spray pumps will start anytime power is available to its respective emergency bus.

> NOTE
>
> "B" and "C" Charging Pumps will be tripped if a SI signal is present and power is being supplied by diesel generators.
>
> "B" and "C" Component Cooling Pumps will be tripped if a SI and a spray signal are present and power is being supplied by diesel generators.
>
> "B" and "C" Component Cooling Pumps will not start automatically on low pressure if the diesel generator is supplying its respective bus.

41. 026 G2.2.42 001

Which ONE (1) of the following combinations of Containment Heat removal equipment will require immediate entry into LCO 3.0.3?

A. Both HVH trains inoperable.

B. Three Containment HVH units inoperable.

C✓ Two Containment Spray Pumps inoperable.

D. One Containment Spray Pump and one HVH train inoperable.

The correct answer is C.

A) Incorrect. Plausible since two HVH trains are required per LCO 3.6.6. Condition D would apply for this and you would have 72 hours to correct. Also plausible since if you have two spray trains inoperable, you go to LCO 3.0.3.

B) Incorrect. Plausible because condition F of LCO 3.6.6 says that if any combination of three trains (spray and HVH) are inoperable, you go to LCO 3.0.3. This is not three trains, this is three HVH units which could be misunderstood as three trains. You are in condition D.

C) Correct. Two containment spray pumps inoperable puts you in condition F of LCO 3.6.6 which says IMMEDIATELY you will go to LCO 3.0.3.

D) Incorrect. Plausible because you would be in Conditions A and C of LCO 3.6.6. Students may think that there are not two seperate entry conditions for each train.

Question: 41
Tier/Group: 2/1
K/A Importance Rating: RO 3.9 SRO 4.6

K/A: 026 Containment Spray System (CSS)
G2.2: Equipment Control
G2.2.42: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Reference(s): ITS 3.6.6
Proposed References to be provided to applicants during examination: None
Learning Objective: CV Spray 12b
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7
Comments:

This question meets the KA because the candidate must recognize system parameters to determine entry into Technical Specifications.

Correct

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

Correct

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 5.	6 hours 36 hours
F. Two containment spray trains inoperable. OR Any combination of three or more trains inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

Incorrect

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours AND 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 84 hours
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO

(continued)

INCORRECT

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1	Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	E.2	Be in MODE 5.	36 hours
F. Two containment spray trains inoperable. <u>OR</u> Any combination of three or more trains inoperable.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

Incorrect

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two containment cooling trains inoperable.	D.1	Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1	Be in MODE 3.	6 hours
	AND		
	E.2	Be in MODE 5.	36 hours
F. Two containment spray trains inoperable. OR Any combination of three or more trains inoperable.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Verify each containment spray manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days

(continued)

Incorrect

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours <u>AND</u> 10 days from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 84 hours
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days <u>AND</u> 10 days from discovery of failure to meet the LCO

(continued)

42. 039 A4.04 001

Given the following plant conditions:

-A loss of offsite power and a reactor trip has occurred

-Two minutes later APP-007-F5, SD AFW PMP LO DISCH PRESS TRIP alarms and remains locked in

-The pump has been verified to be coasting down locally due to an overspeed trip

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

SDAFW Pump Steam Shutoff Valves, V1-8A, V1-8B, and V1-8C should be __(1)__ at this time. The BOP operator is required to __(2)__ closure of SDAFW Pump Discharge Valves, V2-14A, V2-14B, and V2-14C.

A. (1) open
   (2) verify automatic

B. (1) open
   (2) perform manual

C. (1) closed
   (2) verify automatic

D.✓ (1) closed
   (2) perform manual

The correct answer is D.

A. Incorrect. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. Plausible because on the overspeed trip the SDAFW Pump throttle valve closes to isolate the steam supply, so it is not necessary for the steam supply valves to close to stop the pump. The discharge valves do not close automatically, but APP-007-F5 directs manual closure. Plausible that an auto signal would close the valve to isolate a discharge line break that caused the low discharge pressure trip, since some valves auto close on the trip.

B. Incorrect. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. Plausible because on the overspeed trip the SDAFW Pump throttle valve closes to isolate the steam supply, so it is not necessary for the steam supply valves to close to stop the pump.

C. Incorrect. The discharge valves do not close automatically, but APP-007-F5 directs manual closure. Plausible that an auto signal would close the valve to isolate a discharge line break that caused the low discharge pressure trip, since some valves auto close on the trip.

D. Correct. V1-8A, B, and C close as a result of discharge pressure lowering to 650 psig during coastdown of the pump following the overspeed trip. APP-007-F5, directs manual closure of V2-14A, B, and C.

Question: 42
Tier/Group: 2/1
K/A Importance Rating: RO 3.8 SRO 3.9

K/A: 039 Main and Reheat Steam System (MRSS)
A4: Ability to manually operate and/or monitor in the control room:
A4.04: Emergency feedwater pump turbines

Reference(s): ST-042, APP-007
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 10 of AFW Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7, 10
Comments:

This question meets the K/A because the candidate is required to determine proper pump configuration in the control room for the conditions given.

A knurled knob on the speed governor can be used to control the speed of the turbine at a setpoint less than full speed. For normal automatic operation, the knob is set at the highest point so the turbine will operate at the maximum set speed during emergency conditions.

Instrument Air is used to force the governor valve from a fully open position when the SDAFW pump is shutdown. This is to prevent an overspeed condition during a pump start. When any of the V1-8 steam supply valves is approximately 95% open, either of two solenoid valves will isolate instrument air allowing the Woodward governor to control the pump speed. The solenoids can be independently tested with local pushbuttons (EC67828).

The SDAFW Pump may be manually tripped by pushing the "RED" trip button. Reset of the trip is by pushing in the trip lever elbow. (See Figure 11)

The SDAFW pump will trip on overspeed; it also trips on low discharge pressure to protect the pump from a loss of suction supply.

- 650 psig setpoint with a 30 second delay, 2/2 coincidence [EC 68931]
- Shuts steam inlet valves (V1-8A, B and C)

NOTE

Overspeed trip shuts V1-8A, B C due to 650 psig discharge pressure

ALARM

SD AFW PMP LO DISCH PRESS TRIP

NOTE: 1/2 pressure switches will cause alarm without pump trip.

Alarm may actuate during pump start until pump discharge pressure is greater than 650 psig. The trip is defeated for 30 seconds to allow pump to develop sufficient pressure. (NCR 242583)

AUTOMATIC ACTIONS

1. STEAM DRIVEN AFW PUMP trip (2/2 pressure switches)

CAUSE

1. Pump failure
2. Loss of suction supply
3. Local Pump trip (over speed trip device)

OBSERVATIONS

1. SDAFW Pump STEAM SHUTOFF valve position indication (V1-8A, V1-8B, V1-8C)
2. AFW Pump Discharge Pressure (PI-1421B)
3. Condensate Storage Tank Level (LI-1454A & LI-1454B)

ACTIONS

CK (✓)

1. **IF** the STEAM DRIVEN AFW PUMP has **NOT** tripped, **THEN DISPATCH** personnel to check pressure switch valve alignment. _____

CAUTION

Leaving the Steam Driven AFW Pump Discharge valves open could result in a Steamline •P Safety Injection signal.

2. **IF** the pump has tripped due to low discharge pressure, **THEN PERFORM** the following:
   1) **IF** the Steam Driven AFW Pump Discharge Valve(s) are open, **THEN CLOSE** the PUMP DISCH Valve(s). _____
      - V2-14A, PUMP DISCH
      - V2-14B, PUMP DISCH
      - V2-14C, PUMP DISCH
   2) **IF** needed to maintain Steam Generator Levels, **THEN START** AFW PUMP "A" or AFW PUMP "B". _____
   2) **IF** CST level is low, **THEN TRANSFER** AFW Suction Supply IAW the EOP network. _____
   3) **IF** CST level is **NOT** low, **THEN DISPATCH** personnel to check the following: _____
      - AFW Pump suction alignment
      - overspeed trip device
      - Indication of pump mechanical problem
      - Indication of a pipe break upstream of flow control valve.

43. 059 A2.12 001

Given the following plant conditions:

-The Plant is at 50% RTP perform post outage power ascension.

-FCV-478, FRV A, develops an air leak at the valve operator and the valve starts to slowly drift in the close direction

-The air leak increases slowly for 5 minutes and then the air line completely separates

EOP-E-0, REACTOR TRIP OR SAFETY INJECTION
AOP-010, MAIN FEEDWATER/CONDENSATE MALFUNCTION

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

Prior to the line separation, FCV-478 controller output will ___(1)___. After the line separates the crew will ___(2)___ IAW AOP-010.

A. (1) rise
(2) stabilize "A" S/G Level using the FRV Bypass Valve

B. (1) lower
(2) stabilize "A" S/G Level using the FRV Bypass Valve

C.✓ (1) rise
(2) trip the Reactor and Go To EOP-E-0

D. (1) lower
(2) trip the Reactor and Go To EOP-E-0

The correct answer is C.

A. Incorrect. The first part of the distractor is correct. The second half of the distractor is plausible because attempting to stabilize S/G level with the FRV and FRV Bypass Valve is part of AOP-010 immediate actions. However, with the current power level and the FRV fully closed the S/G level cannot be stabilized.

B. Incorrect. The controller output lowering is incorrect. Plausible since several of our controllers are reverse acting, however, this one is not. If valve is drifting closed then feed flow and S/G level lower so the SGLC system has to compensate by sending a larger signal to the valve. Plausible if candidate does not tie the increasing air leak to increased drift. The second half of the distractor is plausible because attempting to stabilize S/G level with the FRV and FRV Bypass Valve is part of AOP-010 immediate actions. However, with the current power level and the FRV fully closed the S/G level cannot be stabilized.

C. Correct. Since for conditions given the air leak is already causing the FRV to drift in the closed direction, the valve will continue to drift as the air leak progresses. This causes S/G level to start lowering and the SGLC system raise the output of the controller to attempt to restore S/G Level. After the air line separates the FRV fails closed, resulting in a loss of ability to control level in the affected S/G. AOP-010 calls for a reactor trip for this condition.

D. Incorrect. The controller output lowering is incorrect. Plausible since several of our controllers are reverse acting, however, this one is not. If valve is drifting closed then feed flow and S/G level lower so the SGLC system has to compensate by sending a larger signal to the valve Plausible if candidate does not tie the increasing air leak to increased drift.

Question: 43
Tier/Group: 2/1
K/A Importance Rating: RO 3.1 SRO 3.4

K/A: 059 Main Feedwater (MFW) System
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunction or operations:
A2.12: Failure of feedwater regulating valves

Reference(s): AOP-010
Proposed References to be provided to applicants during examination: None
Learning Objective: AOP-010 Obj. 4
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.4, 10
Comments:

This question meets the K/A becasuse the candidate must predict the effect the malfunction has on the system and then select the procedural step to mitigate the problem.

Correct

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 <u>OR</u> 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. <u>IF</u> unable to control S/G level, <u>THEN</u> trip the Reactor <u>AND</u> Go To EOP-E-0, REACTOR TRIP or SAFETY INJECTION. f. Go To Step 37
* 2.	Check Reactor Trip Setpoint - BEING APPROACHED	<u>IF</u> a Reactor Trip Setpoint is approached, <u>THEN</u> 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	

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 1 is immediate action step.

1. Check Feedwater Regulating Valves - OPERATING PROPERLY (MANUAL OR AUTO):

   - FCV-478, FRV "A"
   - FCV-488, FRV "B"
   - FCV-498, FRV "C"

   Response Not Obtained: 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

This cannot Be Done Due To The initial Power Level. FRV Bypasses are only good up To a max Pwr of 20%.

* 2. Check Reactor Trip Setpoint - BEING APPROACHED

   Response Not Obtained: 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

NOTE: Step 8.4.21 is a continuous action step that should be performed whenever plant conditions require Feedwater flow through the FRVs and conditions are suitable for automatic S/G water level control.

Feedwater Regulating Valves should be transferred to automatic control one at a time.

FCV-1446, CONDENSATE RECIRC, is controlled by FS-1446, COND PMP RECIRC VLV FLOW SWITCH. FS-1446 is set to open FCV-1446 at a flow of 1050 gpm with the valve closing at a Condensate System flow of 4200 gpm flow through the GS Condenser and SGBD Heat Exchangers. (ESR 00-00208)

The Push Button to reset FS-1446 is located in the same enclosure as FS-1446. This enclosure is located approximately 15 feet northwest of FCV-1446 on a concrete column.

21. **WHEN** Reactor Power is at 15% to 20%, **OR** the Feedwater Regulating Bypass Valves are at 60% to 90% demand signal, **THEN SHIFT** each Feedwater Regulating Valve to AUTO as follows:

    a. **CHECK** CLOSED FCV-1446, CONDENSER RECIRC. ____

    b. **IF** FCV-1446 does not indicate shut, **THEN PERFORM** the following:

        (1) **DEPRESS AND HOLD** the FS-1446 Push Button. ____

        (2) **WHEN** FCV-1446 indicates full shut, **THEN** release the FS-1446 Push Button. ____

CAUTION

Shutting either C-18A, FCV-1446 INLET, or C-18B, FCV-1446 OUTLET, could cause a running Condensate Pump to overheat on a low flow condition if unit load is reduced prior to correcting the problem with FCV-1446.

        (3) **IF** FCV-1446 is failed open **OR** reopens when it should stay closed, **THEN SHUT** either

- C-18A ____

OR

- C-18B ____

44. 061 K6.02 001

Given the following plant conditions:

-The plant is in Mode 3

-The SDAFW Pump is under clearance

-MDAFW Pump B is running

Subsequently,

-A line break occurs between FCV-1425, MDAFW PUMP B FCV and V2-20B, AFW HDR SECTION valve

-MDAFW Pump B trips on overcurrent due to runout through line break

-The break has been isolated using single valve isolation

Which ONE (1) of the following identifies the S/Gs available to be fed by the remaining AFW Pump?

A. S/G A ONLY

B. S/G B ONLY

C. S/Gs A, B, and C

D✓ S/Gs A and B ONLY

The correct answer is D.

A. Incorrect. May also feed B SG. Plausible because the candidate must have knowledge of system configuration and break location to eliminate the answer as a possible selection.

B. Incorrect. May also feed A SG. Plausible because the candidate must have knowledge of system configuration and break location to eliminate the answer as a possible selection.

C. Incorrect. Break location would require isolating flow to C SG. Plausible because the candidate must have knowledge of system configuration and break location to to eliminate the answer as a possible selection.

D. Correct. Location of break and valves closed to isolate the leak allow flow to A and B S/Gs because the AFW lines are cross connected upstream of the break location.

Question: 44
Tier/Group: 2/1
K/A Importance Rating: RO 2.6 SRO 2.7

K/A: 061 Auxiliary/Emergency Feedwater (AFW) System
K6: Knowledge of the effect that a loss or malfunction of the following will have on the AFW Components:
K6.02: Pumps

Reference(s): ST-042 AFW
Proposed References to be provided to applicants during examination: None
Learning Objective: AFW003
Question Source: RNP Bank
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.4, 7
Comments:

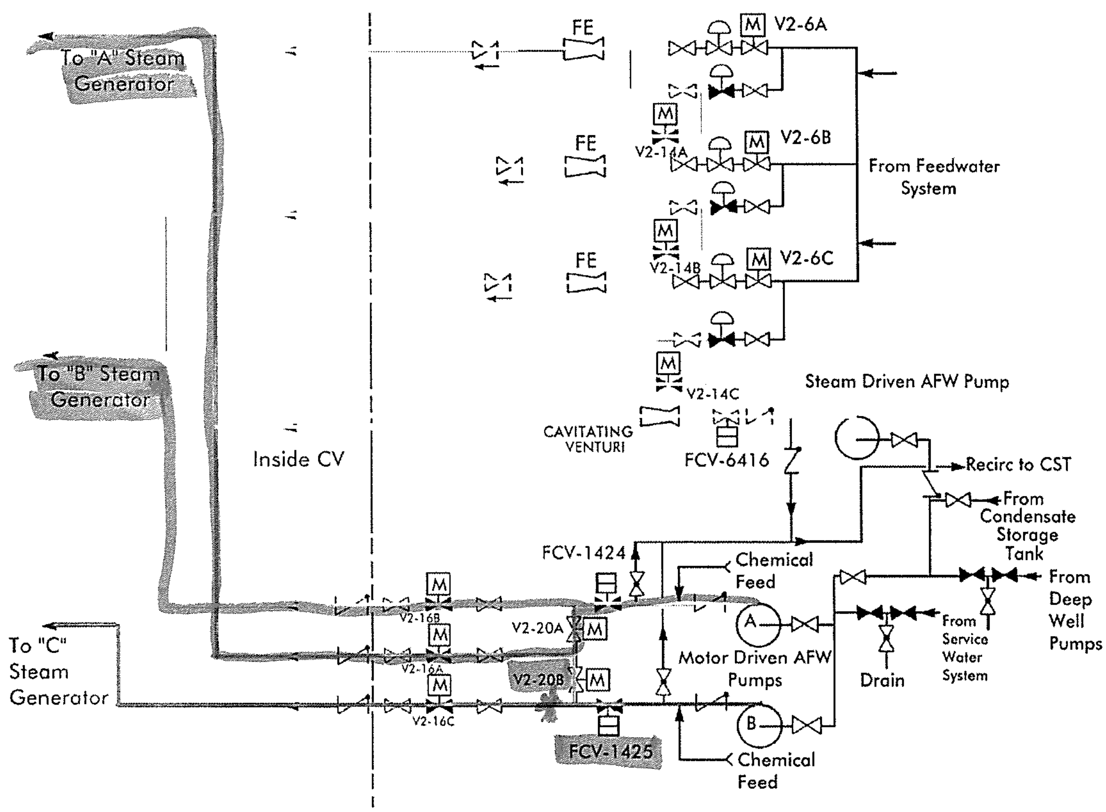
This question meets the K/A because the candidate must know the AFW drawing and the effect a break in the piping would have on which S/G's could still receive flow.

TECHNICAL SPECIFICATIONS

LCO 3.3.8	AFW System Instrumentation
LCO 3.7.4	AFW System
LCO 3.7.5	Condensate Storage Tank

Each of the three flowpaths from the MDAFW pumps constitute a flowpath. The three flowpaths from the SDAFW pumps are counted as only one flowpath.

Figure 18 AFW Flowpaths



LCO 3.7.4 requires that the SDAFW pump be operable prior to entry into mode 3.

SRs 3.7.4.2, 3.7.4.4, and 3.7.4.5 include notes that allow these SRs to be completed after entry into the mode of applicability. As stated in the basis for TS 3.7.4, this allows the

45. 062 K3.01 001

Given the following plant conditions:

-The Plant is at 100% RTP

-A fault has isolated the Startup Transformer

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

This results in a loss of ___(1)___. Emergency Diesel Generator ___(2)___ starts and loads.

A. (1) Spent Fuel Pit Cooling Pump A and Rod Drive MG Set B
(2) A

B✓ (1) Spent Fuel Pit Cooling Pump A and Rod Drive MG Set B
(2) B

C. (1) Spent Fuel Pit Cooling Pump B and PZR Heater Back-up Group A
(2) A

D. (1) Spent Fuel Pit Cooling Pump B and PZR Heater Back-up Group A
(2) B

The correct answer is B.

A. Incorrect. E-2 is de-energized, starting and loading "B" EDG. Plausible because the Robinson bus and load numbering system mixes A, B , C, and D differently from system to system. Case in point, the loads mentioned in the selections have both A and B designations but in both cases the loads are powered from the same bus.

B. Correct. Loss of the Startup Transformer at 100% power only affects the 4KV Bus 3, 480V Bus 3, E-2 and the various loads powered from those Buses. Loss of E-2 starts and loads B EDG. The loads listed are powered from 480V Bus 3.

C. Incorrect. The bus that supplies the loads listed in this selection is ultamately powered from the Unit Auxiliary Transformer which is unaffected by the conditions given in the question. E-2 is de-energized, starting and loading "B" EDG. Plausible because the Robinson bus and load numbering system mixes A, B , C, and D differently from system to system. Case in point, the loads mentioned in the selections have both A and B designations but in both cases the loads are powered from the same bus.

D. Incorrect. The bus that supplies the loads listed in this selection is ultamately powered from the Unit Auxiliary Transformer which is unaffected by the conditions given in the question. Plausible because the Robinson bus and load numbering system mixes A, B , C, and D differently from system to system. Case in point, the loads mentioned in the selections have both A and B designations but in both cases the loads are powered from the same bus.

Question: 45
Tier/Group: 2/1
K/A Importance Rating: RO 3.5 SRO 3.9

K/A: 062 A.C. Electrical Distribution
K3: Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following:
K3.01: Major system loads

Reference(s): EDP-002, ST-039
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 9 of 4Kv and Objective 6a of Spent Fuel Pool Cooling
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 41.7
Comments: The question meets the K/A because the candidate needs to know how a loss of the Startup Transformer affects major system loads.

Correct

4.0 **480V-3**

480V-3 POWER SUPPLY: NORMAL - 4160V BUS 3 (52/15) LOCATION: 4160V SWITCHGEAR ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
13A	**SPARE** (Reserved by EC 53926) N/A	1796	52/13A
13B	**FEED TO MCC-4** MCC-4	N/A	52/13B
13C	**SPENT FUEL PIT COOLING PUMP A** SFPC-PMP-A	227	52/13C
14A	**ROD DRIVE MOTOR GENERATOR SET B** MG-SET-B	73	52/14A
14B	**BLANK** N/A	N/A	N/A
14C	**MOTOR DRIVEN FIRE PUMP** MTR-FIRE-PMP	585	52/14C
15A	**CONDENSER VACUUM PUMP B (*)** VCM-PMP-B	759	52/15A
15B	**FEED TO 480V BUS 3** 480V-3	898B	52/15B
15C	**BLANK** N/A	N/A	N/A
16A	**PT'S & METERING EQUIPMENT (**)** N/A	N/A	N/A
16B	**STATION SERVICE TRANSFORMER 2C TO 480V BUS 3 & 480V BUS DS** 480V-3, 480V-DS	899	52/16B

* Compartment 15A also contains one lockout reset button.

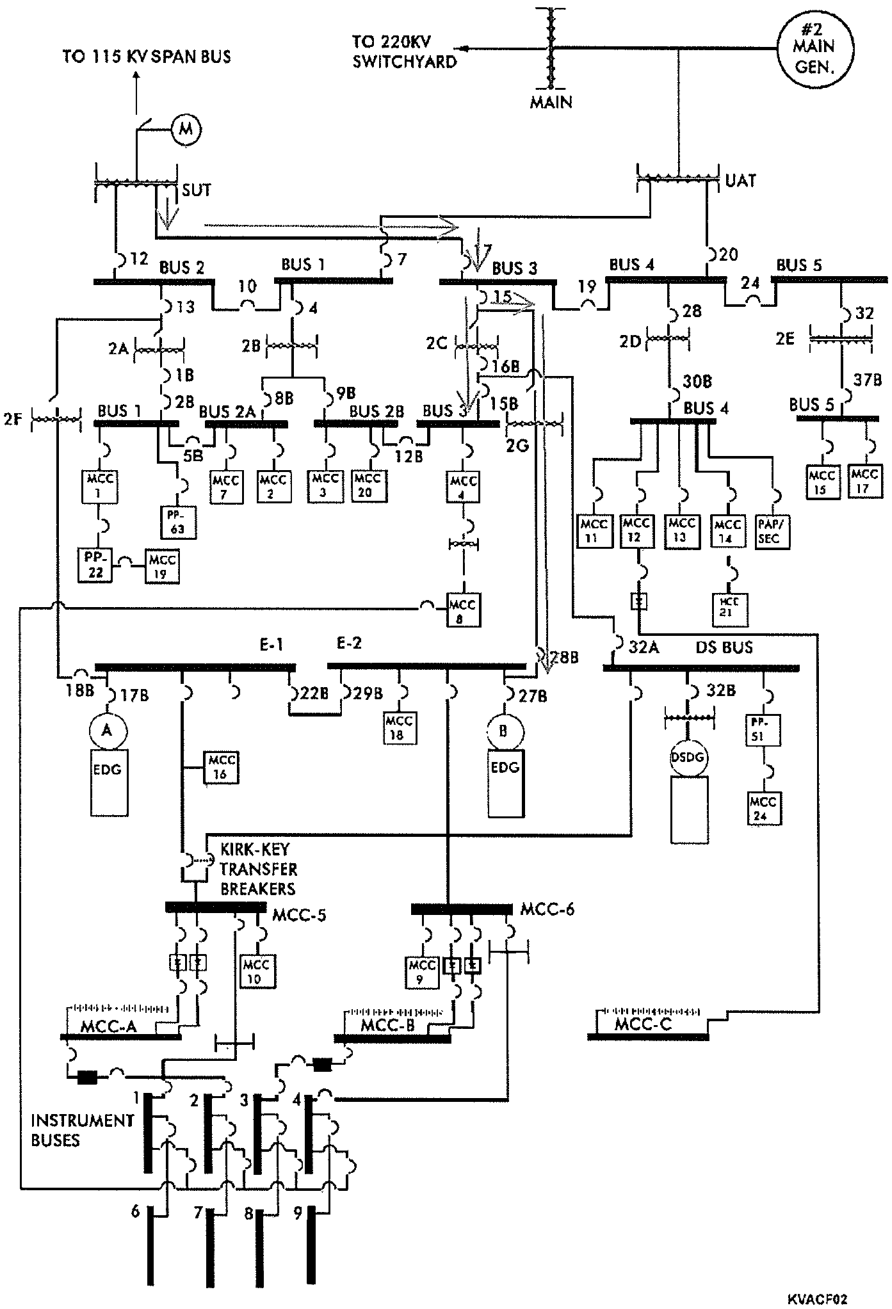
** Compartment 16A also contains ground relay, one undervoltage relay, one AC voltmeter, one voltmeter selector switch, one ground indication, and one ground detection reset.

Correct

Figure 10 Electrical Distribution System KVAC

At 100% power only 4KV bus 3 is receiving power from the SUT

Loss of the SUT de-energizes 4KV bus 3, 480v b3, and 480v bus E-2 which will start the "B" EDG



Incorrect

1.0 480V-1

480V-1 POWER SUPPLY: NORMAL - 4160V BUS 2 (52/13)LOCATION: 4160V SWITCHGEAR ROOM			
CMPT NO.	**LOAD TITLE** EDBS LOAD TAG NO.	CWD NO.	BKR EDBS NO.
1A	**PT'S & METERING EQUIPMENT (*)** N/A	N/A	N/A
1B	**STATION SERVICE TRANSFORMER 2A TO 480V BUS 1** 480V-1	894	52/1B
2A	**BLANK** N/A	N/A	N/A
2B	**FEED TO 480V BUS 1** 480V-1	893	52/2B
2C	**PRESSURIZER HEATER BACK-UP GROUP A** BACKUP-GP-A	131	52/2C
3A	**FEED TO MCC-1** MCC-1	N/A	52/3A
3B	**CV POLAR CRANE BACKUP POWER SUPPLY** POLAR-CRN	1766	52/3B
3C	**FEED TO PP-63** PP-63	N/A	52/3C
4A	**BLANK** N/A	N/A	N/A
4B	**CONSTRUCTION FACILITIES MAIN DISCONNECT** CONST-FAC-DISC-SW	N/A	52/4B
4C	**SPENT FUEL PIT COOLING PUMP B** SFPC-PMP-B	225	52/4C
5A	**METERING EQUIPMENT (**)** N/A	N/A	N/A
5B	**480V BUS 1-2A TIE** 480V-1, 2A	935	52/5B
5C	**CONDENSER VACUUM PUMP A** VCM-PMP-A	757	52/5C

* Compartment 1A also contains one ground relay, one undervoltage relay, one AC voltmeter, one voltmeter selector switch, one ground indication, and one ground detection reset.

** Compartment 5A also contains the Spent Fuel Pit "B" total run time meter and the Vacuum Pump "A" total run time meter.

46. 063 A1.01 002

Given the following plant conditions:

-A LOCA and a Loss of Offsite Power have occurred

-Both battery chargers that were in service are tripped

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

Given the accident above, "A" and "B" Batteries are designed to last for a **MAXIMUM** of __(1)__. Assuming battery load remains constant, battery current will __(2)__ as terminal voltage lowers.

A. (1) 1 hour
(2) lower

B✓ (1) 1 hour
(2) rise

C. (1) 30 minutes
(2) lower

D. (1) 30 minutes
(2) rise

The correct answer is B

A) Incorrect. 1 hour is correct. During this accident, which is the worst case loading profile for our batteries, the batteries are designed to last 1 hour. Battery current will lower is incorrect. Plausible if the candidate misuses the equation P=VI. If the candidate use V=IP, which is similar to V=IR, they would say that current has to lower.

B) Correct. 1 hour is correct. During this accident, which is the worst case loading profile for our batteries, the batteries are designed to last 1 hour. Current will rise is correct. P=VI, in which case Power(P) will remain constant and Voltage (V) will rise. This will mean that Current (I) must rise to maintain the same Power.

C) Incorrect. 30 minutes is incorrect. Plausible because this is the time limit that an operator has to restart the battery chargers if they have tripped, in which they have in this case. Battery current will lower is incorrect. Plausible if the candidate misuses the equation P=VI. If the candidate use V=IP, which is similar to V=IR, they would say that current has to lower.

D) Incorrect. 30 minutes is incorrect. Plausible because this is the time limit that an operator has to restart the battery chargers if they have tripped, in which they have in this case. Current will rise is correct. P=VI, in which case Power(P) will remain constant and Voltage (V) will rise. This will mean that Current (I) must rise to maintain the same Power.

Question: 46
Tier/Group: 2/1
K/A Importance Rating: RO 2.5 SRO 3.3
K/A: 063 DC Electrical Distribution System
A1: Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including:
A1.01: Battery capacity as it is affected by discharge rate
Reference(s): Sim/Plant design, EPP-1 Basis document, Step 36
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 9 of EPP-1 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.5 / 45.5)
Comments:

This question meets the K/A because the candidate must predict what will happen to current using P=VI equation. This affects the batteries discharge rate.

Correct

8.3.2 DC Power System (125 Volt)

As shown in Figure 8.3.1-5, the DC power system consists of five 125 V batteries, each with its' own battery charger(s) and DC buses. Two of the batteries are safety-related. The battery chargers supply the normal DC loads as well as maintaining proper charges on the batteries. Each charger has the capacity to supply all normal DC loads and maintain the battery fully charged. For each safety-related station battery, there are two safety-related battery chargers. One battery charger supplies the normal DC loads while the other is providing 100% back-up capability. Only one safety-related battery charger per station battery will be on-line at a time. The DC power system is shown in Figure 8.3.1-5.

Each of the two safety-related station batteries is sized to carry its expected shutdown loads following a plant trip and a loss of all AC power for a period of 1 hour without battery terminal voltage falling below minimum allowable voltage. Shutdown loads with their approximate operating times on each safety-related battery are listed in Table 8.3.2-1.

Each of the four safety-related battery chargers have been sized to charge it's partially discharged battery within 24 hr while carrying its normal load.

Cells in the "A" battery are type NCN-15 with a capacity of 1070 ampere hours (based on an 8 hour discharge to 1.75 volts/cell). The "A" bank is composed of 60 cells of the lead calcium type. Cells in the "B" battery are type KCR-11 with a capacity of 410 ampere hours (based on an 8 hour discharge to 1.75 volts/cell). It is composed of 60 cells of the lead calcium type. The battery capacities are 525 A-Hr and 204 A-Hr for the NCN-15 and KCR-11 batteries respectively for a 1 hour discharge to 1.75 volts/cell. The actual designed final discharge voltage following an emergency discharge will be based on required equipment voltages.

The safety-related batteries and equipment are separated physically in the plant. The existing configuration provides adequate separation with equipment for one division on the north side of the battery room and the other division on the south side. The fire hazards analysis for the battery room is contained in Section 9.5.1.

The "C" battery (non-safety related) is located on the auxiliary building roof above the battery room. Cells in the "C" battery are rated at 1800 ampere hours minimum (based on an 8 hour discharge to 1.75 volts/cell). The "C" bank is composed of 60 cells of the lead calcium type.

The following non-safety related loads are supplied by the "C" battery:

1. Turbine emergency bearing oil pump (50 HP), and

2. air side seal oil backup pump (10 HP).

The Dedicated Shutdown (DS) Battery (Augmented Q) is located in the 4kV Switchgear Room. The DS Battery consists of 97 nickel cadmium type cells with a 1 hour duty cycle. The DS Battery has a minimum discharge voltage rating of 105 Volts DC.

The Dedicated Shutdown Diesel Generator (DSDG) Battery (Augmented Q) is located in an enclosed panel next to the DSDG Building. The DSDG Battery consists of 92 nickel cadmium type cells with a 1 hour duty cycle. The DS Battery has a minimum discharge voltage rating of 105 Volts DC (based on 1.14 volts/cell as final voltage).

Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Attachment 1

Auto Action Verification

(Page 8 of 8)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
15.	**Check Battery Chargers - ENERGIZED** • APP-036-D1, BATT CHARGER A/A-1 TROUBLE alarm - EXTINGUISHED • APP-036-D2, BATT CHARGER B/B-1 TROUBLE alarm - EXTINGUISHED	Restart battery chargers within 30 minutes of power loss using OP-601, DC Supply System. (46 KW each)
16.	**Stop R-11/12 Sample Pump**	
17.	**Locally Reset And Load Instrument Air Compressor(s) As Necessary (38 KW each):** • Compressor A (MCC-5 CMPT 7M) • Compressor B (MCC-6 CMPT 3G)	
18.	**Perform Crew Update To Include The Following:** • Attachment completion • Manual actions taken • Failed Equipment status • SW status per Step 7.c: • If applicable, perform Supplement M, Component Alignment For Loss Of SW To Turbine Building, as time permits	

- END -

47. 064 G2.1.27 001

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

The purpose of the Emergency Diesels is to provide emergency power in the event of a loss of __(1)__ . The design time to energize the associated bus is within __(2)__ seconds.

A.✓ (1) offsite power
(2) 10

B. (1) offsite power
(2) 15

C. (1) all A/C power
(2) 10

D. (1) all A/C power
(2) 15

The correct answer is A

A) Correct. The purpose of the EDG's is to provide the emergency busses (E1 and E2) with power in the event of a loss of power to these busses. The design of the EDG is to be at rated speed and energize it's associated bus within 10 seconds.

B) Incorrect. The purpose is correct. 15 seconds is incorrect. Plausible because this is the time associated with the second piece of equipment that starts on the Safety Injection Sequencer.

C) Incorrect. The purpose is incorrect. Plausible since this is the purpose of the DSDG. The time is correct.

D) Incorrect. The purpose is incorrect. Plausible since this is the purpose of the DSDG. 15 seconds is incorrect. Plausible because this is the time associated with the second piece of equipment that starts on the Safety Injection Sequencer.

Question: 47
Tier/Group: 1/1
K/A Importance Rating: RO 3.9 SRO 4.0
K/A: 064 Emergency Diesel Generator
G 2.1.27: Knowledge of system purpose and/or function.
Reference(s): Sim/Plant design, Student Text for EDG
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 1 of EDG lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.7)
Comments:

This question meets the K/A because the student must know the system purpose and a portion of its function(time to close in on its de-energized bus).

CHAPTER 2

Correct

General Description

System Purpose

The EDGs provide the Emergency busses (E1 and E2) with power in the event of a loss of power to these busses. In the event of an automatic safeguards actuation the EDGs will startup, ready to supply power to the Emergency busses if necessary. These Emergency bus power supplies are essential for the safe shutdown of the plant.

Design Basis

The EDG system provides an Emergency source of AC electrical power to the On-site Emergency AC Power subsystem, as required, for those events where off-site power is assumed not to be available.

Each EDG Unit shall auto start upon detection of undervoltage on that EDG's respective Emergency bus, except when the Local/Remote control switch is in the LOCAL position. Once the EDG reaches design speed, the system will then close the EDG output breaker and assume the load on its respective bus.

Both EDGs shall auto start upon initiation of SI, except when the Local/Remote control switch is in the LOCAL position. Once the EDG reaches design speed, the system will run but not pickup load unless power is lost to its respective bus.

The EDG system shall provide adequate independency, redundancy, capacity, and testability to permit the functioning of the ESFs and protection systems required to avoid undue risk to the health and safety of the public. The EDG system shall provide this capacity assuming a single failure of a single active component.

CHAPTER 1

Correct

Introduction

Engineered Safety Feature (ESF) equipment circuits are connected to the 480 Volt Emergency Busses (E1 and E2). The normal source of power for Bus E1 is the Unit 2 Generator through the Unit 2 Auxiliary Transformer, 4160 Volt Bus 2, and the Station Service Transformer 2F. The normal source of power for Bus E2 is the 115 KV System through the Unit 2 Startup Transformer and Station Service Transformer 2G. Emergency Diesel Generator (EDG) A is connected to Bus E1 and EDG B to Bus E2. A diesel generator will be automatically started and connected to its associated bus if voltage on its associated bus is lost. The diesel generators will also automatically start when a Safety Injection (SI) signal is received.

During normal full power operation, these two diesel generators are in a standby mode to respond to a loss of voltage on their respective bus.

The 480 Volt system has two levels of protection for an undervoltage condition. Either of these conditions will trip the 480 Volt Bus E1 and/or E2 Normal Incoming Breaker (off-site system) and initiate the start and operation of the EDGs.

The first level of protection (loss of voltage) occurs at 328 volts + or - 10% with a time delay of less than 1.0 seconds (at zero voltage). The second level of protection (degraded voltage) occurs at 430 volts + or - 4 volts with a time delay of 10.0 seconds + or - 0.5 seconds. The EDGs are designed to reach rated speed and energize the associated bus within 10 seconds of sensing this loss of voltage at their respective bus. Refer to Technical Specification LCOs 3.3.5 & 3.8.1, and SRs 3.3.5.2, 3.8.1.7, 3.8.1.9, 3.8.1.10, 3.8.1.13, 3.8.1.15, & 3.8.1.17.

2

General Description

System Purpose

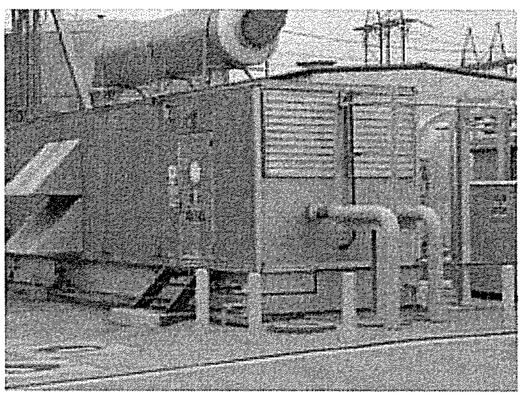
Incorrect

DEDICATED SHUTDOWN DIESEL GENERATOR

The DSDG provides a source of electrical power to bring the plant safely to a hot shutdown condition in the event of a fire in the Control Room, Cable Spread Room, E-1/E-2 Room, Battery Room, or the first floor of the Auxiliary Building.

DSDG will also automatically provide an emergency source of electrical power in the event of a simultaneous loss of all off site power and both Emergency Diesel Generators (EDGs).

Figure 1 DSDG Enclosure



TSC/EOF/SECURITY DIESEL GENERATOR

The TSC/EOF/SECURITY Diesel Generator provides an emergency source of electrical power in the event of a loss of all normal power to either the TSC/EOF or SECURITY.

Design Basis

DSDG AND THE DEDICATED SHUTDOWN ELECTRICAL SYSTEM

The DSDG is one component of the Dedicated Shutdown Electrical System. The system was added due to Fire Protection Modifications required by 10CFR50 Appendix R. This system may be supplied with power from 4KV bus 3 via station service transformer 2C or DS diesel generator. The DS Electrical System provides power for components necessary for safe plant shutdown (480Vac bus DS located in the 4160 Switchgear Room)

1. Service Water Pump D

Incorrect

SYSTEM OPERATION

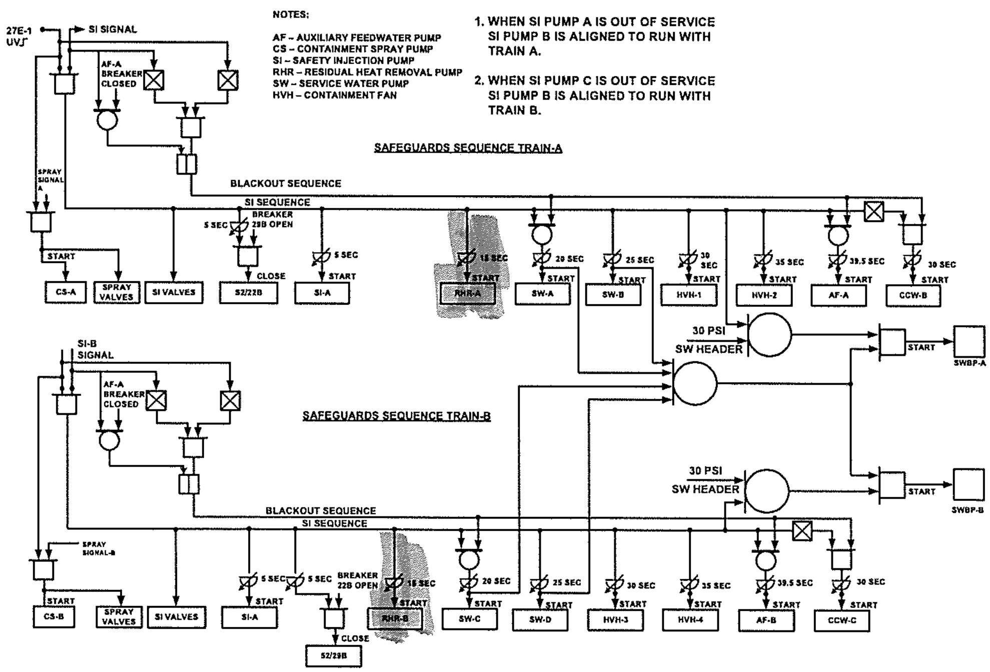
NOTE

During normal operation, this system stays in a standby status. Therefore, the only operations covered here will be emergencies.

Safety Injection Actuation

After an SI signal (also called "S" signal) is received, the signal is automatically locked in until it is reset. Then these evolutions will follow: (Note that the signal cannot be reset until at least two minutes have elapsed from the time of receipt.)

Figure 18 Safety Injection Sequence



REACTOR TRIP

The under voltage coil on the reactor trip breaker will de-energize and the shunt trip coils will be energized causing the breakers to open.

48. 064 K2.01 001

De-energizing which ONE (1) of the following Motor Control Centers will cause a loss of Diesel Air Compressor "A"?

A.✔ MCC-5

B. MCC-6

C. MCC-9

D. MCC-10

The correct answer is A.

A. Correct. Diesel Air Compressor "A" is powered from MCC-5 compartment 3D.

B. Incorrect. Plausible because Diesel Air Compressor "B" is powered from MCC-6.

C. Incorrect. Plausible because MCC-9 is powered from MCC-6.

D. Incorrect. Plausible because MCC-10 is powered from MCC-5.

Question: 48
Tier/Group: 2/1
K/A Importance Rating: RO 2.7 SRO 3.2

K/A: 064 Emergency Diesel Generators (ED/G)
K2: Knowledge of bus power supplies to the following:
K2.01: Air Compressor

Reference(s): EDP-003, ST-016 480/120 VAC
Proposed References to be provided to applicants during examination: None
Learning Objective: EDG006a
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.7
Comments: This question meets the K/A because the candidate must possess knowledge of the power supply to emergency diesel air compressors.

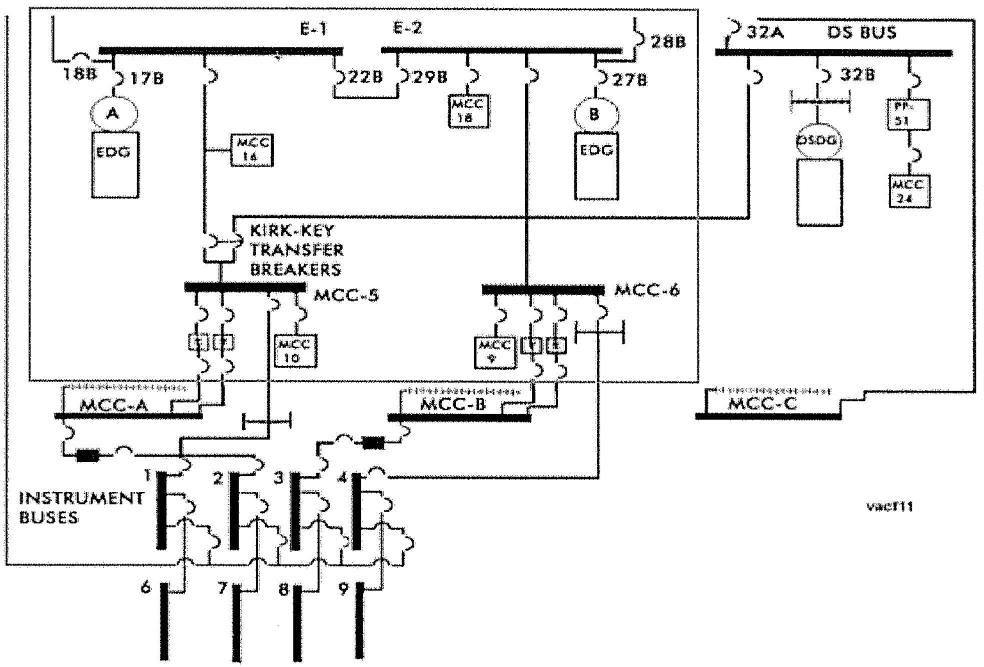
Correct

6.0 MCC-5

MCC-5 POWER SUPPLY: 480V BUS E-1 (52/21A) LOCATION: AUX BLDG HALLWAY			
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
1A	BLANK N/A	N/A	N/A
1CL	FUSES FOR LEAKOFF COLLECTION TANK PUMPS A & B LKOF-PMP-A,B	1759	N/A
1CR	LEAKOFF COLLECTION TANK PUMPS A & B LKOF-PMP-A, B	1759	52/MCC-5(1CR)
1F	V6-12D, NORTH SW HEADER ISOLATION (ALT POWER) V6-12D	839B	52/MCC-5(1F)
1J	FP-256, RCP SPRINKLER & CV FIRE HOSE ISOLATION FP-256	750	52/MCC-5(1J)
1M	FP-248, ELECTRICAL PENETRATION SPRINKLER ISOLATION FP-248	748	52/MCC-5(1M)
2C	SI-878A, SI PUMPS A & B DISCHARGE CROSS CONNECT SI-878A	257	52/MCC-5(2C)
2F	V6-33B, CV RECIRC COOLER HVH-2 SW INLET V6-33B	500	52/MCC-5(2F)
2J	V6-33A, CV RECIRC COOLER HVH-1 SW INLET V6-33A	501	52/MCC-5(2J)
2M	V6-33E, CV RECIRC COOLER HVH-4 SW SELECTIVE INLET V6-33E	504	52/MCC-5(2M)
3B	BORIC ACID TANK A HEATERS BA-TNK-A-HTR-A, B	188	52/MCC-5(3B)
3D	EDG A AIR COMPRESSOR DG-A-AIR-CMP	946	52/MCC-5(3D)
3F	RELAY ROOM AIR CONDITIONING UNIT, HVA-2 HVA-2	567	52/MCC-5(3F)

Incorrect

Figure 4 AC Distribution to DC Busses and Instrument Busses



120VAC INSTRUMENT BUS SYSTEM

The 120Vac instrumentation supply is split into 8 panels, known as Instrument Buses (IBs).

IBs 1, 2, 3, and 4 are 120Vac, single-phase (ø) panels with single pole 'EB' or 'EHB' frame 100 amp supply breakers at the panel and contain single pole 'EB' or 'EHB' frame, 30 amp feeder breakers.

Each IB has a normal and alternate power supply with its supply breakers mechanically interlocked such that only one breaker may be closed for incoming feeds.

IBs 2 and 3 are normally fed from 125Vdc motor control center (MCC) -A and MCC-B respectively via their single-Ø fixed frequency inverters. IBs 1 and 4 are normally fed from 480Vac MCC-5 and MCC-6 respectively via their 480/120Vac constant voltage transformers. The alternate power supply for IBs 1, 2, 3 & 4 is 120Vac MCC-8.

IBs 6, 7, 8, & 9 are fed from IBs 1, 2, 3, & 4, respectively. Hagan channels I, II, III, IV are powered from IB 1 through 4 respectively. Instrument and protection channels are color coded; Red for IB 1, White for IB 2, Blue for IB 3 and Yellow for IB 4.

49. 073 A4.01 001

Given the following plant conditions:

-The reactor is at 100% RTP

-The crew is performing a containment vacuum relief

-R-12, CV AIR OR PLANT STACK, NOBLE GAS, alarms

V12-12, CV VAC RELIEF
V12-13, CV VAC RELIEF
APP-036-D7, AREA MONITOR HI RAD
APP-036-D8, PROCESS MONITOR HI RAD

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

The crew would expect to see ___(1)___ annunciator flashing. They would also see ___(2)___ go shut.

A. (1) APP-036-D7
(2) V12-12 & V12-13 only

B. (1) APP-036-D8
(2) V12-12 & V12-13 only

C. (1) APP-036-D7
(2) V12-12, V12-13 & the CV Intake Damper

D✓ (1) APP-036-D8
(2) V12-12, V12-13 & the CV Intake Damper

The correct answer is D

A) Incorrect. APP-036-D7 is incorrect. Plausible if they think that R-12 is an area monitor instead of a process monitor. V12-12 & V12-13 only is incorrect Plausible since R-12 alarming does these valves, however, the CV intake damper also shuts.

B) Incorrect. APP-036-D8 is the correct alarm since R-12 is a process monitor. V12-12 & V12-13 only is incorrect Plausible since R-12 alarming does these valves, however, the CV intake damper also shuts.

C) Incorrect. APP-036-D7 is incorrect. Plausible if they think that R-12 is an area monitor instead of a process monitor. V12-12, V12-13 and the CV intake damper is correct. These three valves are open during a vacuum relief. When V12-12 and V12-13 shut, this shuts the CV Intake Damper.

D) Correct. APP-036-D8 is the correct alarm since R-12 is a process monitor. V12-12, V12-13 and the CV intake damper is correct. These three valves are open during a vacuum relief. When V12-12 and V12-13 shut, this shuts the CV Intake Damper.

Question: 49
Tier/Group: 2/1
K/A Importance Rating: RO 3.9 SRO 3.9
K/A: 073 Process Radiation Monitoring (PRM) System
A4: Ability to manually operate and/or monitor in the control room:
A4.01: Effluent release
Reference(s): SIM/Plant Design, APP-036-D8, OP-921
Proposed References to be provided to applicants during examination: None
Learning Objective: LP-019 RMS Obj. 8
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: 41.7 / 45.5 to 45.8
Comments:

This question meets the K/A because the candidate must know which pieces of equipment and alarms to monitor during an effluent release.

Correct

ALARM

PROCESS MONITOR HI RAD *** WILL REFLASH ***

AUTOMATIC ACTIONS/CAUSES

CHANNEL	AUTO ACTION	CAUSE
R-11 R-12	1) HVE-1A **AND** HVE-1B stop. 2) V12-6 closes 3) V12-7 closes 4) V12-8 closes 5) V12-9 closes 6) V12-10 closes 7) V12-11 closes 8) V12-12 closes 9) V12-13 closes	RCS leak R-11 spike may be caused by closure of OCBs 52/8 and 52/9 Cycling LCV-115A in any of the 3 different switch positions may cause this alarm
R-14C	RCV-014 closes.	WGDT leak or Safety lift
R-15	None	Primary to Secondary leak
R-18	RCV-018 closes.	WDS Effluent leakage
ALL	V1-31 closes.	Primary to Secondary leak
R-19A	1) FCV-1930A **AND** B close. 2) FCV-1933A **AND** B close. 3) FCV-4204A closes.	
R-19B	1) FCV-1931A **AND** B close. 2) FCV-1934A **AND** B close. 3) FCV-4204B closes	
R-19C	1) FCV-1932A **AND** B close. 2) FCV-1935A **AND** B close. 3) FCV-4204C closes.	
R-21	HVE-15 stops.	Fuel Handling accident. Low level in SFP.

OBSERVATIONS

1. Reading on affected channel(s)

ACTIONS CK (✓)

1. **IF** the cause of the alarm is known to be the movement of radioactive material **OR** is an expected alarm due to actions under operator control, **THEN** no further actions for this APP are required. _____

2. **OBSERVE** affected radiation monitor for radiation levels **AND** evidence of short term spiking. _____

3. **IF** a valid alarm is on R-11 **OR** R-12 **AND** the associated Automatic Actions did **NOT** occur, **THEN DEPRESS AND LATCH** the HV OFF pushbutton for R-11 **OR** R-12 **AND CHECK** that the Automatic Actions occurred for CV Ventilation Isolation. _____

4. **IF** short term spiking is evidenced, **THEN** allow the indicated level to lower prior to performing step 5. _____

correct

CONTINUOUS USE

INIT

8.4.2 Containment Vacuum Relief When Containment Integrity is Required

1. Initial Conditions

a. This revision has been verified to be the latest revision available. ________ Date ____

> **NOTE:** The CV NAR RANGE HI/LO PRESS annunciator (LO setpoint) alarms at -0.4 psig. (APP-002-B7)

b. Electrical Distribution is in service IAW OP-603. ____

c. Instrument and Station Air System is in service IAW OP-905. ____

d. **VERIFY** Containment Ventilation Isolation is **RESET**. ____

e. **VERIFY** the Containment Purge Supply and Exhaust valves **CLOSED**: (ITS LCO 3.6.3)

V12-6 ____

V12-7 ____

V12-8 ____

V12-9 ____

Correct

8.4.2 (Continued) INIT VERI

2. Instructions

a. **PLACE** CV VAC RELIEF V12-12 & V12-13 Control Switch to the open position. ____

b. **CHECK**, by position indicating lights, the following are **OPEN**.

- Showing here that the CV intake Damper auto shuts when V12-12/13 Go shut.

V12-12	____	
V12-13	____	
CV Intake Damper	____	

c. **WHEN** Containment pressure reaches between -0.025 psig and 0.0 psig on PI-950B, **THEN PERFORM** the following:

1) **PLACE** the CV VAC RELIEF V12-12 & V12-13 Control Switch to the close position. ____

2) **VERIFY**, by position indicating lights, the following are **CLOSED**.

V12-12	____	____
V12-13	____	____
CV Intake Damper	____	

	Initials	Name (Print)	Date
Performed By:	________	________________	________
	________	________________	________
	________	________________	________

Approved By: ______________________________ ____________
Shift Manager Date

50. 073 K4.01 001

Given the following plant conditions:

-The reactor is at 100% RTP

-The following radiation monitor alarms are received:

-R-15, CONDENSER AIR EJECTOR GAS MONITOR

-R-19A, S/G "A" RADIATION MONITOR

-R-24A, S/G "A" LEAKAGE MONITOR

-All other radiation monitors are normal

Which ONE (1) of the following describes the valves that CLOSE in response to these indications?

A. FCV-1930A, S/G A Blowdown Isolation Valve
V1-31, Blowdown Isolation Valve to Catch Basin

B. V1-31, Blowdown Isolation Valve to Catch Basin
RCV-10549, Condensate Polisher Discharge to Catch Basin

C✓ FCV-1930A, S/G A Blowdown Isolation Valve
FCV-1933B, S/G A Blowdown Sample Isolation Valve

D. FCV-1933B, S/G A Blowdown Sample Isolation Valve
RCV-10549, Condensate Polisher Discharge to Catch Basin

The correct answer is C.

A. Incorrect. V1-31 will only close if all three R-19s are in alarm. Plausible because part of the logic for closure is satisfied with the given conditions.

B. Incorrect. V1-31 will only close if all three R-19s are in alarm. Plausible because part of the logic for closure is satisfied with the given conditions. RCV-10549 does not close due to the conditions given in the stem. Plausible because RCV-10549 does receive a close signal on high radiation from another radiation monitor due to activity in the secondary. This signal comes from R-37, Condensate Polisher Waste Effluent Monitor.

C. Correct. FCV-1930A and FCV-1933B close as a result of the R-19A high radiation.

D. Incorrect. RCV-10549 does not close due to the conditions given in the stem. Plausible because RCV-10549 does receive a close signal on high radiation from another radiation monitor due to activity in the secondary. This signal comes from R-37, Condensate Polisher Waste Effluent Monitor.

Question: 50
Tier/Group: 2/1
K/A Importance Rating: RO 4.0 SRO 4.3

K/A: 073 Process Radiation Monitoring (PRM) System
K4: Knowledge of the PRM system design feature(s) and/or interlock(s) which provide for the following:
K4.01: Release termination when radiation exceeds setpoint

Reference(s): ST-019
Proposed References to be provided to applicants during examination: None
Learning Objective: LP-019 RMS Obj. 8
Question Source: Modified RNP Bank (RM-009 002)
Question History: 01-01 NRC Exam
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.11
Comments: This question meets the K/A because the candidate must possess knowledge of the PRM interlocks that close valves to terminate potential release points on high secondary activity.

MONITOR	MEDIUM MONITORED	FUNCTION
R-19A	SG "A" Blowdown	Closes; blowdown isolation valves FCV-1930A & FCV-1930B, sample isolation valves FCV-1933A & FCV-1933B, rate flow control valve FCV-4204A.
R-19B	SG"B" Blowdown	Closes; blowdown isolation valves FCV-1931A & FCV-1931B, sample isolation valves FCV-1934A & FCV-1934B, rate flow control valve FCV-4204B.
R-19C	SG "C" Blowdown	Closes; blowdown isolation valves FCV-1932A & FCV-1932B, sample isolation valves FCV-1935A & FCV-1935B, rate flow control valve FCV-4204C.

Correct

MONITOR	MEDIUM MONITORED	FUNCTION
R-21	Fuel Handling Bldg Upper Level Gas	Stops HVE-15 which will stop HVS-4.
R-37	Condensate Polisher	Isolates waste line to settling ponds (RCV-10549).

Incorrect

MONITOR	MEDIUM MONITORED	FUNCTION
R-11	CV Air or Stack Particulate	Closes C.V. purge supply and exhaust; pressure and vacuum relief valves.
R-12	CV Air or Stack Gas	Same function as R-11
R-14C	Stack Gas (Low Range)	Closes waste gas decay tank release valve (RCV-014); swaps R-14 Skid over to high range (two different setpoints).
R-14D	Stack Gas (Mid Range)	Swaps R-14 Skid over to low range.
R-18	Liquid Waste Disposal	Closes waste disposal system liquid release valve (RCV-018)
NOTE:	The blowdown tank release isolation valve (V1-31) will close if all three SG monitors (R-19A, R-19B and R-19C) are in alarm.	

51. 076 A1.02 001

Given the following plant conditions:

-The reactor is at 100% RTP

-Service Water pumps A & C are running

-Service Water Header pressure is 42 psig and stable as indicated on PI-1616 and PI-1684

-Both CCW Heat Exchangers are in service

-Main Generator Exciter Cooler air discharge temperature is 145°F and rising slowly

TCV-1673, TURBINE LUBE OIL TEMP CONTROL VALVE
TCV-1650, HYDROGEN COOLER TEMP CONTRL VALVE

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

IAW OP-903, Service Water System, the maximum allowable Main Generator Exciter Cooler air discharge temperature is ___(1)___°F. To restore Main Generator Exciter Cooler air discharge temperature the crew will first ___(2)____.

A. (1) 140
(2) start "D" Service Water pump

B.✓ (1) 188
(2) start "D" Service Water pump

C. (1) 140
(2) raise the setpoint for TCV-1673 and TCV-1650

D. (1) 188
(2) raise the setpoint for TCV-1673 and TCV-1650

The correct answer is B.

A. Incorrect. OP-903, P & L 5.6 states that the max allowable temperature is 188°F. 140°F is plausible because it is a trigger for performing section 8.4.12 to control temperature, but it is not the max allowable temperature.

B. Correct. OP-903, P & L 5.6 states that the max allowable temperature is 188°F. Section 8.4.12 has the start of additional SW pumps as the first physical action to take to raise SW flow to the Exciter and Iso-phase Bus Duct HX to control temperature.

C. Incorrect. OP-903, P & L 5.6 states that the max allowable temperature is 188°F. 140°F is plausible because it is a trigger for performing section 8.4.12 to control temperature, but it is not the max allowable temperature. Section 8.4.12 has the start of additional SW pumps as the first physical action to take to raise SW flow to the Exciter and Iso-phase Bus Duct HX to control temperature. Raising the setpoint of TCV-1673 and TCV-1650 is plausible because it is a method used in OP-908, section 8.4.12, but the step states that it is performed only after SW Pump and CCW Heat Exchanger flow adjustments and then only if it is determined that margin exists to raise the setpoint.

D. Incorrect. Section 8.4.12 has the start of additional SW pumps as the first physical action to take to raise SW flow to the Exciter and Iso-phase Bus Duct HX to control temperature. Raising the setpoint of TCV-1673 and TCV-1650 is plausible because it is a method used in OP-908, section 8.4.12, but the step states that it is performed only after SW Pump and CCW Heat Exchanger flow adjustments and then only if it is determined that margin exists to raise the setpoint.

Question: 51
Tier/Group: 2/1
K/A Importance Rating: RO 2.6 SRO 2.6

K/A: 076 Service water system (SWS)
A1: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including:
A1.02: Reactor and turbine building closed cooling water temperatures

Reference(s): OP-903
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.10
Comments:

5.6 The following applies to the Exciter air temperature:

Temperature Limits

- Maximum allowable Exciter inlet air temperature is 140°F (60°C) as indicated on ERFIS points TGT3310A or TGT3311A. Correct

- Maximum allowable Exciter air discharge temperature is 188°F (87•C) as indicated by ERFIS Point TGT3313A **OR** on the RTGB meter TI-1362.

- Exciter air temperatures exceeding any of these values with the unit in service should be investigated by Engineering.

- Operation below the temperature limit is acceptable. To reduce the potential for condensation formation attempts should be made to maintain exciter air inlet temperature at or above 40°C (ERFIS points TGT-3311A). In addition attempts should be made to maintain exciter air discharge temperatures below 60°C (TI-1362 on RTGB) to reduce the temperature effects on the electrical insulation.

5.7 When the Exciter is in service, **ALL** the individual Exciter coolers must remain in operation to prevent an excessive temperature rise, unless an evaluation has been performed by Engineering which supports removing a cooler from service. (NCR23925)

5.8 A GEN H_2 HI TEMP (APP-009-B3) alarm will be received when the temperature reaches 52•C. This may result from Generator overloading due to winding faults or winding degradation resulting in the high temperature.

5.9 A ISO-PHASE BUS DUCT AIR HI TEMP (APP-009-F4) alarm will be received when the air temperature reaches 70•C.

5.10 Engineering Surveillance Test, EST-069, verifies that each HVH Unit Combined Containment Air Recirc. Unit and associated motor cooler discharge line is being sampled by Radiation Monitor,R-16. These discharge line flows shall be balanced IAW EST-069 to establish that R-16 is fully performing its intended function.

INIT

8.4.12 Increased Service Water Cooling to Exciter Cooler and Iso-Phase Bus Duct Heat Exchanger

1. Initial Conditions

   a. This revision has been verified to be the latest revision available. _______ Date _____

   b. The Exciter Cooler inlet air temperature is approaching **OR** has exceeded 135°F (57°C) as indicated by ERFIS point TGT3310A **OR** TGT3311A,

   OR

   The Exciter Cooler discharge air temperature is approaching **OR** has exceeded 140°F (60°C) as indicated by ERFIS point TGT3313A **OR** on the RTGB,

   Incorrect

   OR

   Engineering has determined additional Service Water cooling flow is necessary for the Isolated-Phase Bus Duct Heat Exchanger _____

2. Instructions

INIT VERI

NOTE:	The Service Water Header pressures, as indicated on PI-1616 and PI-1684, should be maintained within the following normal ranges: - 40 to 50 psig with less than four Service Water pumps operating - 40 to 55 psig with four Service Water Pumps operating

CAUTION

Service Water header pressures of up to 57 psig are acceptable, provided none of the Service Water Pumps are operated continuously in the unstable flow range of 60 to 68 psig pump discharge pressure.

a. **START** additional Service Water Pump(s), as required to raise flows. (N/A pumps **NOT** started)

Correct

SW Pump "A" ____

SW Pump "B" ____

SW Pump "C" ____

SW Pump "D" ____

b. **IF** only CCW Heat Exchanger "A" is in service, **THEN THROTTLE** SW-739, CCW HEAT EXCHANGER "A" RETURN, as necessary to raise flows to the secondary side coolers while maintaining the following conditions:

- SW-739 throttled open less than or equal to POS 2 ____ ____

8.4.12.2.b (Continued)

INIT VERI

- SW header pressure 40 to 50 psig with three or less Service Water Pumps operating **OR** SW header pressure 40 to 55 psig with four Service Water Pumps operating ____
- CCW Heat Exchanger outlet temperature less than or equal to 105• F ____

c. **IF** only CCW Heat Exchanger "B" is in service, **THEN THROTTLE** SW-740, CCW HEAT EXCHANGER "B" RETURN, as necessary to raise flows to the secondary side coolers while maintaining the following conditions:

- SW-740 throttled open less than or equal to POS 2 ____ ____
- SW header pressure 40 to 50 psig with three or less Service Water Pumps operating **OR** SW header pressure 40 to 55 psig with four Service Water Pumps operating ____
- CCW Heat Exchanger outlet temperature less than or equal to 105• F ____

8.4.12.2 (Continued)

	INIT	VERI
d. **IF** both CCW Heat Exchangers are in service, **THEN THROTTLE** SW-739 **AND** SW-740 as necessary to raise flows to the secondary side coolers while maintaining the following conditions:		
• SW-739 throttled open less than or equal to POS 1	____	____
• SW-740 throttled open less than or equal to POS 1	____	____
• SW header pressure 40 to 50 psig with three or less Service Water Pumps operating **OR** SW header pressure 40 to 55 psig with four Service Water Pumps operating	____	
• CCW Heat Exchangers outlet temperature less than or equal to 105•F	____	

8.4.12.2 (Continued)

INIT

NOTE:	Parameters that would need to be addressed and monitored to determine if increasing the set point for TCV-1673 or TCV-1650 would be beneficial are: (EC 47139) TCV-1673: Service Water pressure, Lube Oil temperatures, Turbine Bearing temperatures (refer to OP-506, Attachment 10.2), Service Water temperatures at the Lube Oil Cooler, Turbine/Generator/ Exciter vibrations levels. TCV-1650: Service Water pressure, Hydrogen temperatures, Differential temperatures between high and low Hydrogen Gas Discharge temperature, Service Water temperatures at the Lube Oil Cooler. Small incremental temperature adjustment in both TCV-1673 and TCV-1650 is more desirable than a larger temperature adjustment in just one of the temperature control valves.

e. **IF** sufficient cooling water flow was **NOT** provided to the Exciter Cooler **OR** the Iso-Phase Bus Duct Heat Exchanger following Service Water Pump **AND** CCW Heat Exchanger adjustments, **THEN** **PERFORM** the following to provide additional Service Water cooling flow to these components:

incorrect

1) **DETERMINE** if sufficient margin exists in Turbine oil and bearing temperatures **AND** in the generator gas temperatures to allow increasing the set point for either of the following: ____

- TCV-1673, TURBINE LUBE OIL TEMP CONTROL VALVE
- TCV-1650, HYDROGEN COOLER TEMP CONTROL VALVE

Incorrect

8.4.12.2.e (Continued)

INIT

2) **IF** margin is available for increasing TCV-1673 **OR** TCV-1650 temperature set point, **THEN PERFORM** the following to raise the set point of applicable TCV(s):

.a) **RECORD** initial TCV temperature setting:

- TCV-1673 ______ °F
- TCV-1650 ______ °F ____

.b) **INCREMENTALLY RAISE** the temperature set point for TCV-1673 **AND** TCV-1650 as necessary to raise cooling flow to the Exciter Cooler **OR** the Iso-Phase Bus Duct Heat Exchanger while monitoring system temperatures. ____

NOTE: A power reduction may be necessary to reduce heat loads on the coolers.

f. **IF** sufficient cooling water flow is **NOT** available for the Exciter Cooler **OR** the Iso-Phase Bus Duct Heat Exchanger following TCV temperature set point adjustment, **THEN CONSULT** with Management to determine additional actions to be taken. ____

	Initials	Name (Print)	Date
Performed By:	______	______________________	____
	______	______________________	____
	______	______________________	____
Approved By:		______________________ Shift Manager	____ Date

52. 076 K2.01 001

Given the following plant conditions:

-MCC-5 de-energizes due to the supply breaker tripping open

-During the transient, a turbine trip occurs and the South Service Water header pressure decreases to 28 psig

V6-16B, SW SOUTH HEADER SUPPLY TO TURBINE BUILDING
V6-16C, SW ISOLATION TO TURBINE BUILDING

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

TWO (2) minutes after the turbine trip, V6-16B is ___(1)___ and V6-16C is ___(2)___ .

A. (1) closed
(2) closed

B. (1) closed
(2) open

C. (1) open
(2) open

D.✓ (1) open
(2) closed

The correct answer is D.

A) Incorrect. V6-16B is closed is incorrect. Plausible if the student doesn't realize that the valve lost power and can not shut. V6-16C is closed. It has an ABT that auto swaps from MCC-10 to MCC-9 on a loss of power.

B) Incorrect. V6-16B is closed is incorrect. Plausible if the student doesn't realize that the valve lost power and can not shut. V6-16C is open is incorrect. Plausible if the student forgets about the ABT that auto swaps the valve's power supply from MCC-10 to MCC-9.

C) Incorrect. V6-16B is open is correct. Because the valve has lost power, it cannot shut despite needing to shut because of the service water pressure. V6-16C is open is incorrect. Plausible if the student forgets about the ABT that auto swaps the valve's power supply from MCC-10 to MCC-9.

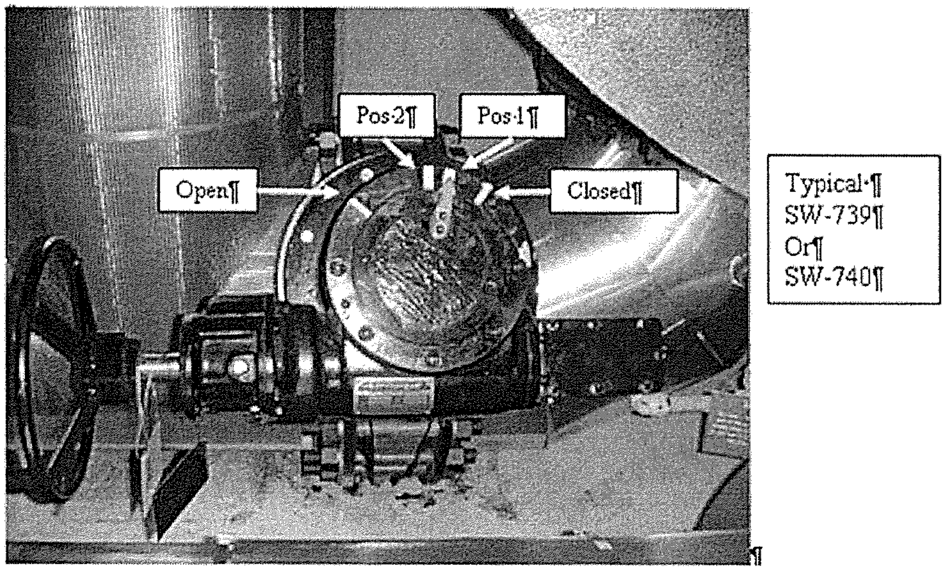
D) Correct. V6-16B is open is correct. Because the valve has lost power, it cannot shut despite needing to shut because of the service water pressure. V6-16C is closed. It has an ABT that auto swaps from MCC-10 to MCC-9 on a loss of power.

Question: 52
Tier/Group: 2/1
K/A Importance Rating: RO 2.7 SRO 2.7

K/A: 076 Service water System (SWS)
K2: Knowledge of bus power supplies to the following
K2.01: Service water

Reference(s): ST-004, ST-016
Proposed References to be provided to applicants during examination: None
Learning Objective: LP-04 Obj. 6
Question Source: RNP Bank
Question History: Not used on a previous NRC exam
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.7
Comments: This question meets the K/A because the candidate must have knowedge of the power supplies to Service Water valves to answer the question.

Figure 5 SW Throttle Valve for SW Pressure



The SW North and South Headers also supply water to the Turbine Building Header. The North Header is supplied through Valve V6-16A and the South Header is supplied through Valve V6 16B. In addition, Valve V6-16C is used to completely isolate the North and South Header from the Turbine Building Header. The normal power supply of Valve V6-16C is MCC 10. If MCC 10 loses its power V6-16C motor control will automatically switch to MCC 9 as the power supply. If a sustained low SW Header pressure occurs in coincidence with a Turbine Trip Signal all three valves will automatically close; with an alarm provided on APP-008. If SW to the Turbine Building is to be secured, the secondary side loads must be secured (4KV pumps, Main Turbine, Main Generator). Supplement M - Component Alignment for Loss of SW to Turbine Building can be used to secure the necessary equipment.

If directed by the EOP network, Supplement Q, Local Manual Isolation of Turbine Building Service Water, can be used to close SW-969, a chain operated valve located in the Turbine Building. This manual valve is used to isolate SW to the turbine Building if there is only one EDG operating with one SW pump and the MOV's will not work. (EC 74133)

The South and North SW Header can be further isolated by removing the spacer plate downstream of V6-12A or V6-12D and substituting a blank plate.

During Refueling Outage (RFO) 19, a portion of the North SW Header underground piping was replaced with above ground piping. This replacement was due to several minor leaks in this portion of the SW System. It is believed that a combination of failure of the

V6-16A, B & C AUTO CLOSURE FEATURE TO ISOLATE TURBINE BUILDING:

- Valve V6-16A will close if PSL-1616A reaches 31 psig decreasing for 60 seconds with a 20ET Turbine Trip signal present

- Valve V6-16B will close if PSL-1684A reaches 31 psig decreasing for 60 seconds with a 20AST Turbine Trip signal present

- Valve V6-16C will close if PSL-1616B OR PSL-1684B reaches 31 psig decreasing for 60 seconds with a 20ET OR 20AST Turbine Trip signal present

Key lock switches (located in the Cable Spread Room) are used during maintenance, testing, or when the unit is in Cold Shutdown to inhibit the auto-close interlock.

Component	Power Supply
V6-16B (South Header Turbine Bldg Isolation)	MCC-10
V6-16C (Turbine Bldg header Isolation)	MCC-10/9

Electrical Manhole Sump Pumps and Power Cable Trenches

The Service Water Pump Power cables are routed between the Intake Structure and the Reactor Auxiliary Building (RAB) through a concrete pre-fabricated trench (EC 52696). The cables enter the trenches though two Manholes between the RAB and the Radioactive Waste Building and the cables exit the trenches though two Manholes at the intake structure. Due to the high water table, the manholes are constantly being filled with water. Permanent sump pumps in Manholes M35 and M36 have been installed in order to improve the life span and reliability of the cables installed in these manholes (ESR 98-00319). The sump pumps have level switches installed in the manholes to provide automatic start of the pumps when water accumulates in the manhole sump. A sump pump control panel has been installed in the area of the manholes and has a manual pump switch installed to provide for manual operation of the pump. The control panel also has a pump run light for each pump that will be illuminated when the pumps are in operation. A level switch has been installed in the manholes to energize an alarm light and horn on the control panel if the water level in either of the manholes rises above 18 inches.

The discharge piping of the sump pumps has been installed with freeze protection cable and insulation to ensure that the piping does not freeze during cold weather. The control panel includes a freeze protection test switch for each of the discharge lines freeze protection circuits and an ammeter to provide indication of the circuits' current draw. This modification also installed three (3) spare cables from PP-61 to the sump pump control panel. A storage box has been installed near the sump pump control panel in order to store the Annunciator Procedure APP-051. This modification has also installed a sump pump and control box with freeze protection for Service Water Pit No. 3 Sump.

The original SW pumps power cables were directly buried and had cable splices where they had been damaged. The cables were degrading and reliability to last another 20 years was in question. The solution was to replace the cables. The EC to replace the SW power cables involves the use of new cable in three trenches (Safety Train "A", Safety Train "B" and non-safety). The Safety cables run from the intake through the reinforced

Motor Control Centers

The MCCs provide power for various pumps, fans, and other auxiliary equipment.

The MCCs and their power supply are as follows:

- MCC-1 supplied from 480Vac Bus No. 1
- MCC-2 supplied from 480Vac Bus No. 2A
- MCC-3 supplied from 480Vac Bus No. 2B
- MCC-4 supplied from 480Vac Bus No. 3
- MCC-4A supplied from 480Vac MCC-4
- MCC-5 supplied from 480Vac Bus E-1 or Bus DS (Alternate Supply)
- MCC-6 supplied from 480Vac Bus E-2
- MCC-7 supplied from 480Vac Bus No. 2A
- MCC-8 supplied from 480Vac MCC-4 (transformed to 208Vac)
- MCC-9 supplied from 480Vac MCC-6 (transformed to 208Vac)
- MCC-10 supplied from 480Vac MCC-5 (transformed to 208Vac)
- MCC-11 supplied from 480Vac Bus No. 4
- MCC-12 supplied from 480Vac Bus No. 4
- MCC-13 supplied from 480Vac Bus No. 4
- MCC-14 supplied from 480Vac Bus No. 4
- MCC-15 supplied from 480Vac Bus No. 5
- MCC-16 supplied from 480Vac Bus E1 (In parallel with MCC-5)
- MCC-17 supplied from 480Vac Bus No. 5
- MCC-18 supplied from 480Vac Bus E2
- MCC-19 supplied from PP No. 22
- MCC-20 supplied from 480Vac Bus No. 2B

53. 078 G2.4.50 001

Given the following plant conditions:

-APP-002-F7, INSTR AIR HDR LO PRESS alarms

-PI-1702, INST AIR HEADER PRESS, dropped from 100 psig to 0 psig in less than 10 seconds

-Pressurizer Level is 54% and **STABLE**

-Steam Generator Narrow Range Levels are 52% and **STABLE**

-The reactor is at 100% RTP and **STABLE**

AOP-017, LOSS OF INSTRUMENT AIR
EOP-E-0, REACTOR TRIP OR SAFETY INJECTION

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

APP-002-F7 alarm setpoint is ___(1)___ psig. The crew is required to ___(2)___ .

A. (1) 60
(2) initiate action to repair the transmitter and monitor Instrument Air pressure locally

B.✓ (1) 85
(2) initiate action to repair the transmitter and monitor Instrument Air pressure locally

C. (1) 60
(2) trip the reactor and implement AOP-017 and EOP-E-0 concurrently

D. (1) 85
(2) trip the reactor and implement AOP-017 and EOP-E-0 concurrently

The correct answer is B.

A. Incorrect. Per APP-002-F7 the setpoint for the INSTR AIR HDR LOW PRESS is 85 psig. 60 psig is plausible because it is a value tied to lowering instrument air in AOP-017. If it were an actual problem pressurizer and S/G levels, and reactor power would not be stable. Per APP-002-F7 the crew should initiate action to repair the instrument and monitor instrument air pressure locally for a failed instrument.

B. Correct. Per APP-002-F7 the setpoint for the INSTR AIR HDR LOW PRESS is 85 psig. The indications given indicate a failed instrument vice an air leak or compressor problem. If it were an actual problem pressurizer and S/G levels, and reactor power would not be stable. Per APP-002-F7 the crew should initiate action to repair the instrument and monitor instrument air pressure locally for a failed instrument.

C. Incorrect. Per APP-002-F7 the setpoint for the INSTR AIR HDR LOW PRESS is 85 psig. 60 psig is plausible because it is a value tied to lowering instrument air in AOP-017. The action would be correct if the malfunction were an actual lowering of IA pressure.

D. Incorrect. The action would be correct if the malfunction were an actual lowering of IA pressure. The action is plausible because a trip would be required if the malfunction was something other than an instrument malfunction.

Question: 53
Tier/Group: 2/1
K/A Importance Rating: RO 4.2 SRO 4.0

K/A: 078 Instrument Air System
G2.4: Emergency Procedures/Plan
G2.4.50: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Reference(s): APP-002, AOP-017
Proposed References to be provided to applicants during examination: None
Learning Objective: AOP-017 Obj. 9
Question Source: New
Question History:
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.4, 10
Comments:

This question meets the K/A because the candidate must know when the alarm comes in and analyze additional confirmatory parameters to determine the correct course of action.

Correct

ALARM

INSTR AIR HDR LO PRESS

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Compressor Failure
2. Relief Valve Open **OR** Leaking
3. Ruptured Line
4. Transmitter failure.

OBSERVATIONS

1. Instrument Air Header Pressure (PI-1702)

ACTIONS	CK (✓)
1. **IF** Instrument Air pressure is dropping, **THEN REFER TO** AOP-017.	______
2. **IF** PT-1702 has failed, **THEN INITIATE** action to repair the transmitter **AND MONITOR** Instrument Air pressure locally.	______

DEVICE/SETPOINTS

1. PSL-1702 / 85 psig

POSSIBLE PLANT EFFECTS

1. Plant Shutdown
2. Loss of operability of various air operated valves

REFERENCES

1. AOP-017, Loss of Instrument Air
2. CWD B-190628, Sheet 590, Cable H

Incorrect

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
1.	Check Plant Status - MODE 1 OR MODE 2	Go To Step 4.
* 2.	Check IA Header Pressure - LESS THAN 60 PSIG	IF IA pressure lowers to less than 60 psig, THEN Go To Step 3. Go To Step 4.
3.	Perform The Following: a. Trip the Reactor b. Go To PATH-1 OR EOP-E-0, Reactor Trip or Safety Injection, while continuing with this procedure	
4.	Make PA Announcement For Procedure Entry	
5.	Verify Instrument Air Compressor D - RUNNING	
6.	Verify The Primary Air Compressor - RUNNING	
* 7.	Check IA Header Pressure - LESS THAN 80 PSIG	IF IA pressure lowers to less than 80 psig, THEN observe NOTE prior to Steps 8 and 9 and perform Steps 8 and 9. Observe the NOTE Prior To Step 10 and Go To Step 10.

54. 078 K1.04 001

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

Instrument Air Compressor A and B cylinder water jackets are cooled by __(1)___ Water. Flow is ___(2)___ to the cylinder water jackets when the air compressors are shutdown.

A. (1) Service
   (2) maintained

B. (1) Component Cooling
   (2) maintained

C✓ (1) Service
   (2) isolated

D. (1) Component Cooling
   (2) isolated

The correct answer is C.

A. Incorrect. Plausible, since many components continue to have flow aligned when shutdown.

B. Incorrect. Plausible because Component Cooling Water supplies cooling to other components located in the Auxiliary Building. Plausible, since many components continue to have flow aligned when shutdown.

C. Correct. Service Water supplies cooling to IA Compressors A and B cylinder jackets and the associated after coolers. Water flow is isolated by a solenoid valve when the associated compressor is shutdown to prevent condensation in the cylinder from moist air.

D. Plausible because Component Cooling Water supplies cooling to other components located in the Auxiliary Building.

Question: 54
Tier/Group: 2/1
K/A Importance Rating: RO 2.6 SRO 2.9

K/A: 078 Instrument Air System (IAS)
K1: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:
K1.04: Cooling water to compressor

Reference(s): ST-017 IA-SA
Proposed References to be provided to applicants during examination:
Learning Objective: LP-017 IA/SA, Objective 5c
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.4
Comments: This question meets the K/A because the candidate must know the physical connections for cooling water to compressors and the cause-effect relationship of securing the instrument air compressor in relation to cooling water flow.

COMPONENT DESCRIPTION

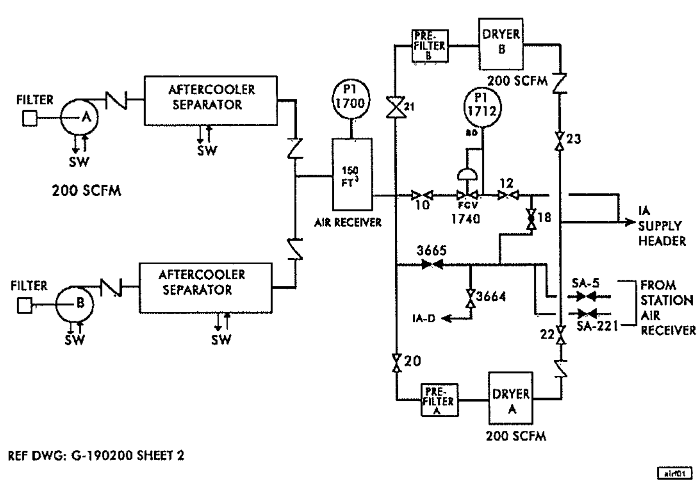
IA "A" & "B" Compressor Package

The "A" & "B" IACs are 200 scfm, Ingersoll Rand, non-lubricated, single stage reciprocating, 100 psig compressors. The water jackets for cylinder cooling are supplied by SW and limits air discharge temperature to 415°F. The aftercoolers are shell and tube heat exchangers with SW on the shell side and will reduce the air outlet temperature.

Air temperature at the receiver discharge is maintained at about 110°F by controlling the SW flow to the compressor water jackets and the aftercoolers. SW is automatically secured when the compressor is stopped, via a solenoid valve. This prevents condensation buildup in the cylinder.

The compressors and receiver are equipped with internal baffles, traps, and relief valves. The compressors are powered from MCC-5 for IAC "A" and MCC-6 for IAC "B". The compressors discharge into a 150 cubic foot vertical air receiver, located in the Auxiliary Building hallway. The air dryers have a capacity of 200 scfm each, and are of the refrigeration type, which will maintain air dew point of minus 10°F.

Figure 2 "A" and "B" Instrument Air Compressors (Drawing)



55. 103 K4.04 001

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

To ensure the LCO is met for the Personnel Airlock as required by Technical Specification 3.6.2 in MODES ___(1)___, the personnel hatch doors are ___(2)___ to ensure only one door is open at any time.

A. (1) 1-4 only
   (2) administratively controlled

B. (1) 1-4, and 6
   (2) administratively controlled

C.✓ (1) 1-4 only
   (2) physically interlocked

D. (1) 1-4, and 6
   (2) physically interlocked

The correct answer is C.

A. Incorrect. Administrative control does not meet the LCO. Plausible because the action does allow for administrative control of the doors by using a dedicated operator during access and locking one door closed when not accessing the containment.

B. Incorrect. Per TS 3.6.2, the applicability of the TS is MODES 1-4. In MODE 6 the airlock is and interlocks are not required to be operable however one door must remain closed. Plausible because MODE 6 does require some control of the airlock doors. Administrative control does not meet the LCO for 3.6.2. Plausible because the action does allow for administrative control of the doors by using a dedicated operator during access and locking one door closed when not accessing the containment.

C. Correct. Per TS 3.6.2, the applicability of the TS is MODES 1-4. To meet the LCO the door interlocks must be OPERABLE. If not Action B of TS 3.6.2 applies.

D. Incorrect. Per TS 3.6.2, the applicability of the TS is MODES 1-4. In MODE 6 the airlock is and interlocks are not required to be operable however one door must remain closed. Plausible because MODE 6 does require some control of the airlock doors.

Question: 55
Tier/Group: 2/1
K/A Importance Rating: RO 2.5 SRO 3.2

K/A: 103 Containment
K4: Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following:
K4.04: Personnel access hatch and emergency access hatch

Reference(s): ITS 3.6.2
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Fundamental Knowledge
10 CFR Part 55 Content: 41.7, 9
Comments: This question meets the K/A because the candidate must possess knowledge of the interlocks associated with the personnel airlock.

3.6 CONTAINMENT SYSTEMS

3.6.2 Containment Air Lock

Correct

LCO 3.6.2 The containment air lock shall be OPERABLE.

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

ACTIONS

------------------------------------NOTES------------------------------------

1. Entry and exit is permissible to perform repairs on the affected air lock components.

2. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when air lock leakage results in exceeding the overall containment leakage rate.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment air lock door inoperable.	------------NOTES------------ 1. Required Actions A.1, A.2, and A.3 are not applicable if both doors are inoperable and Condition C is entered. 2. Entry and exit is permissible for 7 days under administrative controls. -----------------------------	
	A.1 Verify the OPERABLE door is closed.	1 hour
	<u>AND</u>	
	A.2 Lock the OPERABLE door closed.	24 hours
	<u>AND</u>	(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

leakage. The containment has an allowable leakage rate of 0.1% of containment air weight per day at 42 psig (Ref. 2).

The containment air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

The containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in the air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

Correct

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

Incorrect

(continued)

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by four bolts;

b. One door in the air lock closed; and

incorrect

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During movement of recently irradiated fuel assemblies within containment. Mode 6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of recently irradiated fuel assemblies within containment.	Immediately

56. 002 A2.04 001

Given the following plant conditions:

-The reactor was at 100% RTP.

-Subsequently, the reactor was tripped due to a Steam Line leak on the 72 inch Main Steam Header

-All MSIVs are stuck full open.

-All AFW Pumps failed to start

-FRP-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, has been entered

-S/G Wide Range levels are all at 3%

-CETC Temperature indicates 505°F with a subcooling of 115°F

-Safety Injection has been manually initiated

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

Safety injection ___(1)___ injecting into the core. In order to initiate Bleed and Feed, the crew will **FIRST** try to open ___(2)___.

A. (1) is
(2) all head vents

B. (1) is **NOT**
(2) all head vents

C. (1) is
(2) both PZR PORV's

D✓ (1) is **NOT**
(2) both PZR PORV's

The correct answer is D

A) Incorrect. Based on the information given, RCS pressure is approximately 1800 psig. Therefore, SI is not injecting since 1800 psig is above the shutoff head of the pumps. Plausible if the students do not take the subcooling value into account when determining the RCS pressure. The head vents are incorrect. Plausible because if the operators can't open the PORV's, they will next try the head vents.

B) Incorrect. The first part of the distractor is correct. The head vents are incorrect. Plausible because if the operators can't open the PORV's, they will next try the head vents.

C) Incorrect. Based on the information given, RCS pressure is approximately 1800 psig. Therefore, SI is not injecting since 1800 psig is above the shutoff head of the pumps. Plausible if the students do not take the subcooling value into account when determining the RCS pressure. The second part of the distractor is correct.

D) Correct. SI is not injecting into the RCS since pressure is above the shutoff head of the SI pumps. FRP-H.1 will have you first try to open both PZR PORV's to initiate the feed and bleed.

Question: 56
Tier/Group: 2/2
K/A Importance Rating: RO 4.3 SRO 4.6

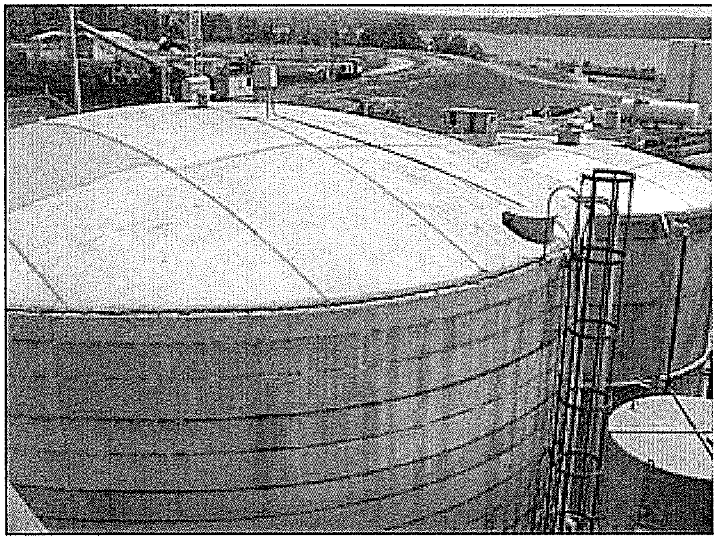
K/A: 002 Reactor Coolant System (RCS)
A2: Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.04: Loss of heat sinks

Reference(s): Simulator/plant design, FRP-H.1
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: New
Question History:
Question Cognitive Level: Analysis
10 CFR Part 55 Content: 41.5, 10
Comments:

This question meets the K/A because it requires the student to predict what will happen to RCS pressure, based off that, they will determine whether or not there is SI flow going into the core. From that, they will determine which method is used first to initiate Feed and Bleed from FRP-H.1 which is by opening the PZR PORV's.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
35.	**Verify Adequate RCS Bleed Path As Follows:** • PZR PORVs - BOTH OPEN • PZR PORV Block Valves - BOTH OPEN	Go To Step 37. Correct
36.	**Go To Step 41**	
37.	**Place the Key Switches for the following Vent Valves to the OPEN Position:** • RC-568, HEAD VENT • RC-570, PZR VENT • RC-572, CV ATMOS • RC-567, HEAD VENT • RC-569, PZR VENT • RC-571, PRT ISO	Incorrect
38.	**Depressurize At Least One Intact S/G To Atmospheric Pressure Using Steam Line PORVs**	

Figure 9 RWST



SI Pumps

Manufacturer	Worthington Corp.
Number	3
Model/Type	WT/Horizontal centrifugal
Design pressure	1750 psig
Design temperature	300°F
Operating Temperature	50 - 280°F
Shutoff head	1500 psi
Design flow rate , each	375 gpm
Design head	2500 ft.
Maximum flow, each	650 gpm

57. 011 K6.03 001

Given the following plant conditions:

-The crew is lowering power from 100% to 90% RTP IAW OP-105, MANEUVERING THE PLANT WHEN GREATER THAN 25% POWER

-All PZR heaters are energized

-B Charging pump is in manual

-C Charging pump is in AUTO

-LT-460, CH II PRZR LEVEL, fails low

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

The PZR Backup heaters are ___(1)___ . Because of this failure, C charging pump speed will ___(2)___.

A.✓ (1) off
(2) lower

B. (1) on
(2) lower

C. (1) off
(2) rise

D. (1) on
(2) rise

The correct answer is A

A) Correct. LT-460 trips all heaters regardless of them being in AUTO or MANUAL control. LT-460 trips the heaters at 14.4%. C charging pump is in AUTO and is being controlled by LC-459G and LT-459. PZR level is rising because letdown is isolated, this would cause charging pump speed to lower.

B) Incorrect. PZR Backup heaters will NOT be on. Plausible because if LT-459 failed low, they heaters would be still on since this transmitter only trips the backup heaters if they are in auto. C charging pump is in AUTO and is being controlled by LC-459G and LT-459. PZR level is rising because letdown is isolated, this would cause charging pump speed to lower.

C) Incorrect. LT-460 trips all heaters regardless of them being in AUTO or MANUAL control. LT-460 trips the heaters at 14.4%. C Charging pump speed will not rise. Plausible because if LT-459 failed low, C charging pump speed would rise since it is controlled by LC-459G and LT-459.

D) Incorrect. PZR Backup heaters will NOT be on. Plausible because if LT-459 failed low, they heaters would be still on since this transmitter only trips the backup heaters if they are in auto. C Charging pump speed will not rise. Plausible because if LT-459 failed low, C charging pump speed would rise since it is controlled by LC-459G and LT-459.

Question: 57
Tier/Group: 2/2
K/A Importance Rating: RO 2.9 SRO 3.3
K/A: 011 Pressurizer Level Control System (PZR LCS)
K6: Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS:
K6.03: Relationship between PZR level and PZR heater control circuit
References: Sim/Plant design,
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3 of PZR/PRT Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR part 55: (CFR 41.7 / 45.7)

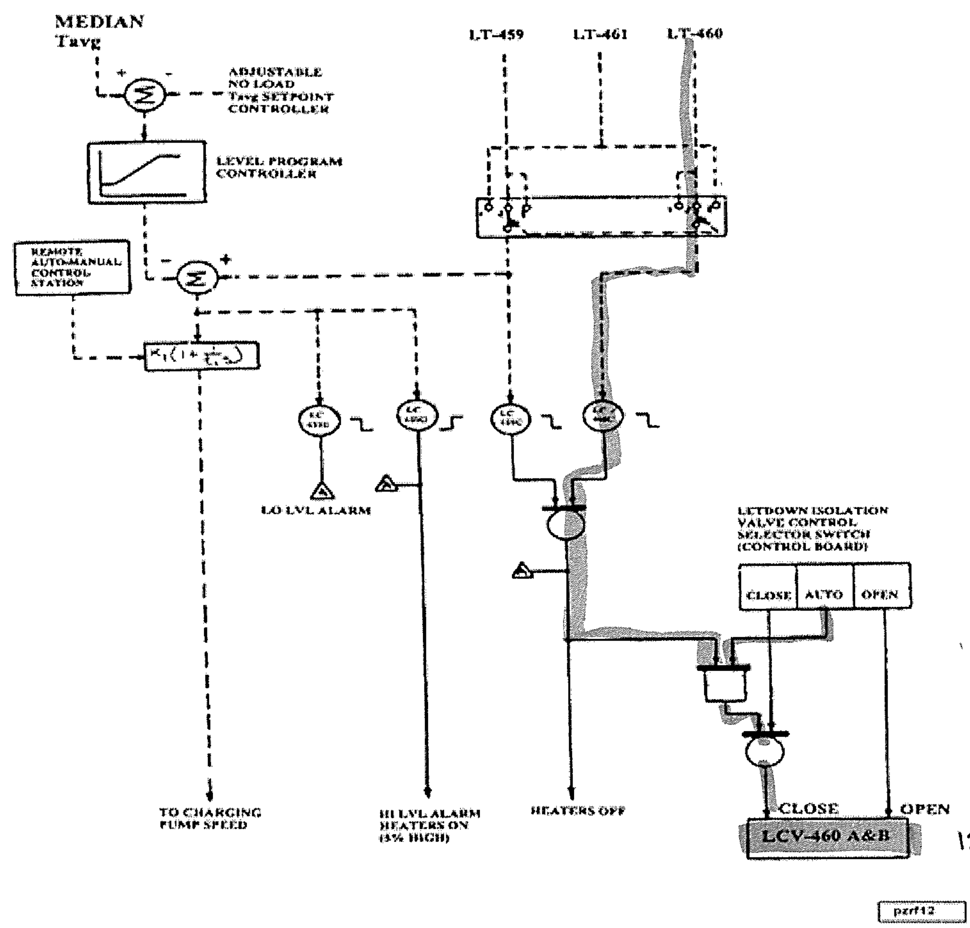
Meets the K/A because the question asks the effect the loss of LT-460(PZR Level) will have on the PZR heaters.

Correct

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would de-energize and any backup heater in manual would remain energized.

Figure 23 Level Control Logic



isolates letdown

incorrect

CHAPTER 4

INSTRUMENTATION

PZR Instrumentation

TEMPERATURE INSTRUMENTATION

The following temperature elements provide indication and alarm on the RTGB:

1. PZR Liquid Space (TE-453)
2. PZR Steam Space (TE-454)
3. PZR Spray Line (TE-451 and 452)
4. PZR Surge Line (TE-450)
5. Discharge of PORV's (TE-463) and each Safety Valve (TE-465, 467, and 469)

LEVEL

Three PZR level transmitters, calibrated at normal operating temperatures, are used to provide signals for reactor protection (High Level Trip).

1. LT-459
2. LT-460
3. LT-461

One PZR level signal, LT-462, is provided for indication when the system is in cold condition and therefore is calibrated at cold conditions.

Channels 459, 460, and 461 are used in protection and are available for control functions by a switch on the RTGB. Normally, Channels 459 and 460 are used for control, and either can be replaced by Channel 461. Channel 459 normally provides signal for charging pump speed control, set point deviation alarm, and will turn on backup heaters (if they are in automatic) on a high level error signal to heat incoming water. Either control channel can provide a letdown isolation signal (shut 460A and B in the CVCS system) and turn off all PZR heaters.

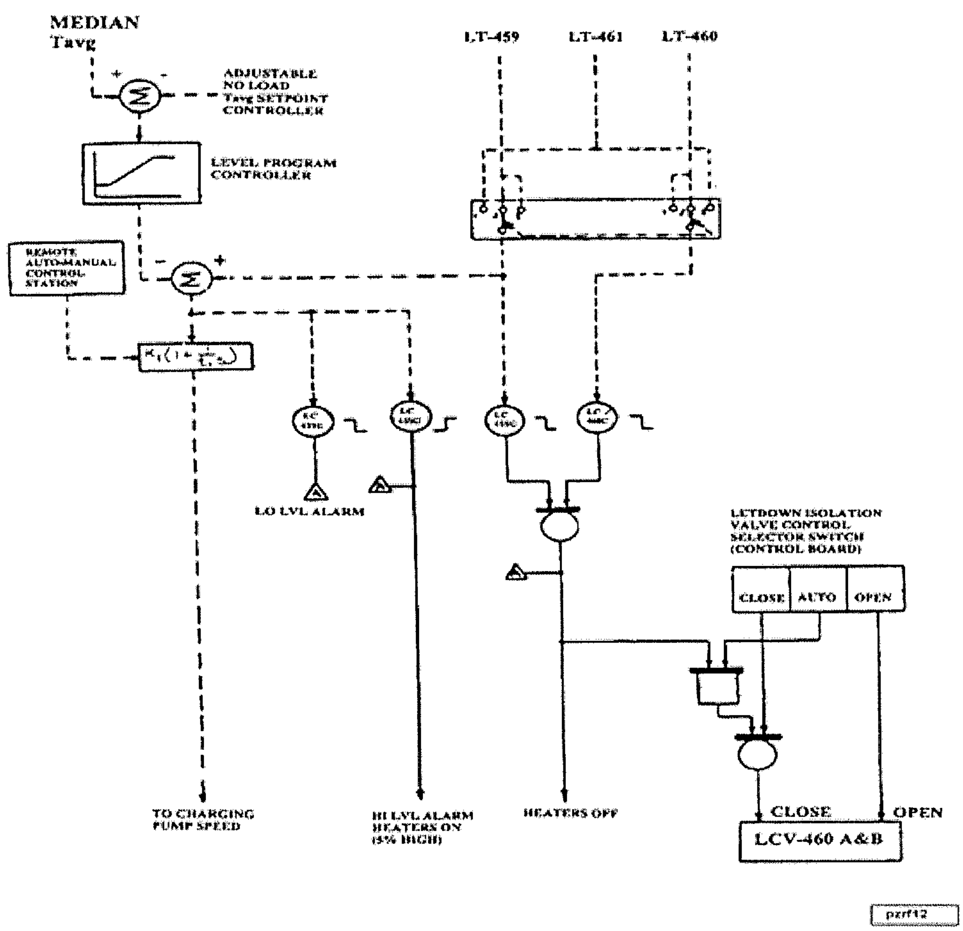
There is an additional PZR level signal LT-607D used for indication on the Dedicated Shutdown panel.

incorrect

On PZR low level of 14.4%, proportional and backup heaters are deenergized and letdown is isolated by shutting LCV-460A & B if respective control switches are in auto. LC-459 and the LC-460, the low level bistables, are normally supplied by LT-459 and LT-460 respectively but either can be replaced by LT-461 with a selector switch on the RTGB.

LC-459 will only turn off the backup heaters that are selected to Automatic where LC-460 will turn off the backup heaters in Automatic or Manual. The only time this would have any bearing would be in the event of an instrument failure. If the channel feeding LC-459, usually LT-459, were to fail low the proportional heaters and any backup heaters in Automatic would de-energize and any backup heater in manual would remain energized.

Figure 23 Level Control Logic



58. 014 G2.1.30 002

Given the following plant conditions:

-The reactor is at 100% RTP

-The crew has entered AOP-001, MALFUNCTION OF REACTOR CONTROL SYSTEM, for a dropped rod in Control Bank D

-They are at the step to place the Lift Coil Disconnect Switch for the dropped rod to the OFF position

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

The operator performing this step will go to the ___(1)___ . This is done to allow operators to ___(2)___ .

A. (1) Rod Control Room
(2) withdraw the dropped rod

B. (1) back of the RTGB
(2) withdraw the dropped rod

C. (1) Rod Control Room
(2) control flux during the downpower

D✓ (1) back of the RTGB
(2) control flux during the downpower

The correct answer is D.

A) Incorrect. The operator will go to the back of the RTGB. Plausible since most of the Rod Control components are in the Rod Control Room. Placing the Lift Coil Disconnect switch to the OFF position for the dropped rod allows the operators to control flux while reducing power. They reduce power so that they can restore the dropped rod. Plausible if the student doesn't realize they haven't reduced power yet, which is necessary to restore the dropped rod. To withdraw the dropped rod, the crew will place all the Lift Coil Disconnect switches for that bank in OFF, then withdraw the rod. This allows only the dropped rod to be withdrawn.

B) Incorrect. The operator performing this step will go to the back of the RTGB to perform this task. Placing the Lift Coil Disconnect switch to the OFF position for the dropped rod allows the operators to control flux while reducing power. They reduce power so that they can restore the dropped rod. Plausible if the student doesn't realize they haven't reduced power yet, which is necessary to restore the dropped rod. To withdraw the dropped rod, the crew will place all the Lift Coil Disconnect switches for that bank in OFF, then withdraw the rod. This allows only the dropped rod to be withdrawn.

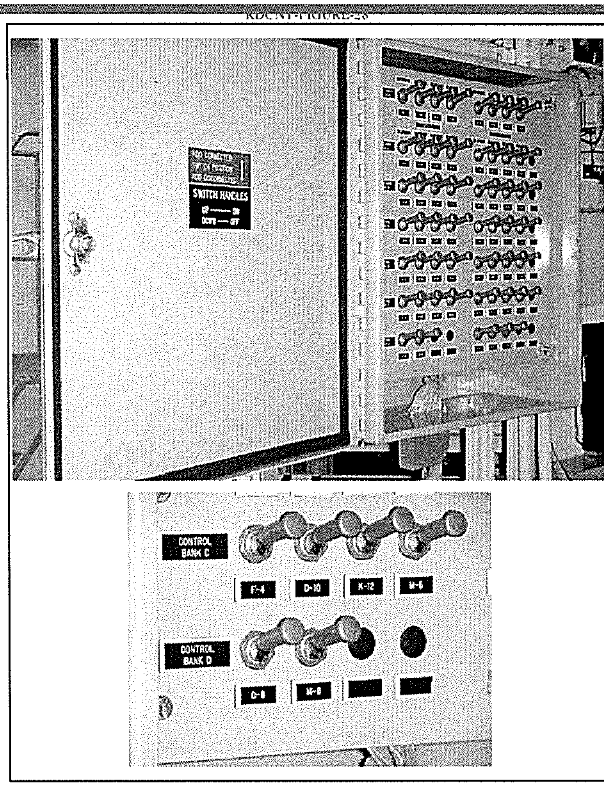
C) Incorrect. The operator will go to the back of the RTGB. Plausible since most of the Rod Control components are in the Rod Control Room. The Lift Coil Disconnect switch for the Dropped Rod is placed in OFF so that operators can control flux during the downpower, which is necessary to do first prior to restoring the dropped rod.

D) Correct. The operator performing this step will go to the back of the RTGB to perform this task. The Lift Coil Disconnect switch for the Dropped Rod is placed in OFF so that operators can control flux during the downpower, which is necessary to do first prior to restoring the dropped rod.

Question: 58
Tier/Group: 2/2
K/A Importance Rating: RO 4.4 SRO 4.0
K/A: 014 Rod Position Indication System
G2.1.30: Ability to locate and operate components, including local controls.
Reference(s): Simulator/Plant Design, AOP-001
Proposed References to be provided to applicants during examination:
Learning Objective: Objective 5 of AOP-001 Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (41.7 / 45.7)
Comments:
Meets the K/A because it asks the student to locate where they would operate the Lift Coil Disconnnect Swtiches for the Rod Control System.

correct

Lift Coil Disconnect Switches



Located in rear panel of RTGB, behind Rod Control section. This cabinet is locked.

Section A - Dropped Rod

Correct

Step	Description
1	If the reactor is Mode 2 or 3 and a rod drop occurs the operator will be directed to a series of steps that assure investigation and stabilization dependent on the Mode. Technical Specifications for Rod Alignment are only applicable in Modes 1 and 2. A dropped rod has little impact while in Mode 3, however could have impact in Modes 1 and 2. If in Mode 1 a plant transient is likely to have occurred, as well as particular concerns for ITS that may be present. This step provides diagnostic direction to the sections of the procedure applicable to Mode 1, Mode 2, or Mode 3.
N3	This note identifies the key required to open the Lift Coil Disconnect Panel door.
2-3	If the dropped rod is in a controlling bank (Normally Bank D) and the cause of the dropped rod was momentary, subsequent steps to control flux could result in inadvertently withdrawing the rod without adequate controls and safeguards. To prevent this condition, the Lift Coil Disconnect Switch for the affected rod is opened. If the rod is in a non-controlling bank the steps are by-passed.
C4	This Caution warns the Operator that corrective actions could result in movement of the dropped rod and, therefore, should not be performed until directed by this section.
4	This step notifies appropriate Engineering and I&C personnel to assist in recovery from the dropped rod. The rod should not be recovered until the cause has been determined and corrected.
5	ITS requires RCS to be borated within limits of the COLR SDM within 1 hour on a misaligned rod. A dropped rod is misaligned greater than allowable by ITS.
N6	This note reminds the Operator of restrictions on Reactor power with a misaligned rod.
6	This step ensures compliance with ITS 3.1.4 which limits <u>reactor</u> power to 70% if the rod cannot be realigned within 1 hour. This allows the use of rods to achieve 70%. If the Urgent Failure is present the operator is directed to borate and reduce turbine and reactor power below 70%.
7	Keeping the Load Dispatcher informed of the plant status and capability is essential so that he may make accurate predictions of power transmission and distribution needs.

Section A (Continued)

Incorrect

Step Description

25 Prior to recovering the dropped rod, the Group Step Counter is reset to zero. Since only the dropped rod will actually be changing position while the other rods in the bank will not be moved, the Group Step Counter will indicate the position of only the dropped rod during the recovery (until the rod is fully recovered). This step records the Group Step Counter reading to aid in realignment of the rod and to ensure the Group Step Counter indication reflects actual bank position when the realignment is completed.

N26 This note identifies the key required to open the Lift Coil Disconnect Panel door.

26 Only the dropped rod is to be moved during the recovery. In order for the other rods in the bank to not move upon a withdrawal signal, the lift coil switches must be turned off. The step is worded in a manner to allow for the switch for the affected rod to be open when the step is reached. The intent of the step is to place them in the required configuration regardless of switch position when the step is reached.

27 Recovering the rod will add positive reactivity to the Reactor. Since the Reactor is critical above the POAH, this will change RCS Tavg if Secondary loads are not adjusted or boron in the RCS is not adjusted. This step alerts the Operator to maintain the RCS temperature within the prescribed band during the recovery by either adjusting Secondary loads or adjusting RCS boron concentration. (SOER 84-02)

1N28 The Note alerts the Operator to the fact that Rod Control System Urgent Failure will be received. By doing this, the Operator will expect the alarm and not think that a new problem with the Rod Control System has occurred.

2N28 This note alerts the operator to the fact that the annunciator, APP-005-A5 may illuminate as the rod is recovered. If the rod is in a control bank the P-A Convertor will continue to receive counts as the dropped rod is withdrawn. Since this is fed back as an input to the alarm, when the total counts for the bank arrives at 225, the alarm will illuminate. The alarm will clear when the P-A Convertor is reset in subsequent steps.

3N28 This note alerts the operator to the fact that the annunciator, APP-005-F2 may reflash as the rod is recovered. As the rod is being withdrawn, the stepping action may result in the rod bottom bistable actuating and then resetting until the rod is stepped through these setpoints.

28 This step recovers the dropped rod at the rate recorded earlier.

59. 017 A1.01 001

Given the following plant conditions:

-The crew is in FRP-C.1, RESPONSE TO INADEQUATE CORE COOLING

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

From the ICCM Panel, the crew will monitor ___(1)___ IAW FRP-C.1. The minimum temperature at which core damage has occurred with RCP's running is ___(2)___ .

A. (1) RTD's
(2) 1200°F

B. (1) RTD's
(2) 2300°F

C.✓ (1) CETC's
(2) 1200°F

D. (1) CETC's
(2) 2300°F

The correct answer is C

A) Incorrect. RTD's is incorrect. Plausible because you can read RTD temperature from the ICCM Panel. 1200°F is correct.

B) Incorrect. RTD's is incorrect. Plausible because you can read RTD temperature from the ICCM Panel. 2300°F is incorrect. Plausible because this is the max temperature that can be read on the CETC's.

C) Correct. FRP-C.1 will have you monitor CETC's and this will be done from the ICCM Panel. With RCP's running and CETC's at 1200°F, This indicates that attempts to restore core cooling have failed and core damage can not be prevented. This is the point at which the crew would transition to the SAMG's.

D) Incorrect. FRP-C.1 will have you monitor CETC's and this will be done from the ICCM Panel. 2300°F is incorrect. Plausible because this is the max temperature that can be read on the CETC's.

Question: 59
Tier/Group: 2/2
K/A Importance Rating: RO 3.7 SRO 3.9
K/A: 017 In-Core Temperature Monitor (ITM) System
A1: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including:
A1.01: Core exit temperature
References: Sim/Plant design,FRP-C.1 and its basis document
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 5 of FRP-C.1
Question Source: New
Question History:
Question Cognitive Level: High/Analysis
10 CFR part 55: (CFR 41.5 / 45.7)

Meets the K/A because candidate must know what to monitor from the ICCM Panel in regards to core exit temperature for the given procedure.

Correct

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
7.	**Determine SI Accumulator Isolation Valve Status As Follows:**	
	a. Check SI ACCUM DISCHs - POWER AVAILABLE	a. Locally close the breakers for the following valves: • SI-865C, ACCUMULATOR C DISCHARGE (MCC-5, CMPT 9F) • SI-865A, ACCUMULATOR A DISCHARGE (MCC-5, CMPT 14F) • SI-865B, ACCUMULATOR B DISCHARGE (MCC-6, CMPT 10J)
	b. Check ACCUM DISCHs - OPEN • SI-865A • SI-865B • SI-865C	b. Open the ACCUM DISCH Valves unless closed after Accumulators discharged.
8.	**Check Core Exit T/Cs - LESS THAN 1200°F**	Go To Step 17.
9.	**Check RCP Status - ANY RUNNING**	Go To Step 11.
10.	**Reset SPDS <u>AND</u> Return To Procedure And Step In Effect**	
11.	**Check RVLIS Full Range Indication - GREATER THAN 41%**	Go To Step 13.
12.	**Reset SPDS <u>AND</u> Return To Procedure And Step In Effect**	
13.	**Check RVLIS Trend - STABLE <u>OR</u> LOWERING**	Observe <u>CAUTION</u> prior to Step 1 and Go To Step 1.
14.	**Check Core Exit T/Cs - LESS THAN 700°F**	Go To Step 16.
15.	**Reset SPDS <u>AND</u> Return To Procedure And Step In Effect**	
16.	**Check Core Exit T/C Trend - STABLE <u>OR</u> RISING**	Observe <u>CAUTION</u> prior to Step 1 and Go To Step 1.

RNP STEP	WOG STEP	BASIS/DIFFERENCES

Correct

34 20 WOG BASIS

PURPOSE: To determine if severe conditions exist that require a transition to the SAMGs

BASIS:

The Severe Accident Management Guidelines (SAMGs) are entered from the ERGs by the control room operators when core damage occurs. The ERG to SAMG transition uses, as part of the transition criteria, a core exit thermocouple temperature indication of greater than 1200°F to indicate the need to transition from the ERGs to the SAMGs. The 1200°F criteria for transition from the ERGs to the SAMGs is identical to the 1200°F criteria on the Core Cooling Critical Safety Function Status Tree.

If the operator enters this step and core exit TC temperatures are greater than 1200°F and increasing and RCPs are running in all available RCS cooling loops, the operator should transition to the SAMGs. This condition indicates that all attempts to restore core cooling have failed and core damage can not be prevented and the operator should go to the SAMGs.

If the operator enters this step and core exit TC temperatures are greater than 1200°F and decreasing, the operator should return to step 18 and continue attempts to reduce core exit temperature.

RNP DIFFERENCES/REASONS

The transition steps within the RNO have been split in order to make the transitions more clear. The RNP step has included an RNO transition for the case in which T/Cs are rising and not all available RCPs are running. The ERG did not include this transition; however this was implied in the basis description above. Normally it should not be possible to arrive at this step without having started all available RCPs, however if the right set of circumstances were to occur with temperatures decreasing below 1200°F following RCP start, then rising back above 1200°F this transition would be needed.

SSD DETERMINATION

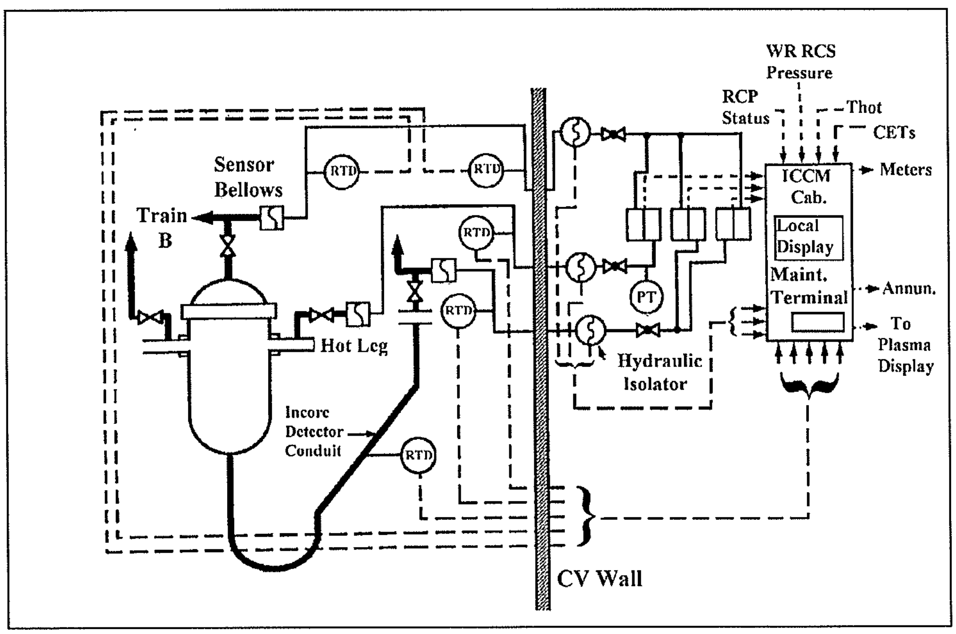
This is an SSD per criterion 10 and 11.

Incorrect

Building. The capillary sensing lines are sealed at the RCS end with a Sensor Bellows, which serves as hydraulic coupling for the pressure measurement. The filled capillary lines extend from the Sensor Bellows through containment penetration S-18 to the Hydraulic Isolators, which also provide hydraulic coupling, seal and isolation of lines. The capillary lines extend from the Hydraulic Isolators to the DP transmitters, where instrument valves are provided for isolation. A wide range pressure transmitter for each train is also connected to the filled capillary line coming from the hot legs.

The impulse lines inside containment are exposed to the containment temperature increase during an accident. Since the vertical runs of impulse lines form the reference legs for DP measurement, strap on RTD's are utilized to compensate the DP measurement during all phases of operation, including accident temperature changes.

Figure 3 Single Train of RVLIS

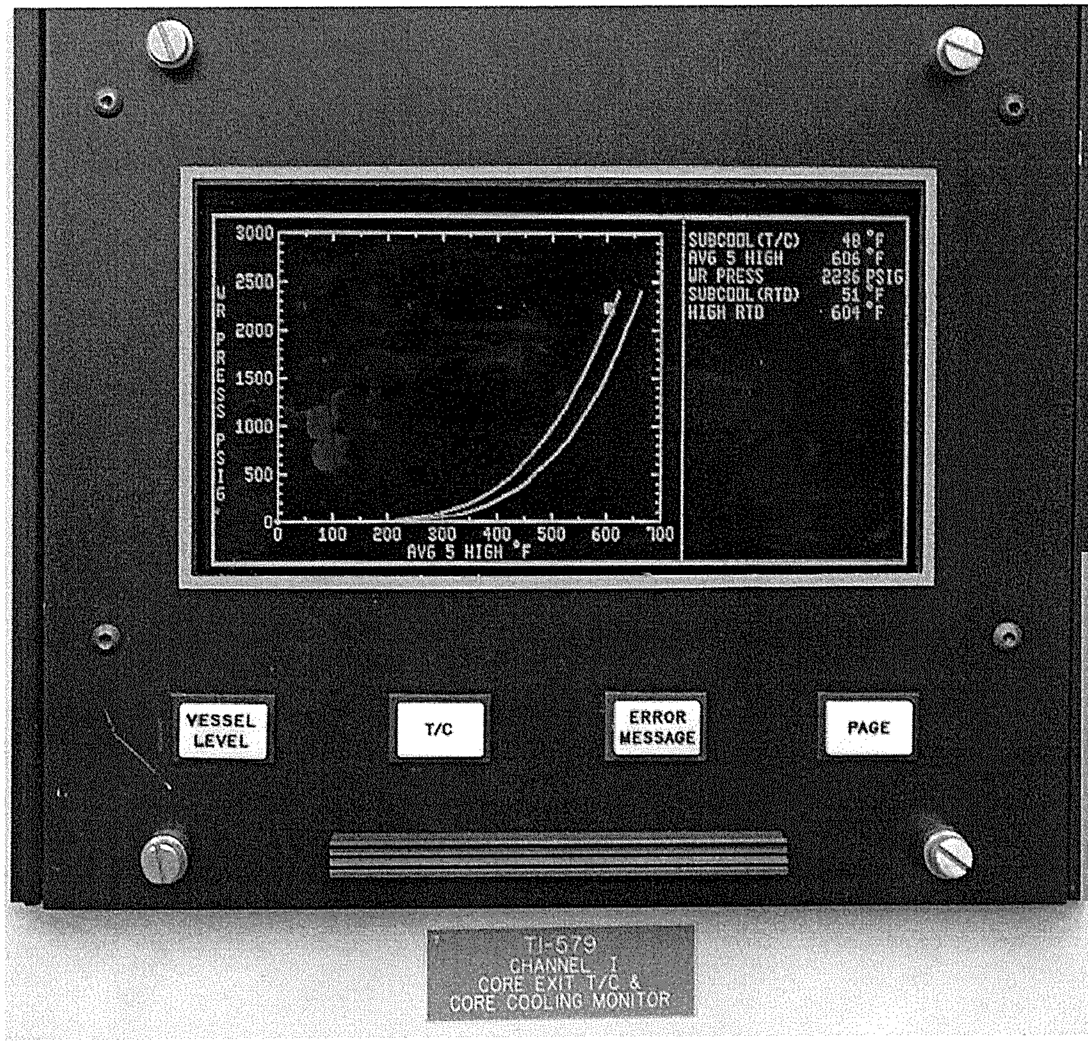


CETM (Core Exit Temperature Monitor)

The bottom mounted thermocouples provide core exit temperatures. They are located in the instrument thimble of the incore nuclear instrumentation. The CETs measure temperatures at the top of the actual core. They are used for core exit temperature indication and subcooled margin computation (this is the subcooling indication used for emergency procedures). They are capable of reading up to 2300°F.

Incorrect

Subcooled Margin Monitor (SMM)



Shows you can read RTDs from the ICCM Panel

Op. Fund.: "Monitoring" Use of multiple independent & diverse indications

60. 027 K2.01 001

Which ONE (1) of the following is the power supply for HVE-4, CV AIR IODINE REMOVAL EXHAUST FAN?

A. MCC-5

B.✓ MCC-6

C. MCC-9

D. MCC-10

The correct answer is B

A) Incorrect. Plausible since MCC-5 is a vital bus, however, HVE-4 is powered from MCC-6. HVE-3, CV Air Iodine Removal Exhaust Fan, is powered from MCC-5.

B) Correct. HVE-4 is powered from MCC-6 which is a vital bus.

C) Incorrect. Plausible since MCC-9 is a vital bus, however HVE-4 is powered from MCC-6. Additionally, MCC-9 is powered from MCC-6.

D) Incorrect. Plausible since MCC-10 is a vital bus, however HVE-4 is powered from MCC-6. HVE-3, CV Air Iodine Removal Exhaust Fan, is powered from MCC-5. MCC-10 is powered from MCC-5.

Question:60
Tier/Group: 2/2
K/A Importance Rating: RO 3.1 SRO 3.4
K/A: 027 Containment Iodine Removal System (CIRS)
    K2: Knowledge of bus power supplies to the following:
    K2.01: Fans
References: Sim/Plant design, EDP-003
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 6 of CVHVAC Lesson Plan
Question Source: RNP Bank
Question History: ILC-09 NRC exam, adjusted the question so that it only asks the power supply to HVE-4. Also adjusted the answers so that all the answers are MCC's that are vital busses.
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.7)

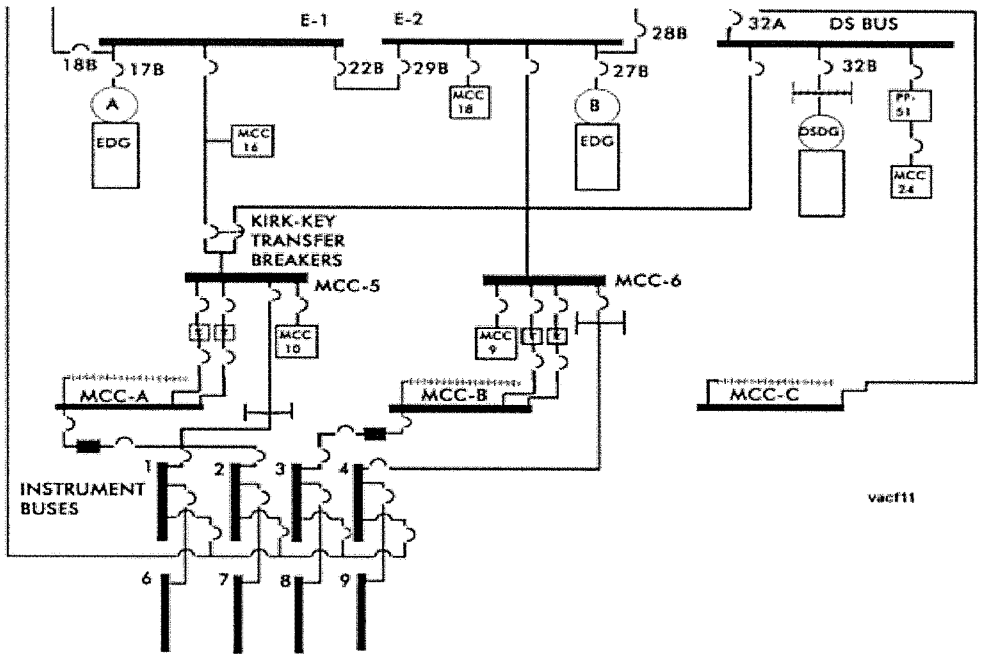
Meets the K/A because it asks the power supply to one of the CV Air Iodine Removal Exhaust Fan.

Correct

MCC-6			
POWER SUPPLY: 480V BUS E-2 (52/23C)		LOCATION: E-1/E-2 ROOM	
CMPT NO.	LOAD TITLE LOAD EDBS TAG NO.	CWD NO.	BKR EDBS NO.
17B	FUEL HANDLING BUILDING LOWER LEVEL EXHAUST GAS, R-20 R-20	86	52/MCC-6(17B)
17D	EDG FUEL OIL TRANSFER PUMP B DG-FO-XFER-PMP-B	953	52/MCC-6(17D)
17F	CV IODINE REMOVAL UNIT, HVE-4 HVE-4	522	52/MCC-6(17F)
17J	V6-34D, CV RECIRC COOLER HVH-4 SW OUTLET V6-34D	508	52/MCC-6(17J)
17M	V6-34C, CV RECIRC COOLER HVH-3 SW OUTLET V6-34C	506	52/MCC-6(17M)
18C	V6-33C, CV RECIRC COOLER HVH-3 SW INLET V6-33C	502	52/MCC-6(18C)
18F	V6-33D, CV RECIRC COOLER HVH-4 SW INLET V6-33D	503	52/MCC-6(18F)
18J	V6-33F, CV RECIRC COOLER HVH-2 SW SELECTIVE INLET V6-33F	505	52/MCC-6(18J)
18M	V1-8C, SDAFW PUMP STEAM ISOLATION MS-V1-8C	633A	52/MCC-6(18M)

incorrect

Figure 4 AC Distribution to DC Busses and Instrument Busses



120VAC INSTRUMENT BUS SYSTEM

The 120Vac instrumentation supply is split into 8 panels, known as Instrument Buses (IBs).

IBs 1, 2, 3, and 4 are 120Vac, single-phase (ø) panels with single pole 'EB' or 'EHB' frame 100 amp supply breakers at the panel and contain single pole 'EB' or 'EHB' frame, 30 amp feeder breakers.

Each IB has a normal and alternate power supply with its supply breakers mechanically interlocked such that only one breaker may be closed for incoming feeds.

IBs 2 and 3 are normally fed from 125Vdc motor control center (MCC) -A and MCC-B respectively via their single-Ø fixed frequency inverters. IBs 1 and 4 are normally fed from 480Vac MCC-5 and MCC-6 respectively via their 480/120Vac constant voltage transformers. The alternate power supply for IBs 1, 2, 3 & 4 is 120Vac MCC-8.

IBs 6, 7, 8, & 9 are fed from IBs 1, 2, 3, & 4, respectively. Hagan channels I, II, III, IV are powered from IB 1 through 4 respectively. Instrument and protection channels are color coded; Red for IB 1, White for IB 2, Blue for IB 3 and Yellow for IB 4.

61. 029 A4.04 001

Given the following plant conditions:

-One AO is in containment verifying a valve's position

-R-11, CV AIR & Plant Vent Particulate, alarms

-The crew has entered AOP-005, RADIATION MONITORING SYSTEM

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

The OAC will place the VLC switch to the ___(1)___ position and will then Hold the CV EVACUATION HORN for ___(2)___ seconds IAW AOP-005.

A. (1) OVERRIDE
(2) 5

B. (1) OVERRIDE
(2) 15

C. (1) EMERG
(2) 5

D.✓ (1) EMERG
(2) 15

The correct answer is D.

A) Incorrect. The VLC switch will be placed in the EMERG position. OVERRIDE is plausible since this is a selectable postion for this switch. Five seconds is incorrect. Plausible because this is how long they sound the ALL CLEAR alarm in AOP-041 after the fire is reported out.

B) Incorrect. The VLC switch will be placed in the EMERG position. OVERRIDE is plausible since this is a selectable postion for this switch. 15 seconds is correct IAW AOP-005.

C) Incorrect. The VLC switch will be placed in the EMERG position IAW AOP-005. Five seconds is incorrect. Plausible because this is how long they sound the ALL CLEAR alarm in AOP-041 after the fire is reported out.

D) Correct. The VLC switch will be placed in the EMERG position IAW AOP-005. 15 seconds is correct IAW AOP-005.

Question: 61
Tier/Group: 2/2
K/A Importance Rating: RO 3.5 SRO 3.6

K/A: 029 Containment Purge System (CPS)
A4: Ability to manually operate and/or monitor in the control room:
A4.04: Containment evacuation signal

Reference(s): Sim/Plant Design, AOP-005
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 3 of AOP-005 lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: 41.7 / 45.5 to 45.8
Comments:

Meets the K/A because the student must know the correct position to place the VLC switch in and know for how long to hold the CV EVACUATION HORN for.

correct

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED

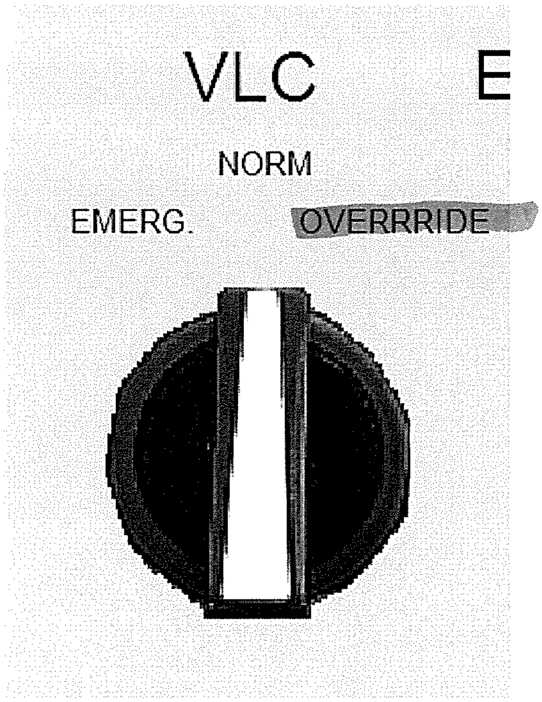
ATTACHMENT 12

PROCESS MONITOR R-11/R-12 - CV AIR & PLANT VENT

(Page 2 of 3)

STEP	INSTRUCTIONS	RESPONSE NOT OBTAINED
7.	Check Personnel - IN CV	Go To Step 13.
8.	Place VLC Switch To EMERG Position	
9.	Depress And Hold CV EVACUATION HORN Pushbutton For 15 SECONDS	
10.	Announce The Following Over Plant PA System: "ATTENTION ALL PERSONNEL. ATTENTION ALL PERSONNEL. A HIGH RADIATION ALARM HAS BEEN RECEIVED ON CV VENT PROCESS MONITOR, R-11 (R-12). ALL NON-ESSENTIAL PERSONNEL EVACUATE CV UNTIL FURTHER NOTICE"	
11.	Repeat CV Evacuation Announcement Over PA System	
12.	Place VLC Switch To NORM Position	
13.	Check CONTAINMENT VENTILATION ISOLATION Valves - CLOSED	Perform the following: a. Depress H.V. OFF on R-11 <u>OR</u> R-12 to initiate Containment Ventilation Isolation. b. <u>IF</u> any CONTAINMENT VENTILATION ISOLATION Valve fails to close, <u>THEN</u> locally verify penetration is isolated from outside CV.
14.	Place The Following CV IODINE REMOVAL FAN Control Switches To PREPURGE Position: • HVE-3 • HVE-4	

incorrect



Incorrect

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Sounding the ALL CLEAR should not be performed until the fire area atmosphere has been evaluated for normal access.

40. When Recommended By FBIC, Then Sound The ALL CLEAR Alarm For 5 Seconds <u>AND</u> Announce The Status Of The Fire.

41. Verify The VLC Switch For The PA System Is In The NORMAL Position.

42. Perform FP-001 Section 8.4, Unit 2 Control Room Post Fire Activities.

43. Notify RES Duty Manager To Notify RES Fire Protection Staff And Other RES Personnel As Necessary. (CR 96-01227)

44. Contact Plant Operations Staff To Determine Subsequent Recovery Actions.

45. Refer To FP-002 For Fire Incident Report.

46. Return To Procedure And Step In Effect.

62. 033 K4.01 001

Given the following plant conditions:

-The reactor is at 100% RTP

-APP-036-B6, SPENT FUEL PIT LO LEVEL, alarm comes in

-AO reports that APP-036-B6 is a valid alarm

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

APP-036-B6 alarm setpoint is ___(1)___. The crew will fill the Spent Fuel Pit level using water from the ___(2)___ .

A. (1) 35ft
(2) RWST

B. (1) 36ft, 2.5 inches
(2) RWST

C. (1) 35ft
(2) Demineralized Water System

D✓ (1) 36ft, 2.5 inches
(2) Demineralized Water System

The correct answer is D

A) Incorrect. 35ft is incorrect. Plausible since this level corresponds to 21ft above the irradiated fuel assemblies which is an LCO requirement. The crew will not use the RWST to fill the Spent Fuel Pit. Plausible since this method is used, however, only for large level additions. Also, you cannot fill the Spent Fuel Pit using the RWST in MODES 1, 2, 3, 4.

B) Incorrect. 36ft, 2.5 inches is correct. The crew will not use the RWST to fill the Spent Fuel Pit. Plausible since this method is used, however, only for large level additions. Also, you cannot fill the Spent Fuel Pit using the RWST in MODES 1, 2, 3, 4.

C) Incorrect. 35ft is incorrect. Plausible since this level corresponds to 21ft above the irradiated fuel assemblies which is an LCO requirement. The crew will use the Demineralized Water System to fill the Spent Fuel Pit.

D) Correct. 36ft, 2.5 inches is correct. The crew will use the Demineralized Water System to fill the Spent Fuel Pit per OP-910.

Question: 62
Tier/Group: 2/2
K/A Importance Rating: RO 2.9 SRO 3.2
K/A: 033 Spent Fuel Pool Cooling System (SFPCS)
K4: Knowledge of design feature(s) and/or interlock(s) which provide for the following:
K4.01: Maintenance of spent fuel level
References: Sim/Plant design, OP-910, APP-036-B6
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 10a of Spent Fuel Pit Cooling Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.7)

Meets the K/A because this alarm is a design feature that prevents the plant from draining Spent Fuel Pit level too low. The student must also know from what source to refill the Spent Fuel Pit for this situation which is the maintenance of the Spent Fuel Level.

Correct

ALARM

SPENT FUEL PIT LO LEVEL

AUTOMATIC ACTIONS

1. None Applicable

CAUSE

1. Transfer of water from SFP to RWST
2. Spent Fuel Pit Leaking
3. Valves in SFP Loop Leaking
4. Improper Valve Alignment

OBSERVATIONS

1. None Applicable

ACTIONS — CK (✓)

1. **IF** loss is due to hostile action, **THEN REFER** to AOP-036, SFP Events. _____
2. **IF** loss is **NOT** due to hostile action, **THEN PERFORM** the following:
   a. **DISPATCH** an operator to investigate local SFP conditions. _____
   b. **IF** level is less than 34 feet, **THEN REFER** to AOP-036, SFP Events. _____
   c. **IF** SFP level is low, **THEN RAISE** SFP level using OP-910. _____
   d. **REFER** to ITS 3.7.12. _____

DEVICE/SETPOINTS

1. LA-651 / 36 feet 2 1/2 inches

POSSIBLE PLANT EFFECTS

1. High radiation level in the SFP

REFERENCES

1. ITS LCO 3.7.12
2. OP-910, Spent Fuel Pit Cooling and Purification System
3. AOP-036, SFP Events
4. CWD B-190628, Sheet 488, Cable AW

cocccU

INIT

8.4 INFREQUENT OPERATION

8.4.1 Filling the Spent Fuel Pit From the Demineralized Water System for Minor Level Adjustment

NOTE:	Large level additions to the SFP are performed in accordance with OP-913.

1. Initial Conditions

   a. This revision has been verified to be the latest revision available. ______ ____
   Date

   b. Demineralized Water System is aligned and in operation in accordance with OP-915-1. ____

   c. E&RC has been notified that the SFP will be filled. ____

2. Instructions

NOTE:	DW-269, DEMIN WATER TO FUEL HANDLING BLDG SERVICE CONN, is located at the North West corner of the SFP.

   a. **CONNECT** a hose to DW-269, DEMIN WATER TO FUEL HANDLING BLDG SERVICE CONN. ____

CAUTION

Do **NOT** allow hose to come in contact with SFP water at anytime to prevent possible back flow of SFP water into Demineralized Water header.

   b. **SECURE OR HOLD** end of hose such that the hose does **NOT** come in contact with SFP water. ____

B 3.7 PLANT SYSTEMS

B 3.7.12 Fuel Storage Pool Water Level

Incorrect

BASES

BACKGROUND

The minimum water level of 21 ft above the top of the fuel in the fuel storage pool exceeds the assumptions of iodine decontamination factors following a fuel handling accident and bounds the sensible heat sink assumptions used in "time to boil" calculations. With the fuel storage racks installed in the spent fuel storage pool, a water level 21 ft above the fuel corresponds to approximately 35 ft pool water depth. The specified water level shields and minimizes the general area dose when the storage racks are filled to their maximum capacity. The water also provides shielding during the movement of spent fuel.

A general description of the fuel storage pool design is given in the UFSAR, Section 9.1.2 (Ref. 1). A description of the Spent Fuel Pool Cooling and Cleanup System is given in the UFSAR, Section 9.1.3 (Ref. 2). The assumptions of the fuel handling accident are given in the UFSAR, Section 15.7.4 (Ref. 3).

APPLICABLE SAFETY ANALYSES

The minimum water level in the fuel storage pool meets the assumptions of the fuel handling accident described in Reference 3. The resultant 2 hour thyroid dose per person at the exclusion area boundary is a small fraction of the 10 CFR 50.67 (Ref. 4) limits.

According to the fuel storage pool fuel handling accident analysis (Ref. 3), the minimum level of 21 ft over the top of irradiated fuel assemblies seated in the storage racks exceeds the submergence requirements necessary to obtain the assumed decontamination factor (DF) for inorganic iodines released from damaged fuel as a result of the accident.

The fuel storage pool water level satisfies Criterion 2 of the NRC Policy Statement.

LCO

The fuel storage pool water level is required to be ≥ 21 ft over the top of irradiated fuel assemblies seated in the

(continued)

Incorrect

9.4. INFREQUENT OPERATION

INIT

9.4.1. Adding Borated Makeup to the Spent Fuel Pit (SFP) from the Refueling Water Storage Tank (RWST) while SFP is on purification through Deminerlizer and Filter.

1. Initial Conditions

   a. This revision has been verified to be the latest revision available. _____

   b. All the prerequisites of Section 5.0 have been completed. _____

CAUTION
Opening SFPC-805A, RWP PUMP SUCTION FROM RWST, makes the RWST inoperable per ITS LCO 3.5.4 due to the purification loop not being seismically qualified. This section **SHALL NOT** be performed in Modes 1 through 4.

   c. **RECORD** current Plant Mode:

      Plant Mode_________________________ _____

   d. **IF** the current plant Mode is 1, 2, 3, or 4, **THEN EXIT** this procedure. [CAPR 00524619] _____

   e. The RWST contains Borated Water with the concentration greater than or equal to 1950 ppm. _____

   f. **IF** needed to provide makeup to the RWST under current plant conditions,
      THEN the Chemical and Volume Control System (CVCS) is available as per OP-301. _____

63. 035 K5.01 001

Given the following plant conditions:

-The reactor is at 100% RTP

-The OAO reports a steam leak near the Main Steam Isolation Valves

-The following indications are noted in the Control Room:
- T_{avg} is lowering
- Steam flow and feed flow have risen
- Power Limit Warning alarm on ERFIS has been received
- Reactor power is 100.3% and slowly rising

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

The time in core life that will result in the smallest reactivity excursion is ___(1)___ . The Reactor is required to be operated at less than or equal to ___(2)___ MW thermal IAW OMM-001-2, SHIFT ROUTINES AND OPERATING PRACTICES.

A. (1) BOL
(2) 2300

B.✓ (1) BOL
(2) 2339

C. (1) EOL
(2) 2300

D. (1) EOL
(2) 2339

The correct answer is B

A) Incorrect. BOL does provide you with the smallest reactivity excursion for this transient. 2300 is incorrect. Plausible because if FWUFM was not available(Normally in service) we would be limited to 2300 MW thermal.

B) Correct. BOL does provide you with the smallest reactivity excursion for this transient. At BOL the moderator temperature coefficient (MTC) is smaller (or less negative) than at EOL. This causes a smaller change in reactivity for a given change in RCS temperature. 2339 is correct. FWUFM is available(Normally in service) and therefore we are allowed to operate up to 2339 MW thermal.

C) Incorrect. EOL provides you with the largest reactivity excursion for this transient. At EOL the moderator temperature coefficient (MTC) is much larger than at BOL, therefore you get a much larger change in reactivity for a given change in RCS temperature. 2300 is incorrect. Plausible because if FWUFM was not available(Normally in service) we would be limited to 2300 MW thermal.

D) Incorrect. EOL provides you with the largest reactivity excursion for this transient. At EOL the moderator temperature coefficient (MTC) is much larger than at BOL, therefore you get a much larger change in reactivity for a given change in RCS temperature. 2339 is correct. FWUFM is available(Normally in service) and therefore we are allowed to operate up to 2339 MW thermal.

Question: 63
Tier/Group: 2/2
K/A Importance Rating: RO 3.4 SRO 3.9
K/A: 035 Steam Generator System (S/GS)
K5: Knowledge of operational implications of the following concepts as the apply to the S/GS:
K5.01: Effect of secondary parameters, pressure, and temperature on reactivity
References: Sim/Plant design, GFES Reactory Theory lesson, OMM-001-2
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 8 of EOP-E-3 Lesson Plan
Question Source: NEW
Question History: NEW
Question Cognitive Level: High
10 CFR part 55: (CFR 41.5 / 45.7)

Meets the K/A because the student has to determine how a steam leak(secondary parameters) will affect reactivity at EOL and BOL. The operational implication is knowing the Thermal MW rating that they cannot exceed.

9.2 Continuous Calorimetric Program

Correct

1. The Continuous Calorimetric Program (CCP) is used to monitor core thermal power during periods of STEADY STATE operation. The CCP does **NOT** account for changing volumes or changes in stored energy. In addition, averaging routines built into the CCP will require as much as 5 minutes before a change in an input value is fully reflected in the display.
2. ERFIS simultaneously calculates calorimetrics using FWUFM, Feedwater Flow and Steam Flow. Logic built into the Continuous Calorimetric program (CALO) will cause the calculation to halt and the result to be "0" and labeled NCAL for most situations **NOT** allowed by the uncertainty calculations.
   - FWUFM CALO should be used when it is available.
   - **IF** FWUFM CALO is **NOT** available, **AND** the actions of TRM 3.25 Condition B are **NOT** completed, **THEN** the following are required until FWUFM is restored:
     - Use NIN0001, Average Power Range Percent Power.
     - **IF** NIN0001, Average Power Range Percent Power is **NOT** available, **THEN** use the highest individual one minute average NI Power on ERFIS (NIN0041M, NIN0042M, NIN0043M, **OR** NIN0044M).
     - **IF** ERFIS is **NOT** available, **THEN** use the highest reading Power Range Nuclear Instrument on N-45.
     - Maintain less than or equal to 100% reactor power using indication described above.
     - **NO** positive reactivity changes are permitted except for power reductions (adds positive reactivity due to power defect) **OR** forward flowing of AFW to the Steam Generators (in accordance with approved plant procedures which account for the positive reactivity addition by requiring a power reduction).
3. Diverse indications of Reactor power; such as NIs, Loop Delta T, Unit Load, and Turbine 1st Stage Pressure, should be monitored at all times to ensure the Reactor is operated at less than or equal to 2339 MW thermal when the FWUFM System is in service and 2300 MW thermal when FWUFM is **NOT** in service as required by TRM 3.25 limitation. As specified in EST-155, at 100% RTP (2339 MW thermal) the allowed range for ΔT is 57°F to 58°F and the allowed range for Turbine 1st Stage Pressure is 654.33 psi to 668.73 psi. Diverse indications of Reactor Power during transient conditions may be affected by changes in RCS temperature resulting in NI shielding and changes in secondary efficiency affecting Unit Load and Turbine 1st Stage Pressure. Loop Delta T indications are the least affected by RCS temperature changes and secondary efficiency changes. Due to the TR-412 trend display scale and normal fluctuation in the ΔT digital displays of ΔT and ΔT power it may be difficult to identify small changes in power. Steam Flow Calorimetric is an effective tool to validate LEFM response, especially during transient conditions. LEFM increases that are solely a result of increases in feed flow would **NOT** be seen as a power increase using the Steam Flow Calorimetric. **[R5] [R6]**
4. **WHEN** above 98% power, **THEN** reactivity should **NOT** be changed by more than one method at a time (control rods, turbine steam demand, or boron dilution), to prevent exceeding 100% power. (OPEX 266297)

9.2 Continuous Calorimetric Program

Incorrect

1. The Continuous Calorimetric Program (CCP) is used to monitor core thermal power during periods of STEADY STATE operation. The CCP does **NOT** account for changing volumes or changes in stored energy. In addition, averaging routines built into the CCP will require as much as 5 minutes before a change in an input value is fully reflected in the display.

2. ERFIS simultaneously calculates calorimetrics using FWUFM, Feedwater Flow and Steam Flow. Logic built into the Continuous Calorimetric program (CALO) will cause the calculation to halt and the result to be "0" and labeled NCAL for most situations **NOT** allowed by the uncertainty calculations.

   - FWUFM CALO should be used when it is available.
   - **IF** FWUFM CALO is **NOT** available, **AND** the actions of TRM 3.25 Condition B are **NOT** completed, **THEN** the following are required until FWUFM is restored:
     - Use NIN0001, Average Power Range Percent Power.
     - **IF** NIN0001, Average Power Range Percent Power is **NOT** available, **THEN** use the highest individual one minute average NI Power on ERFIS (NIN0041M, NIN0042M, NIN0043M, **OR** NIN0044M).
     - **IF** ERFIS is **NOT** available, **THEN** use the highest reading Power Range Nuclear Instrument on N-45.
     - Maintain less than or equal to 100% reactor power using indication described above.
     - **NO** positive reactivity changes are permitted except for power reductions (adds positive reactivity due to power defect) **OR** forward flowing of AFW to the Steam Generators (in accordance with approved plant procedures which account for the positive reactivity addition by requiring a power reduction).

3. Diverse indications of Reactor power; such as NIs, Loop Delta T, Unit Load, and Turbine 1st Stage Pressure, should be monitored at all times to ensure the Reactor is operated at less than or equal to 2339 MW thermal when the FWUFM System is in service and 2300 MW thermal when FWUFM is **NOT** in service as required by TRM 3.25 limitation. As specified in EST-155, at 100% RTP (2339 MW thermal) the allowed range for ΔT is 57°F to 58°F and the allowed range for Turbine 1st Stage Pressure is 654.33 psi to 668.73 psi. Diverse indications of Reactor Power during transient conditions may be affected by changes in RCS temperature resulting in NI shielding and changes in secondary efficiency affecting Unit Load and Turbine 1st Stage Pressure. Loop Delta T indications are the least affected by RCS temperature changes and secondary efficiency changes. Due to the TR-412 trend display scale and normal fluctuation in the ΔT digital displays of ΔT and ΔT power it may be difficult to identify small changes in power. Steam Flow Calorimetric is an effective tool to validate LEFM response, especially during transient conditions. LEFM increases that are solely a result of increases in feed flow would **NOT** be seen as a power increase using the Steam Flow Calorimetric. **[R5] [R6]**

4. **WHEN** above 98% power, **THEN** reactivity should **NOT** be changed by more than one method at a time (control rods, turbine steam demand, or boron dilution), to prevent exceeding 100% power. (OPEX 266297)

64. 015 A3.02 001

Given the following plant conditions:

-The plant is performing a power ascension and is at 8% RTP

Which ONE (1) of the following completes the statements below? lsb 8/6/2013

The REACTOR TRIP BLOCK P-7 status light will ~~illuminate~~ EXTINGUISH when __(1)__ Power Range NI's read above 10%. One of the Reactor Trips that this enables is __(2)__ .

A.✓ (1) 2/4
(2) PZR High Level

B. (1) 3/4
(2) PZR High Level

C. (1) 2/4
(2) PZR High Pressure

D. (1) 3/4
(2) PZR High Pressure

The correct answer is A

A) Correct. It takes 2/4 Power Range NI's to read above 10% to get the REACTOR TRIP BLOCK P-7 status light to illuminate. PZR High Level is a reactor trip that is enabled when above P-7.

B) Incorrect. 3/4 Power Range NI's is incorrect. Plausible since this is the number of Power Range NI's that it takes to disable the at power Reactor trips. PZR High Level is a reactor trip that is enabled when above P-7.

C) Incorrect. It takes 2/4 Power Range NI's to read above 10% to get the REACTOR TRIP BLOCK P-7 status light to illuminate. Plausible because the PZR High Pressure is a reactor trip, however, it is not enabled by power being above P-7.

D) Incorrect. 3/4 Power Range NI's is incorrect. Plausible since this is the number of Power Range NI's that it takes to disable the at power Reactor trips. Plausible because the PZR High Pressure is a reactor trip, however, it is not enabled by power being above P-7.

Question: 64
Tier/Group: 2/2
K/A Importance Rating: RO 3.7SRO 3.9

K/A: 015 Nuclear Instrumentation System (NIS)
A3: Ability to monitor automatic operation of the NIS, including:
A3.02: Annunciator and alarm signals

Reference(s): Simulator/plant design, Student Text for NIS
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 10F of NIS Lesson Plan
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR Part 55 Content: (41.7 / 45.5)
Comments:

This question meets the K/A because the student must know when to expect the REACTOR TRIP BLOCK P-7 status light(Annuniciator) to illuminate since this happens automatically.

Correct

Purpose and Entry Conditions

(Page 2 of 4)

2. **SYMPTOMS AND ENTRY CONDITIONS**

   a. The following are symptoms that require a reactor trip, if one has not occurred:

REACTOR TRIP SIGNAL	LOGIC (INTERLOCK)	SETPOINT
SR High Flux	1/2 (P-6 and P-10)	10^5 CPS
IR High Flux	1/2 (P-10)	Current equal to 25%
PR High Flux Low Range	2/4 (P-10)	24%
PR High Flux High Range	2/4	108%
PZR High Pressure	2/3	2376 psig
PZR Low Pressure	2/3 (P-7)	1844 psig
PZR High Level	2/3 (P-7)	91%
Low RCS Flow	2/3 on 2/3 (P-7) 1/3 (P-8)	94.68% rated flow
RCP Breaker	1/1 on 2/3 (P-7) 1/3 (P-8)	Open
RCP Bus Undervoltage	2/3 busses (P-7)	75% (3120 volts)
Over Temperature ΔT	2/3	Variable
Over Power ΔT	2/3	Variable
Safety Injection	Auto or Manual	N/A
Turbine Trip	2/2 SV or 2/3 AST (P-8)	N/A
Low-Low S/G Level	2/3 on 1/3	16%
Low S/G Level With Steam Flow > Feed Flow	1/2 on 1/3 1/2 on 1/3	30% 0.64×10^6 lbm/hr

(CONTINUED NEXT PAGE)

Incorrect

Purpose and Entry Conditions

(Page 2 of 4)

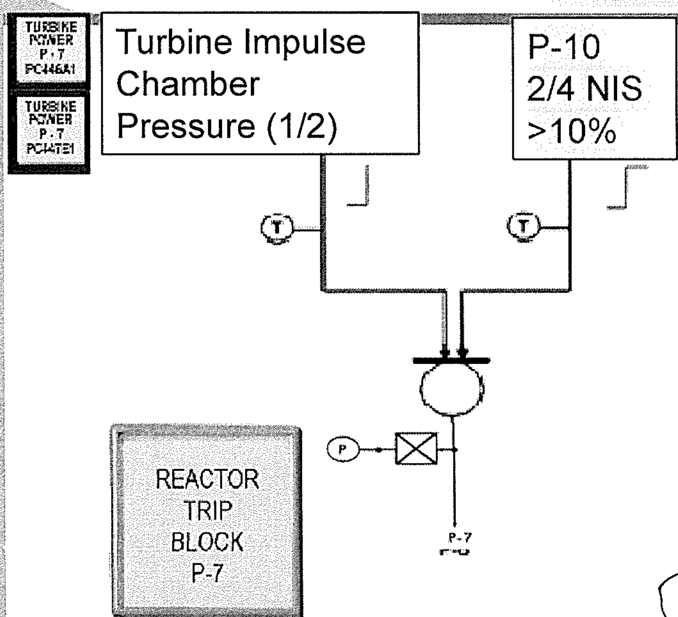
2. **SYMPTOMS AND ENTRY CONDITIONS**

   a. The following are symptoms that require a reactor trip, if one has not occurred:

REACTOR TRIP SIGNAL	LOGIC (INTERLOCK)	SETPOINT
SR High Flux	1/2 (P-6 and P-10)	10^5 CPS
IR High Flux	1/2 (P-10)	Current equal to 25%
PR High Flux Low Range	2/4 (P-10)	24%
PR High Flux High Range	2/4	108%
PZR High Pressure	2/3	2376 psig
PZR Low Pressure	2/3 (P-7)	1844 psig
PZR High Level	2/3 (P-7)	91%
Low RCS Flow	2/3 on 2/3 (P-7) 1/3 (P-8)	94.68% rated flow
RCP Breaker	1/1 on 2/3 (P-7) 1/3 (P-8)	Open
RCP Bus Undervoltage	2/3 busses (P-7)	75% (3120 volts)
Over Temperature ΔT	2/3	Variable
Over Power ΔT	2/3	Variable
Safety Injection	Auto or Manual	N/A
Turbine Trip	2/2 SV or 2/3 AST (P-8)	N/A
Low-Low S/G Level	2/3 on 1/3	16%
Low S/G Level With Steam Flow > Feed Flow	1/2 on 1/3 1/2 on 1/3	30% 0.64×10^6 lbm/hr

(CONTINUED NEXT PAGE)

P-7



1/2 Turbine First Stage Pressure above setpoint (10%)
OR
2/4 Power Ranges above setpoint **(10% from P-10)**

Enables the "at-power" Rx trips:

1. RCS Low Flow
2. RCP Breakers Open
3. Under voltage
4. PZR low Pressure
5. PZR High Level

correct

2/2 Turbine First Stage Pressure below setpoint **(10%)**
AND
3/4 Power Ranges below setpoint **(10% from P-10)**

Blocks the "at-power" Rx trips

incorre

65. 068 K1.07 002

Which of the following discharge to the Reactor Coolant Drain Tank?

A. Charging pump seals

B. RCV-609, CC SURGE TANK VENT

C. CVC-203 A/B, LETDOWN RELIEF VALVES

D✓ CVC-389, EXCESS LETDOWN DIVERSION

The correct answer is D

A) Incorrect. Charging pumps seals drain to the AUX BLDG Sump Tank which drains to the WHUT. Plausible since this is another tank in the Liquid Rad Waste System.

B) Incorrect. RCV-609 drains to the WHUT. Plausible since this is another tank in the Liquid Rad Waste System.

C) Incorrect. CVC-203 A/B relieve to the PRT, which is in containment.

D) Correct. CVC-389 if leaking by, will cause RCDT level to rise.

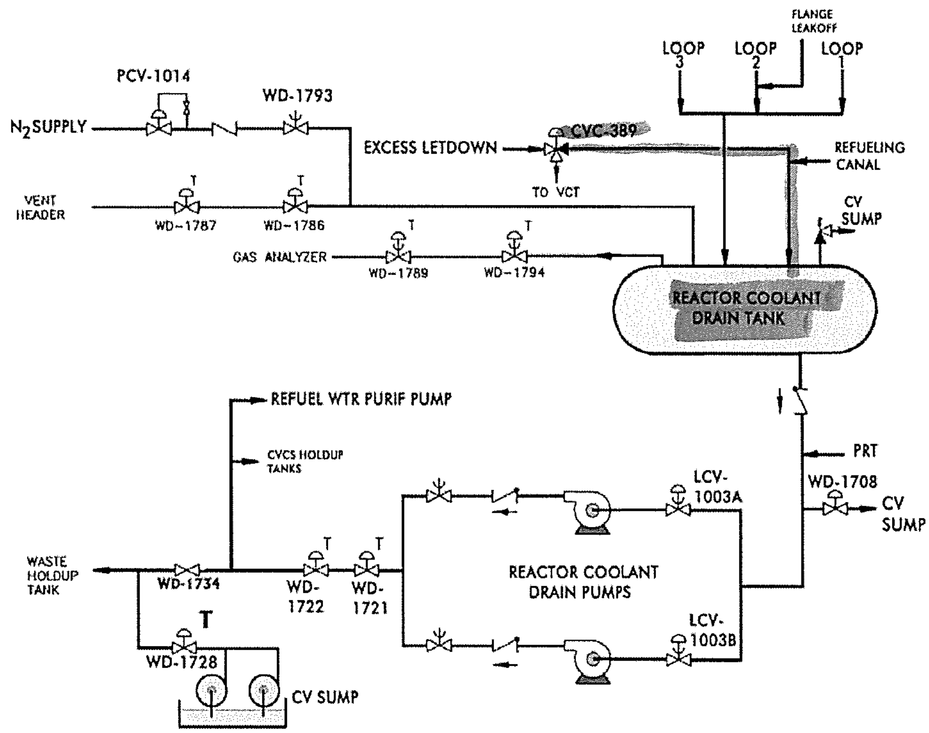
Question: 65
Tier/Group: 2/2
K/A Importance Rating: RO 2.7 SRO 2.9
K/A: 068 Liquid Radwaste System (LRS)
K1: Knowledge of the physical connections and/or cause effect relationships between the Liquid Radwaste System and the following systems:
K1.07: Sources of liquid wastes for LRS
Reference(s): Sim/Plant design, Student Text for Waste Disposal System
Proposed References to be provided to applicants during examination: None
Learning Objective: Objective 2 of Waste Disposal System lesson plan
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
Comments:

This question meets the K/A because the student must know the physical connections going into the RCDT.

Correct

controlled valve (from the waste disposal panel) can be opened to drain the Reactor Coolant Drain Tank to the Containment Vessel sump. The Reactor Coolant Drain Tank is vented to the vent header and is sampled by the gas analyzer.

Figure 2 Reactor Coolant Drain Tank

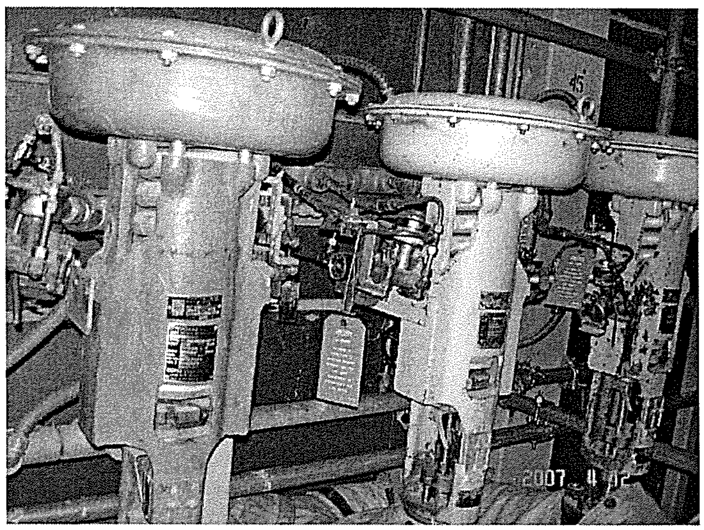


Incorrect

LETDOWN ORIFICE ISOLATION VALVES (CVC-200A, 200B, 200C) AND LETDOWN RELIEF VALVES (CVC-203A, 203B)

Three air operated valves, controlled by three OPEN/CLOSE RTGB switches, are provided to control which letdown orifice(s) is/are in service. One orifice (CVC-200A) will pass 45 gpm and the other two orifices (CVC-200B & C) will pass 60 gpm each when the RCS is at normal pressure and letdown pressure is adjusted to approximately 300 psig. Care should be taken not to exceed design flow rate through the demineralizers. These valves are located next to the letdown orifices just outside the Regenerative Heat Exchanger cubicle. These valves will close on a Phase "A" containment isolation signal (T signal), loss of air, loss of electrical power, or can be manually closed.

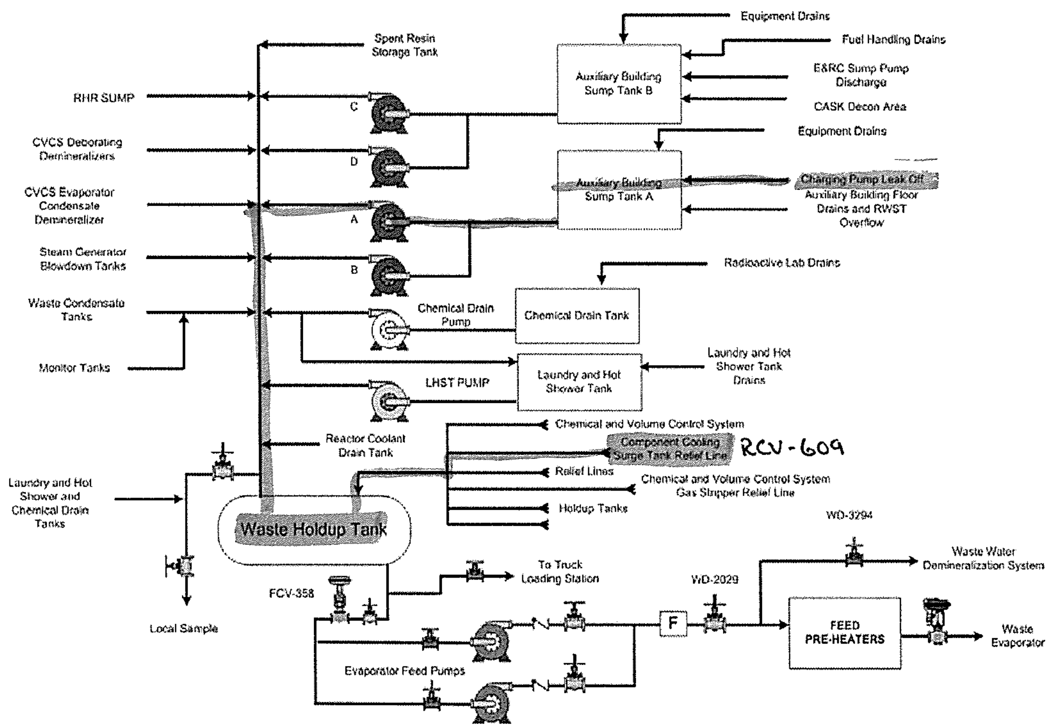
Figure 42 CVC-200A, B, C



When establishing letdown, valves CVC-204A and CVC-204B must be open before opening an orifice isolation valve to prevent actuation of relief valves CVC-203A and CVC-203B. CVC-203A & B relieve to the Pressurizer Relief Tank (PRT). Their setpoints are 500 and 630 psig, respectively. Staggered setpoints prevent relief valve chattering from low relief flowrate. There is an RTD on the relief discharge line to indicate a open (or leaking) relief valve. At 200°F, this RTD causes annunciator APP-001-E6, LP LTDN RELIEF HI TEMP, to alarm.

incorrect

Figure 1 Waste Holdup Tank Loads



66. G2.1.14 001

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

IAW OPS-NGGC-1000, FLEET CONDUCT OF OPERATIONS, a plant announcement will be made for the starting of ___(1)___ motors. The announcement will include ___(2)___ .

A. (1) all
   (2) only the applicable motor

B. (1) only large
   (2) only the applicable motor

C. (1) all
   (2) the applicable motor and its switchgear

D✓ (1) only large
   (2) the applicable motor and its switchgear

The correct answer is D

A) Incorrect. OPS-NGGC-1000 says you will make a plant announcement for starts (or closing breakers) of large motors. The announcement should include direction to stand clear of the applicable pump and switchgear as applicable. Both parts are plausible if they do not know the requirements for plant announcements.

B) Incorrect. Only large motors is correct. The announcement should include direction to stand clear of the applicable pump and switchgear as applicable. Plausible if they do not know the requirements for plant announcements.

C) Incorrect. OPS-NGGC-1000 says you will make a plant announcement for starts (or closing breakers) of large motors. The announcement will include the applicable pump and its switchgear.

D) Correct. OPS-NGGC-1000 says you will make a plant announcement for starts (or closing breakers) of large motors. The announcement should include direction to stand clear of the applicable pump and switchgear as applicable. Both parts are plausible if they do not know the requirements for plant announcements.

NRC Exam

Question: 66
Tier/Group: 3
K/A Importance Rating: RO 3.1 SRO 3.1
K/A: G2.1.7: Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc.
References: OPS-NGGC-1000
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 43.5 / 45.12)

Meets the K/A because the operator must know the criteria (Which motors) that require plant wide announcements and what must be stated in those announcements.

9.6.5 Control Board Manipulations

Operators ensure that the correct components are operated, and system response is as expected when performing system operations. This will ensure that equipment is operated properly, and that plant safety and efficiency are ensured.

1. Standard
   a. Manipulation of Control Room components may only be made by:
      1) Actively licensed operators.
      2) Inactive licensed operators enrolled in a license activation watch and under instruction of an active license.
      3) Licensed Reactor Operator or Senior Reactor Operator candidates in an approved training program (Licensed Operator Initial Training) who have completed the training requirements for assignment to Control Room On The Job Training may operate controls under the direction and presence of a licensed on-shift operator.
      4) During Event Procedures actions, the STA may only manipulate components necessary to obtain information necessary to perform the assessment role.

   b. Manipulation of satellite control room or system controls will be made only by qualified personnel. Personnel under OJT may manipulate controls under the supervision of a watchstander qualified individual.

   c. The CRS, STA and Shift Manager shall not manipulate controls on the control board due to the potential for degrading oversight of plant operations unless the action is immediately necessary for the protection of the health and safety of the public or plant personnel, or prevent major equipment damage. [SOER 96-1, Rec. 1.c]

2. **Expectations**
   a. Component Manipulation
      1) Components are not manipulated unless under the control of an approved document such as a procedure, surveillance, clearance, other controlling document or meets the requirements for simple evolutions per Attachment 4.
      2) The CRS is notified of impending system/component operation prior to the manipulation or procedure implementation except during transient situations where information and communications "overload" may distract the operator from performing the proper actions necessary to ensure the health and safety of the general public and plant personnel. [BNP—Action Item 95-02283]
      3) Plant announcements are made for starts (or closing breakers) of large motors, turbines, and breaker operation (>4kV). The announcement should include direction to stand clear of the applicable pump and switchgear as applicable.

67. G2.1.40 001

Given the following plant conditions:

-The plant is in MODE 6 for refueling operations

-Core alterations are in progress

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

IAW LCO 3.9.2, NUCLEAR INSTRUMENTATION, ___(1)___ source range neutron flux monitor(s) shall be OPERABLE. NI-51 and NI-52, ___(2)___ capable of providing audible indication inside CV.

A. (1) one
(2) are

B. (1) two
(2) are

C. (1) one
(2) are **NOT**

D.✓ (1) two
(2) are **NOT**

The correct answer is D

A) Incorrect. One is incorrect. Two SR NI's shall be operable per TS. If one is out, TS says to verify one Post Accident Monitor(PAM) source range neutron flux monitor is providing indication in the Control Room. Plausible if they think that they do not need two since they have the PAM source range neutron flux monitor is providing indication in the Control Room. NI-51 and NI-52(PAM Monitors) are capable of providing audible indication inside the CV is incorrect. Plausible since if one Source Range NI is inoperable, the LCO has you verify one of the PAM monitors is providing inidcation in the control room. This is visual indication only. Only NI-31 or NI-32 can provide the audible indication inside the CV.

B) Incorrect. Two SR NI's shall be operable is correct. NI-51 and NI-52(PAM Monitors) are capable of providing audible indication inside the CV is incorrect. Plausible since if one Source Range NI is inoperable, the LCO has you verify one of the PAM monitors is providing inidcation in the control room. This is visual indication only. Only NI-31 or NI-32 can provide the audible indication inside the CV.

C) Incorrect. One is incorrect. Two SR NI's shall be operable per TS. If one is out, TS says to verify one Post Accident Monitor(PAM) source range neutron flux monitor is providing indication in the Control Room. Plausible if they think that they do not need two since they have the PAM source range neutron flux monitor is providing indication in the Control Room. NI-51 and NI-52(PAM Monitors) are not capable of providing audible indication inside the CV is correct. They provide visual indication inside the Control Room only.

D) Correct. Two SR NI's shall be operable is correct. NI-51 and NI-52(PAM Monitors) are not capable of providing audible indication inside the CV is correct. They provide visual indication inside the Control Room only.

Question: 67
Tier/Group: 3
K/A Importance Rating: RO 2.8 SRO 3.9
K/A: G2.1.40: Knowledge of refueling administrative requirements.
References: GP-10, LCO 3.9.2
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3 of GP-10 DBIG
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 43.5 / 45.13)

Meets the K/A because the question asks their knowledge of Refueling administrative requirements(number of NI's that need to be operable for refueling, and based off that LCO, if NI-51 or NI-52 can be used to provide audible indication inside CV).

SHIFTLY CHECKS

Correct

NOTE:	These checks are completed prior to unlatching CRDMs **AND** shiftly until the Reactor Vessel head is installed except as defined below. This attachment is **NOT** required to be performed **OR** remain in effect, while the Reactor is defueled.

VERIFY this revision is the latest revision available. ________ ____
Date

NOTE:	Steps and substeps in this attachment may be performed in any order.

CV Checks 07-19 19-07

1. **VERIFY** Containment Closure is being tracked by OMM-033, Implementation of CV Closure. ____ ____

NOTE:	During periods of core alteration, one channel shall have an audible count rate indication available in the containment. ITS 3.9.2 Bases (NCR 299128)

2. **VERIFY** the following:

   only These 2 NIS can Provide audible indication in CV

   – Either Source Range channel N-31 **OR** N-32 providing audible indication inside CV ____ ____

   – CV evacuation alarm tested **AND** audible in all areas ____ ____

   – R-2, CV AREA, is OPERABLE ____ ____

   – R-9, LETDOWN LINE AREA, is OPERABLE ____ ____

3. **IF** a CV Purge is in progress, **THEN VERIFY** the Refueling Surface Ducts set-up for REFUEL on the PURGE OR REFUEL DAMPERS selector switch. ____ ____

4. **IF** moving recently irradiated fuel, **THEN PERFORM** Attachment 10.8. ____ ____

5. **VERIFY** the RCDT Pumps in AUTOMATIC. ____ ____

Correct

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required source range neutron flux monitor inoperable	A.1 Verify one Post Accident Monitor (PAM) source range neutron flux monitor provides indication in the Control Room.	15 minutes
	<u>AND</u>	
	A.2 Log indicated PAM source range neutron monitor count rate.	30 minutes <u>AND</u> Once per 30 minutes thereafter

← Provides indication in Control Room only

(continued)

68. G2.2.12 001

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

SI Accumulator Pressure is checked during ___(1)___. A deviation of greater than ___(2)___ of full scale during a channel check is used as the OOS limit.

A. (1) OST-020, SHIFTLY SURVEILLANCES
   (2) 7.5%

B. (1) OST-020, SHIFTLY SURVEILLANCES
   (2) 5%

C. (1) OST-021, DAILY SURVEILLANCES
   (2) 5%

D. (1) OST-021, DAILY SURVEILLANCES
   (2) 7.5%

The correct answer is B

A) Incorrect. OST-020 is the correct surveillance. 20% is incorrect. Plausible because the range of TS acceptable values is 7.5% of scale. The TS says that pressure has to be between 600-660 psig. The SI Accumulator Pressure Gauge reads 0-800 psig. 60 psig(the acceptable range per TS) is 7.5% of the 0-800 psig gauge.

B) Correct. OST-020 is the correct surveillance. A deviation greater than 5% of full scale is used as the OOS limit. The instrument shall be declared OOS.

C) Incorrect. OST-021 is incorrect. Plausible since this is the other procedure we use as part of log taking. A deviation greater than 5% of full scale is used as the OOS limit. The instrument shall be declared OOS.

D) Incorrect. OST-021 is incorrect. Plausible since this is the other procedure we use as part of log taking. 20% is incorrect. Plausible because the range of TS acceptable values is 7.5% of scale. The TS says that pressure has to be between 600-660 psig. The SI Accumulator Pressure Gauge reads 0-800 psig. 60 psig(the acceptable range per TS) is 7.5% of the 0-800 psig gauge.

Question: 68
Tier/Group: 3
K/A Importance Rating: RO 3.1 SRO 3.1
K/A: G2.2.12: Knowledge of surveillance procedures.
References: OST-020
Proposed references to be provided to applicants during the Exam: None
Learning Obj: This is a task, RO:01110106601, SRO: 02110113001
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 45.13)
Meets the K/A because the students must have knowledge of OST-20(surveillance procedure) to answer the question.

Correct

5.4 The following guidance does **NOT** apply to Steam Flow instruments when turbine load is less than 20% **AND** only applies to RTGB instrumentation. It is **NOT** intended to apply to instrumentation of different types sensing the same parameter (i.e., direct reading local pressure gauge shall **NOT** be compared against a pressure transmitter driving a remote meter): (CR 95-02700)

5.4.1 A deviation greater than 5% of full scale is used as the OOS limit. The instrument shall be declared OOS.

5.5 Steam Flow loops are pressure compensated to account for density changes in the steam that occur from low-load to full-load operation. The calibration of the loops are set up for optimum performance at 100% of rated flow. As a result, the performance of the loops at low flows is not as accurate, and can be operable but may not meet the 5% tolerance. Based on this information, the following should be used for channel check of the redundant Steam Flow instruments: (CR 95-02700)

5.5.1 **IF** Turbine load is less than or equal to 20%, **THEN**

- Steam Flow should be greater than 90% of indicated Feed Flow.

AND

- The tolerance between redundant Steam Flows should be within $0.4x10^6$pph. (ESR 97-00601)

5.5.2 **IF** Turbine load is greater than 20%, **THEN** the tolerance between redundant Steam Flow instruments should be within 5%.

5.5.3 **If** it appears that the 5% tolerance will not be met prior to exceeding 20% Turbine load, **THEN** Engineering should be contacted to evaluate the specific situation.

5.6 A channel check can be performed when a redundant channel is unavailable (OOS). The qualitative check includes observation of the channel for unexplained changes that are not attributable to known activities such as changes in power level, filling, draining, heatup, cooldown, etc. Erratic behavior or unexplained changes in the channel indication would be conditions that would warrant declaring the channel OOS and applying the ITS/TRM/ODCM required actions.

Correct

8.2.3 (Continued)

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7. Tavg
IF Tavg is outside the indicating range due to normal plant conditions, **THEN** mark N/A.
(ITS SR 3.3.2.1, Table 3.3.2-1 Item 1.f, 4.d and 6.b) ____ ____

INSTRUMENT	INSTRUMENT RANGE	MAXIMUM DEVIATION	CALCULATION NUMBER
TI-412D TI-422D TI-432D	540-615• F	4• F	WCAP 11889

8. SI Accumulator Level
(TRMS TR 4.2.1) ____ ____

INSTRUMENT	INSTRUMENT RANGE	MAXIMUM DEVIATION	CALCULATION NUMBER
LI-920 LI-922	0-100%	5%	RNP-I/INST-1052
LI-924 LI-926	0-100%	5%	RNP-I/INST-1052
LI-928 LI-930	0-100%	5%	RNP-I/INST-1052

9. SI Accumulator Pressure
(TRMS TR 4.2.1) ____ ____

INSTRUMENT	INSTRUMENT RANGE	MAXIMUM DEVIATION	CALCULATION NUMBER
PI-921 PI-923	0-800 psig	40 psig	RNP-I/INST-1036
PI-925 PI-927	0-800 psig	40 psig	RNP-I/INST-1036
PI-929 PI-931	0-800 psig	40 psig	RNP-I/INST-1036

ACTIONS (continued) Incorrect

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met.	D.1 Be in MODE 3. AND D.2 Reduce pressurizer pressure to ≤ 1000 psig.	6 hours 12 hours
E. Two or more accumulators inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	Once prior to removing power from the valve operator
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 825 ft^3 and ≤ 841 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 600 psig and ≤ 660 psig.	12 hours

Showing The acceptable range Per T.S. which is 7.5% of Full Scale

(continued)

Incorrect

8.4.1.1 (Continued)

				INIT 07-19	INIT 19-07
b.	**VERIFY** Nitrogen Cover pressure in each SI Accumulator is greater than or equal to 600 psig and less than or equal to 660 psig. (ITS SR 3.5.1.3)				
	– SI Accumulator "A"	(07-19) ______	psig	____	
	– SI Accumulator "B"	(07-19) ______	psig	____	
	– SI Accumulator "C"	(07-19) ______	psig	____	
	– SI Accumulator "A"	(19-07) ______	psig		____
	– SI Accumulator "B"	(19-07) ______	psig		____
	– SI Accumulator "C"	(19-07) ______	psig		____

69. G 2.2.7 001

Which ONE (1) of the following people can be selected to be an Infrequently Performed Test or Evolution (IPTE) Manager IAW OPS-NGGC-1315, CONDUCT OF INFREQUENTLY PERFORMED TESTS OR EVOLUTIONS?

A. Lead Test Performer

B. A Control Room Supervisor

C. The on-duty Shift Manager

D✓ The Manager, Shift Operations

The correct answer is D.

A. Incorrect. OPS-NGGC-1315 calls out the IPTE Manager as an individual fulfilling a role senior to the Shift Manager. Plausible because one of the people involved with the IPTE is the Lead Test Performer.

B. Incorrect. OPS-NGGC-1315 calls out the IPTE Manager as an individual fulfilling a role senior to the Shift Manager. Plausible because one of the primary responsibilities for IPTE Manager and for a CRS is oversight.

C. Incorrect. OPS-NGGC-1315 specifically prohibits the on-duty Shift Manager from performing duties as the IPTE Manager.

D. Correct. OPS-NGGC-1315 calls out the IPTE Manager as an individual fulfilling a role senior to the Shift Manager. Manager, Shift Operations position falls between Manager, Operations and Shift Manager. This position is an SRO licensed position so they could not fulfill this role while standing proficiency watches.

Question: 69
Tier/Group: 3 (RO)
K/A Importance Rating: RO 2.9 SRO 3.6

K/A: 2.2 Equipment Control
2.2.7 Knowledge of the process for conducting special or infrequent tests.

Reference(s): OPS-NGGC-1315
Proposed References to be provided to applicants during examination: None
Learning Objective: Self study procedure
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR Part 55 Content: 41.10
Comments:

This question meets the K/A because the candidate must know requirements for conducting infrequent tests.

Correct

3.0 DEFINITIONS

1. **Infrequently Performed Tests or Evolutions (IPTE)**: An activity that has the potential to significantly degrade the plant's level of nuclear safety and warrant additional management oversight and controls including: **[R1 Rec. #1a]**
   - Evolutions not specifically covered by existing normal or abnormal operating procedures.
   - Evolutions that are seldom performed even though covered by existing normal or abnormal procedures. (e.g. Unit Startup, Activities involving potential significant impact to the Reactor Core.)
   - Special infrequently performed surveillance testing that involves complicated sequencing or placing the plant in unusual configurations. (e.g. Integrated ECCS testing, ECCS check valve testing)
   - Evolutions that require the use of special test procedures in conjunction with existing procedures.
   - Enclosure 1, IPTE Examples provides specific examples
2. **IPTE Manager:** A member of the plant staff or other individual designated by the Plant General Manager, to provide oversight of a specific evolution. Specific required attributes include: **[R1, Rec. #1e], [R5, Rec. #1c]**
   - A senior level member of management, i.e., in a role senior to the Shift Manager The MSO, Manager Shift Operations is senior To shift Manager
   - Is **NOT** involved in performance of the assigned evolution. Specifically, this individual does not replace any individual involved in the test or evolution nor supervises the evolution.
   - Shall possess the requisite knowledge, skills and experience to provide meaningful oversight of the evolution
   - Shall **NOT** be the on-duty Shift Manager
   - Shall provide continuous oversight, i.e., from beginning to end, of the evolution.
3. **Lead Test Performer:** An individual assigned by the Section Manager responsible for the evolution to supervise the entire test/evolution. This could be anyone with an in-depth knowledge of the test/evolution. Lead Test Performer is meant to be a generic title; it could be the Test Director, SVI Coordinator, etc. The Lead Test Performer shall **NOT** be considered an "oversight" role. **[R1 Rec. #1]**

3.0 DEFINITIONS

incorrect

1. **Infrequently Performed Tests or Evolutions (IPTE)**: An activity that has the potential to significantly degrade the plant's level of nuclear safety and warrant additional management oversight and controls including: **[R1 Rec. #1a]**
   - Evolutions not specifically covered by existing normal or abnormal operating procedures.
   - Evolutions that are seldom performed even though covered by existing normal or abnormal procedures. (e.g. Unit Startup, Activities involving potential significant impact to the Reactor Core.)
   - Special infrequently performed surveillance testing that involves complicated sequencing or placing the plant in unusual configurations. (e.g. Integrated ECCS testing, ECCS check valve testing)
   - Evolutions that require the use of special test procedures in conjunction with existing procedures.
   - Enclosure 1, IPTE Examples provides specific examples
2. **IPTE Manager:** A member of the plant staff or other individual designated by the Plant General Manager, to provide oversight of a specific evolution. Specific required attributes include: **[R1, Rec. #1e], [R5, Rec. #1c]**
   - A senior level member of management, i.e., in a role senior to the Shift Manager Can't Be CRS, CRS is Junior To Shift manager
   - Is **NOT** involved in performance of the assigned evolution. Specifically, this individual does not replace any individual involved in the test or evolution nor supervises the evolution.
   - Shall possess the requisite knowledge, skills and experience to provide meaningful oversight of the evolution
   - Shall **NOT** be the on-duty Shift Manager
   - Shall provide continuous oversight, i.e., from beginning to end, of the evolution.
3. **Lead Test Performer:** An individual assigned by the Section Manager responsible for the evolution to supervise the entire test/evolution. This could be anyone with an in-depth knowledge of the test/evolution. Lead Test Performer is meant to be a generic title; it could be the Test Director, SVI Coordinator, etc. The Lead Test Performer shall **NOT** be considered an "oversight" role. **[R1 Rec. #1]**

70. G2.3.14 001

Inside Work Control, the following conditions were found:

-Removable surface contamination is 1500 dpm/cm^2

-Radiation levels are 3 mrem/hr at 30cm

Which ONE (1) of the following identifies the minimum postings for the Work Control area?

A. Radiation area only

B. Contamination area only

C. no postings necessary

D. A contamination and radiation area

The correct answer is B

A) Incorrect. An area accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates. Plausible if the student doesn't know the definition of a radiation area and since there is radiation found in an area where normally there is none.

B) Correct. The only posting necessary is posting as a contamination area. The area exceeds the 1000 dpm/cm^2 which is the limit for a contamination area. The radiation levels do not meet the requirements for being posted as a radiation area.

C) Incorrect. Plausible if the student does not know what the requirements are for radiological postings.

D) Incorrect. The only posting necessary is posting as a contamination area. The area exceeds the 1000 dpm/cm^2 which is the limit for a contamination area. The radiation levels do not meet the requirements for being posted as a radiation area.

Question: 70
Tier/Group: 3
K/A Importance Rating: RO 3.1 SRO 3.1
K/A: G2.3.14: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
References: HPS-NGGC-0003
Proposed references to be provided to applicants during the Exam: None
Learning Obj:
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.12 / 43.4 / 45.10)

Meets the K/A because the student must have knowledge of radiation and contamination hazards(Know when to post an area) during normal activities.

3.4 Airborne Radioactivity Area (ARA)

A room, enclosure, or area in which airborne radioactive materials, composed wholly or partly of licensed material, exist in concentrations:

In excess of the Derived Air Concentrations (DACs) specified in Appendix B of 10CFR20, or

To such a degree that an individual present in the area without respiratory protective equipment could exceed, during the hours an individual is present in a week, an intake of 0.6% of the Annual Limit on Intake (ALI) or 12 DAC-hours.

3.5 Beta-Gamma DAC-Fraction Action Level

Beta-gamma airborne radioactivity above which an air sample should be counted for alpha airborne radioactivity.

3.6 Barrier/Barricade

A physical structure (for example, rope, wall, door, swing gate) that checks the advance of personnel, such that the structure would need to be removed, opened or climbed over or under to gain access.

3.7 Contamination Area (CA)

Any area accessible to personnel where the removable surface contamination is in excess of 1,000 dpm/100 cm^2 beta-gamma and/or 20 dpm/100 cm^2 alpha.

3.8 Direct Surveillance

Under the positive control of an individual who has direct line of sight to the entrance of the area to challenge individuals prior to entering.

3.9 High Radiation Area (HRA)

An area accessible to individuals, in which radiation levels from radiation sources external to the body, could result in an individual receiving a dose equivalent in excess of 100 mrem in one hour at 30 cm from the radiation source or 30 cm from any surface that the radiation penetrates.

3.10 Hot Spot (HS)

A local intense source of radiation with a contact dose rate greater than 100 mrem per hour and greater than five times the dose rate at 30 cm.

3.11 Label

A sticker or tag which is attached to a container of radioactive material for the purpose of providing radiological information necessary to inform the worker of radiation hazards associated with the container. A label can also be pre-printed with caution wording and a radiation symbol supplemented by handwritten hazard information. (See also RAM tag definition)

3.12 Licensed Material

Source material, special nuclear material, or byproduct material received, possessed, used, transferred or disposed of under a general or specific license.

3.13 Locked High Radiation Area (LHRA)

Areas with dose rates greater than 1,000 mrem per hour at 30 centimeters from the radiation source or from any surface penetrated by the radiation, but less than 500 Rads per hour at 1 meter from the radiation source or from any surface penetrated by the radiation. (For CR-3 only, greater than or equal to 1000 mrem/hr at 30 cm applies).

3.14 Rad

A unit of absorbed dose equal to 100 ergs/g in any medium.

3.15 Radioactive Material that Requires Labeling

Material in which the amount of radioactivity exceeds the quantities specified in 10CFR20, Appendix C.

3.16 Radioactive Materials Area (RMA)

Any room or area where radioactive material is used or stored in amounts exceeding 10 times the amount of such material as specified in 10CFR20, Appendix C.

3.17 Radiation Area (RA), includes Neutron Radiation Area

An area accessible to individuals, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 mrem in one hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

3.18 Radiation Control Area (RCA)

Any area within a restricted area that is posted for radiological protection purposes.

3.19 Radioactive Material (RAM) Tag

A radioactive material tag which can be used to meet the labeling requirements of 10CFR20.1904.

3.20 "Radiation Survey Required Prior to Entry" or "Controlled Radiation Protection" Area

Any area within an RCA that is further designated, controlled, and posted for a specific radiological protection purpose.

3.21 Rem

The special unit of any of the quantities expressed as dose equivalent. The dose equivalent in rems is equal to the absorbed dose in rads multiplied by the quality factor (Q).

3.22 Restricted Area

An area, access to which is limited by a physical barrier such as a wall, fence, or by continuous surveillance and control of access by a representative of the company, for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

3.23 Very High Radiation Area (VHRA)

An area accessible to individuals, in which radiation levels from radiation sources external to the body, could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at one meter from a radiation source or one meter from any surface that the radiation penetrates.

71. G2.3.15 001

Which ONE (1) of the following radiation monitors is located in the Auxiliary Building hallway above the Station Air receiver?

A. R-18 (Waste Disposal System liquid effluent)

B. R-17 (Component Cooling Water)

C✓ R-16 (Containment Fan Coolers)

D. R-9 (Letdown line area)

The correct answer is C

A) Incorrect. Plausible because R-18 is located in the AUX BLDG.

B) Incorrect. Plausible because R-17 is located in the AUX BLDG.

C) Correct. R-16 is in the AUX BLDG hallway above the Station Air Receiver.

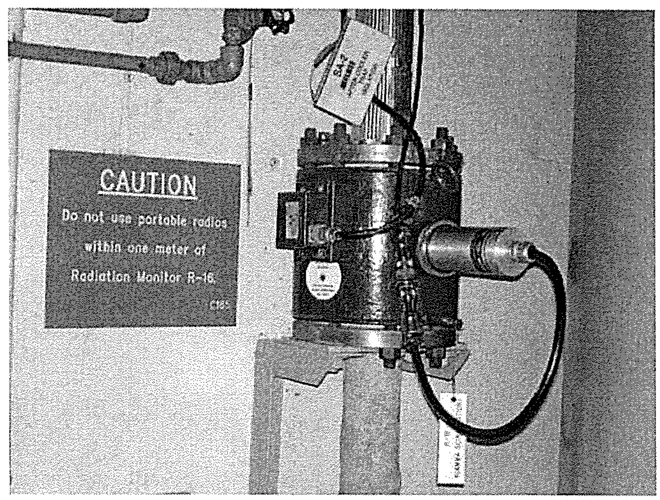
D) Incorrect. Plausible because R-9 is located in the AUX BLDG.

Question: 71
Tier/Group: 3
K/A Importance Rating: RO 2.9 SRO 3.1
K/A: G2.3.15: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
References: Student Text for Radiation Monitoring System
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 3 of RMS Lesson Plan
Question Source: RNP Bank
Question History: None
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.12 / 43.4 / 45.9)

Meets the K/A because the student must have knowledge of the Radiation Monitoring System(their locations) to answer this question.

NRC Process Radiation Monitors
R-16; CV HVH Cooling Water Liquid

1. This channel monitors the CV HVH Units fan and motor cooling water discharge for radiation indicative of a leak from the Containment Vessel atmosphere into Service Water that could occur during a LOCA.



Aux Building Hallway

72. G2.3.4 001

Given the following conditions:

-An RNP Mechanic, previously employed by VC Summer, has been assigned to repack a valve

-The mechanic's current yearly dose from VC Summer was 1 Rem TEDE

-The mechanic has received no dose from RNP

-Projected dose rate in the area is 500 mR/hr

What is the **MAXIMUM** time that the mechanic can work on the valve before reaching the Annual Administrative Dose Limit IAW DOS-NGGC-0004, ADMINISTRATIVE DOSE LIMITS?

A. 1 hour

B. 2 hours

C.✓ 4 hours

D. 8 hours

The correct answer is C

A) Incorrect. Plausible because the administrative limit is 0.5 rem if non RNP employee dose for the current year has not been determined.

B) Incorrect. Plausible because the administrative limit is 2 rem not to exceed 4 rem total dose if non RNP dose for the current year has been determined. The student may incorrectly apply this and think the mechanic can only get 2 rem total.

C) Correct. The mechanic can get 2 rem RNP dose, but they can't exceed 4 rem total dose. Since this mechanic already has 1 rem, the most he can get at RNP is 2 Rem.

D) Incorrect. Plausible because the federal limit is 5 rem total dose. The question asks specifically for the RNP Administrative Dose Limit.

Question: 72
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: DOS-NGGC-0004
Proposed references to be provided to applicants during the Exam: None
Learning Obj:
Question Source: New
Question History:
Question Cognitive Level: High
10 CFR part 55: (CFR 41.12 / 43.4 / 45.10)

Meets the K/A because the student must know radiation exposure limits under normal conditions.

5.0 PREREQUISITES

N/A

6.0 PRECAUTIONS AND LIMITATIONS

N/A

7.0 SPECIAL TOOLS AND EQUIPMENT

N/A

8.0 ACCEPTANCE CRITERIA

N/A

9.0 INSTRUCTIONS

R2.1 9.1 Adult Occupational Dose Limits

9.1.1 Whole Body - The more limiting of a total effective dose equivalent equal to 5 rem or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue other than the lens of the eye equal to 50 rem.

9.1.2 Skin - A shallow dose equivalent equal to 50 rem.

9.1.3 Lens of Eye - A lens dose equivalent equal to 15 rem.

9.1.4 Extremities - A shallow dose equivalent equal to 50 rem.

9.2 Occupational Dose to Minors

Minors shall not be employed to work in radiation control areas, although they may enter as visitors.

9.3 Progress Energy Annual Administrative Dose Limits

9.3.1 0.5 rem Progress Energy dose if non-Progress Energy dose for the current year has not been determined. No dose extension is permitted.

9.3.2 2 rem Progress Energy dose not to exceed 4 rem total dose if non-Progress Energy dose for the current year has been determined.

73. G2.4.12 001

The RO has determined that for the safety of the reactor and the plant, a reactor trip and initiation of Safety Injection(SI) is necessary. He has announced it to the CRS, however, it has been several seconds and the CRS has not responded.

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

The RO should __(1)__ then __(2)__ IAW OPS-NGGC-1000, FLEET CONDUCT OF OPERATIONS.

A. (1) wait for confirmation from the CRS
   (2) trip the reactor and initiate SI

B. (1) wait for confirmation from the CRS
   (2) initiate SI

C.✓ (1) trip the reactor, verify it is tripped
   (2) initiate SI

D. (1) trip the reactor and initiate SI
   (2) perform immediate action steps

The correct answer is C

A) Incorrect. The RO should announce it if time permits and get confirmation from the CRS if time permits. Plausible because normally the RO would wait for the CRS to confirm his actions, however, this is an emergency situation and as a licensed operator, the RO has the ability to trip the reactor and initiate ESF anytime they believe it is necessary. The second part is incorrect. Plausible because you would trip the reactor and initiate SI however, you would verify the reactor is tripped prior to initiating SI.

B) Incorrect. The RO should announce it if time permits and get confirmation from the CRS if time permits. Plausible because normally the RO would wait for the CRS to confirm his actions, however, this is an emergency situation and as a licensed operator, the RO has the ability to trip the reactor and initiate ESF anytime they believe it is necessary. The second part is incorrect. Plausible because you do need to initiate SI and if you do initiate SI, this will give you a reactor trip.

C) Correct. The RO will trip the reactor then verify it is tripped. He does not need to wait on confirmation from the CRS. After he has verified the reactor tripped, he will initiate SI.

D) Incorrect. Plausible because you will trip the reactor and initiate SI. However, you will trip the reactor, verify it is tripped, then initiate SI. The performance of immediate action steps is plausible because you do perform your immediate action steps for a reactor trip and SI, however, you must first verify the reactor is tripped prior to initiating SI per OMM-022.

Question: 73
Tier/Group: 3
K/A Importance Rating: RO 4.0 SRO 4.3
K/A: G2.4.12: Knowledge of general operating crew responsibilities during emergency operations.
References: OPS-NGGC-1000, OPS-NGGC-1306
Proposed references to be provided to applicants during the Exam: None
Learning Obj: Objective 4 of OPS-NGGC-1306, Objective 2 of OPS-NGGC-1000
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 45.12)

Meets the K/A because the question asks the responsibilities of the RO when there is a need to trip the reactor.(emergency operations)

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Steps 1 through 4 are IMMEDIATE ACTION steps.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
1.	**Check Reactor Trip:** • Reactor trip and bypass breakers - OPEN • Rod position indicators - AT ZERO • Rod Bottom lights - ILLUMINATED • Neutron flux - LOWERING	Manually trip reactor. IF reactor power is greater than or equal to 5% OR intermediate range SUR is positive, THEN Go To FRP-S.1, Response To Nuclear Power Generation/ATWS, Step 1.
		Verify 1st Prior To initiating Safety Injection
2.	**Check Turbine Trip:**	
	a. Both turbine stop valves - CLOSED	a. Manually trip turbine. IF turbine will NOT trip, THEN manually run back turbine at maximum rate until all governor valves are closed. IF turbine can NOT be run back, THEN manually close MSIVs and MSIV bypass valves.
	b. Close MSR purge and shutoff valves	b. IF loss of power prevents MSR isolation, THEN manually close MSIV and MSIV bypass valves.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
4.	**Check SI Status:**	
	a. Check if SI is actuated: • SI annunciators - ANY ILLUMINATED <u>OR</u> • SI equipment - AUTO STARTED	a. Check if SI is required: • PZR pressure less than 1715 psig <u>OR</u> • Containment pressure greater than 4 psig <u>OR</u> • Steam line ΔP bistables illuminated <u>OR</u> • High steam flow with low Tavg or low steam pressure bistables illuminated <u>IF</u> SI is required, <u>THEN</u> manually actuate BOTH trains of SI. <u>IF</u> SI is <u>NOT</u> required, <u>THEN</u> perform the following: 1) Reset SPDS and initiate monitoring of Critical Safety Functions Status Trees. 2) Go To EOP-ES-0.1, Reactor Trip Response, Step 1.
	b. Check BOTH trains of SI - ACTUATED • SI pumps - BOTH RUNNING • RHR pumps - BOTH RUNNING	b. Manually actuate BOTH trains of SI.

4.5.13 Shift Technical Advisor (STA) (continued)

j. Provides additional plant technical information to the operating crew as requested to aid in plant event mitigation.

k. Provides concurrence with key diagnostic and transition decisions through independent assessment of plant conditions.

l. Reviews and concurs with Emergency Action Levels and assists in the implementation of the emergency plan as directed by the SM/CRS and applicable procedures.

m. Anticipate plant expected response and make recommendations to the CRS and SM as applicable on upcoming changes in mitigation strategy.

4.6 **Reactor Operator (RO)**

1. During transients, significant evolutions, or abnormal conditions, one Reactor Operator will place primary focus on reactor control, while another Reactor Operator places primary focus on the Balance of Plant. [BNP—Action Item 95-02283]

2. To ensure public, plant and personnel safety, Reactor Operators shall:

   a. Shutdown the reactor when it is determined the safety of the reactor is in jeopardy or when exceedance of RPS setpoints is imminent.

   b. The ROs believe and respond conservatively to instrument indications, and use multiple indications to verify them to be incorrect.

   c. Ensure adherence to the plant technical specifications and licensing basis.

   d. Ensure compliance and operate systems per the Plant Operating Manual.

   e. Initiate the immediate actions necessary to maintain the plant in a safe condition during abnormal and emergency operations.

3. Additionally, the ROs may actuate Emergency Safeguards Features whenever the corresponding setpoints and logic are met or will be met. Due to the significance of these actions, STAR shall be used. When manually actuating ESF, RPS, or inserting a manual scram/trip, the ROs shall inform the CRS about the action. If time permits, this notification should be made prior to taking the action. This communication should not delay action when safety of the reactor or plant is in jeopardy.

4.4.2 All Licensed Operators

1. Remain cognizant during the approach to criticality on reactor startups for any anomalous indications or inadvertent criticalities.
2. Have the responsibility and authority to trip the unit if there is uncertainty as to the unit's status with respect to the control of reactivity and control of the plant. **[R13, Rec. 1]**
3. Place emphasis during turnover and control board walk downs on items important to reactivity management.
4. Have the authority to terminate any activity in which the effects on reactivity control are unknown or non-conservative. **[R13, Rec. 1]**
5. Ensure all control rod movements are made in a deliberate, carefully controlled manner while constantly monitoring nuclear instrumentation and redundant indications of reactor power and neutron flux. **[R14, Rec. 2a]**
6. Are sensitive to the reactivity effects that may result from normal and infrequent evolutions.
7. Ensures the details of events or equipment problems related to the control of reactivity are recorded and initiates corrective actions
8. Maintain a cautious approach to the adjustment or interpretation of power indication by questioning the reasons behind discrepancies that may exist between power measurements. **[R23, Rec. 3]**
9. Take a conservative approach in the performance of normal and infrequent evolutions, questioning personnel concerning their requests to perform evolutions until satisfied that reactivity control is not compromised. **[R13, Rec. 1] [R23, Rec. 1]**
10. Shall not rely solely on one indication when making operational decisions related to reactivity management, but shall seek alternate verification when available. **[R23, Rec. 1]**
11. Ensure that ECC/ECP is approached cautiously. The policy shall be that criticality could be achieved at any time rather than an attitude of over reliance in an ECC/ECP.
12. Do not rely on reactor trip interlocks to trip the reactor. When a reactor trip is warranted, manually trip the reactor to avoid challenging the Reactor Protection System. **[R29, Rec. 1]**
13. Maintain an awareness of reactivity parameters, operating margins, and associated limits.
14. Maintain an awareness of planned reactivity changes and contingency actions for unanticipated changes to core reactivity or reactor power through crew turnover, shift briefs, reactivity plans, and schedule reviews
15. Identify any discrepancies between actual plant response and predicted plant response following any power changes, to ensure appropriate investigations are performed. The discrepancies need to be understood and resolved in a timely manner. **[R23, Rec. 3]**

5.

74. G2.4.26 002

Given the following plant conditions:

-A **SMALL** fire is reported near the "A" and "B" Auxiliary boilers

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

IAW OMM-003, FIRE PROTECTION PRE-PLANS/UNIT NO. 2, the fire brigade must attack this fire with ___(1)___ and the fire brigade is required to protect the ___(2)___.

A. (1) a portable fire extinguisher or hose stream
(2) Primary Air Compressor

B✓ (1) a portable fire extinguisher or hose stream
(2) Steam Driven AFW pump

C. (1) hose stream only
(2) Primary Air Compressor

D. (1) hose stream only
(2) Steam Driven AFW pump

The correct answer is B

A) Incorrect. A portable fire extinguisher or hose stream is correct. The primary air compressor is incorrect. Plausible because this compressor is near the auxiliary boilers.

B) Correct. OMM-003 says that they could use a portable fire extinguisher or a hose stream for this fire. The fire brigade is required to protect the SDAFW pump IAW omm-003.

C) Incorrect. Hose stream only is incorrect. Plausible since it is one of the methods that can be used, however, it's not the only method. The primary air compressor is incorrect. Plausible because this compressor is near the auxiliary boilers.

D) Incorrect. Hose stream only is incorrect. Plausible since it is one of the methods that can be used, however, it's not the only method. The fire brigade is required to protect the SDAFW pump IAW omm-003.

Question: 74
Tier/Group: 3
K/A Importance Rating: RO 3.1 SRO 3.6
K/A: G2.4.6: Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage.
References: OMM-003
Proposed references to be provided to applicants during the Exam: None
Learning Obj: None, this is a self study procedure
Question Source: New
Question History:
Question Cognitive Level: Low
10 CFR part 55: (CFR 41.10 / 43.5)

Meets the K/A because the candidate must know if they can use a portable fire extinguisher or a hose stream or a hose stream only IAW OMM-003. The candidate must also know what equipment must be protected.

Correct

8.45 A&B Aux. Boilers and Associated Fuel Oil Pumps

8.45.1 Fire Hazards

Three fuel oil pumps west of "A" and "B" Auxiliary Boilers and fuel oil in associated piping, approximately 18 gallons total.

8.45.2 Fixed Fire Suppression Systems

None Applicable

8.45.3 Guidelines for Fire Attack

1. Access for Fire Attack

   a. Direction of fire attack should be based on wind direction. Start the attack from the upwind side with the wind on your back.

   b. "A" and "B" Boilers are located in an open area on the ground floor of the Turbine Building. Initial attack should to be from the south **OR** west of the equipment.

2. Method of Attack

   a. Small fire - portable fire extinguishers may be used.

   b. Large or fast spreading fires - hose stream or foam may be used at the discretion of the FBIC.

3. Portable Fire Suppression Equipment Available

   a. Fire extinguishers located near "A" and "B" Auxiliary Boilers foam, dry chemical, and 150# wheeled Halon 1211

   b. Various hose stations on ground floor of Turbine Building

   c. Fire Hydrant and hose house located near the Secondary Sample Lab

   d. Foam fire equipment

Correct

8.45.3 (Continued)

4. Exposure Protection

   a. Fuel oil pumps and fuel oil lines in this area.

   b. Protect Steam Driven Auxiliary Feed Water Pump (SDAFW Pump) from heat and exposure fire.

   c. Check for fire extension on upper levels of the Turbine Building. The CV Access areas and RCA Tool Room may be exposed to radiant heat. Consider personnel safety and possible combustible material ignition.

5. Ventilation

   a. All smoke will vent itself into the surrounding atmosphere.

6. Potential Radiological/Toxic Hazards

   a. Smoke and products of combustion will reduce visibility and produce a toxic atmosphere.

   b. Lube oil or chemical runoff would be directed to the Storm Drains and require monitoring of Settling Pond releases.

   c. As a reminder, the Fire Brigade is not qualified to deal with hazardous materials disposal and should only attempt to control the runoff. E&RC should be contacted if this situation arises.

7. Plant Systems Affecting Fire Fighting Efforts

   a. De-energize affected pumps if pipe leak has occurred to reduce spill.

   b. If a break occurs on the supply side of the pumps, isolate fuel line upstream of break. Fuel oil is gravity fed from IC Turbine tanks on Unit 1.

   c. The Auxiliary Boilers and their support equipment receive electrical power from MCC-3 and MCC-4.

   d. Refer to EDP-003 through 009 for additional guidance on electrical distribution in this area.

75. G2.4.39 001

The Plant is at 100% RTP when the following events occur:

2115 - Power is lost to all Control Room annunciators

2120 - An Unusual Event is declared

2125 - The Emergency Notification form is complete and the SM directs you to make the state and county notifications

2130 - A reactor trip occurs and the SM declares an Alert

2135 - The SM completes a new Emergency Notification form and directs you to make the state and county notifications

Which ONE (1) of the following indicates the latest time the state and counties must **FIRST** be notified?

A. 2130

B.✓ 2135

C. 2140

D. 2145

The correct answer is B.

A. Incorrect. See explanation B. Plausible because 2130 is 15 minutes from the time the event actually happened, not from the time the UE was declared.

B. Correct. Per EPNOT-1 the state and counties must be notified within 15 minutes. EPNOT-1 also states that IF a higher emergency classification is declared prior to completing an in-progress notification, THEN complete the notification of the lower event before starting the notification for the higher classification. Per conditions given, 2135 is the correct time.

C. Incorrect. See explanation B. Plausible because 2140 is 15 minutes from the time the notification first form was completed.

D. Incorrect. See explanation B. Plausible because 2145 is 15 minutes from the time the second declaration.

Question: 75
Tier/Group: 3 (RO)
K/A Importance Rating: RO 3.9 SRO 3.8

K/A: 2.4 Emergency Procedures/Plan
2..4.39 Knowledge of ther RO's responsibilities in emergency plan implementation.

Reference(s): EPNOT-1
Proposed References to be provided to applicants during examination: None
Learning Objective:
Question Source: RNP Bank
Question History: Not used on a previous RNP NRC Exam
Question Cognitive Level: Comprehension
10 CFR Part 55 Content: 41.10
Comments: This question meets the K/A because the question relates to a task the RO could be responsible for during E-Plan implementation.

8.2.4 Electronic Emergency Notification Form Completion

CAUTION

Initial notifications are to be made within 15 minutes. Follow up notifications shall be made within 60 minutes from the completion of the previous notification, or more frequently if warranted by changing conditions. The 60 minute clock will start when the first agency disconnects from notification call.

CAUTION

IF a higher emergency classification is declared prior to completing an in-progress notification, **THEN** complete the notification of the lower event before starting the notification for the higher classification. Both notifications must still be completed within 15 minutes of their respective declarations. If additional resources are available, assign a second communicator to start preparing the notification for the higher classification.

IF a higher emergency classification is declared while preparing an initial notification for the lower emergency classification, **THEN** one of the following two approaches may be used:

IF the notification of the higher event can be prepared, approved, and commenced within 15 minutes of the lower classification, **THEN** prepare the initial notification for the higher event.

OR

IF the notification of the higher event **cannot** be prepared, approved and commenced within 15 minutes of the lower classification, **THEN** complete the notification of the lower event before starting the notification for the higher classification. Both notifications must still be completed within 15 minutes of their respective declarations. If additional resources are available, assign a second communicator to start preparing the notification for the higher classification.

(RIS 2007-02)

1. Instructions for completing the form are included in Attachment 10.5, Nuclear Power Plant Emergency Notification Form, of this procedure.

8.2.5 Transmit State and County Notifications

1. All agencies shall be contacted for each initial and follow-up notification. Agencies that do not respond shall be contacted by any means available, as soon as possible.

2. **ESTABLISH** communications with the State and County agencies using any of the following:

NUCLEAR POWER PLANT EMERGENCY NOTIFICATION FORM
INSTRUCTIONS FOR COMPLETION

CAUTION

Initial notifications are to be made within 15 minutes of the declaration of an emergency. Follow up notifications shall be made within 60 minutes from the completion of the previous notification, or more frequently if warranted by changing conditions. The 60 minute clock will start when the first agency disconnects from notification call.

All efforts should be expended to obtain information required for the Electronic Emergency Notification Form. **IF** an upgrade in classification occurs when the follow-up message is due, **THEN** "upgraded ENF forthcoming" should be annotated in "Remarks". This information is to be promptly transmitted to the State and County agencies, as soon as it is available.

Messages should include an up-to-date description of what is happening at the plant within the constraints of timely notifications. To ensure messages contain adequate and accurate information about current plant conditions, messages should be developed as promptly as possible. ENF reviews will be conducted by EOF Facility personnel, if available. It may be necessary to determine a "cut off time" for new message information, so that these reviews can be made. The ERM will direct EOF personnel through the ENF line by line; and the POA, TAM and RCM will verify the accuracy of the ENF, using the following guidance:

Lines 1, 2, 3, 4 – POA verifies accuracy
Lines 5, 6, 7 – RCM verifies accuracy
Line 8 – TAM verifies accuracy
Line 9 – RCM verifies accuracy
Lines 10, 11, 12 – POA verifies accuracy
Line 13 – POA verifies accuracy
Lines 14, 15, 16 – RCM verifies accuracy (reference the Rascal "Source Term Summary" sheet, as applicable)
Line 17 – ERM approves

Lines 1 through 13 **AND** Line 17 **MUST BE COMPLETED** on an **INITIAL** Electronic Emergency Notification Form. Information for Line 9 may not be available for the initial notification. If met data is not available, then state this in the Remarks (Line 13) and provide the data on a follow-up notification, as soon as the data is available. For **TERMINATION** messages, only Lines 1 through 4, 10, and 17 are required.

Information included on the initial form **AND** Lines 14 through 16 **MUST BE COMPLETED** on a **FOLLOW-UP** Electronic Emergency Notification Form, unless Line 6 (EMERGENCY RELEASE) is selected.

An electronic Emergency Notification Form is available and can be accessed through the NGG Standard Desktop.