QUESTION NUMBER: 001

A reactor trip has occurred from 100% power. While performing the standard post trip actions, the BOPO observes that the main turbine has not tripped and attempted to trip the turbine by pressing both turbine trip pushbuttons.

Which one of the following additional actions is directed by EOP-00, "REACTOR TRIP," to trip the main turbine?

A. Place both EHC pump control switches in after-trip.

B.✓ Place both EHC pump control switches in pull-to-lock.

C. Open the generator output breakers.

D. Open the generator field breaker.

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 1 Rev 0

KA #: 000007 EA2.02 Tier 1 Group 1: Reactor Trip
Ability to determine or interpret the following as they apply to a reactor trip: Proper actions to be taken if the automatic safety functions have not taken place
Importance 4.3 / 4.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
LIST the sixteen (16) immediate actions of EOP-00 in the order they are required to be performed

Question Pedigree
Bank question used on 2002 NRC Exam, reworded and two choices changed. Not considered to be modified question.

K/A Fit:
Applicant is required to describe actions to be taken if the turbine does not trip as designed following a reactor trip.

Choice A:
Distractor; Plausible because both EHC control switches are repositioned, Incorrect because EOP-00 specifies that both EHC pump control switches be placed in pull-to-lock, not after trip.

Choice B:
Correct answer; EOP-00 specifies that both EHC pump control switches be placed in pull-to-lock if the turbine has not tripped.

Choice C:
Distractor; Plausible because tripping the generator output breakers should result in a turbine trip signal, but incorrect per EOP-00

Choice D:
Distractor; Plausible because loss of excitation should result in a turbine trip signal, but incorrect per EOP-00

KA#:	000007 EA2.02	Bank Ref #:	07-18-10 056
LP# / Objective:	0718-10 01.10	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	BANK
Reference:	EOP-00	Handout:	NONE

INSTRUCTIONS

2. Verify the Turbine is tripped as indicated by Stop and Intercept Valves indicating closed.

CONTINGENCY ACTIONS

2.1 Trip the Turbine by performing Step a, b or c:

a. Trip the Turbine (CB-10,11).

b. Stop the EHC Pumps by placing **BOTH** of the following control switches in "PULL TO LOCK":

- EHC-3A
- EHC-3B

c. **(LOCAL)** Trip the Turbine using 1A-3/ST-1 and 1B-3/ST-1, "TURBINE TRIP PUSHBUTTONS" (Turbine Bldg Stairway to Cable Spread Room).

QUESTIONS REPORT
for ILO EXAM BANK

07-18-10 056

A reactor trip has occurred from 100% power. While performing the standard post trip actions, the RO observes that the main turbine has not tripped.

Which one of the following is a procedurally directed action that may be performed to trip the main turbine?

A.✔ Stop the EHC pumps.

B. Deenergize excitation.

C. Reduce load limit pot to zero.

D. Open generator output breakers.

QUESTION NUMBER: 002

The following sequence of events occurred:

- A loss of load event resulted in a high pressure reactor trip
- PORVs and Pressurizer Safety Valves opened
- One of the Safety Valves failed to close when pressurizer pressure lowered below its setpoint
- PPLS actuated
- All RCPs were tripped by the operators per EOP-00, "STANDARD POST TRIP ACTIONS"

Current plant conditions are:

- Pressurizer pressure is 900 psia and steady
- Representative CET temperatures are 532°F and steady
- Reactor Vessel Level is 63%
- Pressurizer level is 90% and rising rapidly
- LPSI pumps SI-1A and SI-1B are running
- HPSI Pump SI-2A is running
- All charging pumps are running

What action should be taken in response to these plant conditions?

A. Stop all but one charging pump and throttle HPSI loop injection valves.

B. Stop all charging pumps and throttle HPSI loop injection valves.

C. Start an additional HPSI pump and restore letdown.

D✓ Start an additional HPSI pump and begin a RCS cooldown

Question # 2 Rev 0

KA #: 000008 AA1.06 Tier 1 Group 1: Pressurizer Vapor Space Accident
Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: Control of PZR level
Importance 3.6 / 3.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
GIVEN a copy of the HPSI Stop and Throttle Criteria floating step, EXPLAIN the four indications used to determine that HPSI Stop and Throttle Criteria are met.

Question Pedigree
New Question.

K/A Fit:
Question addresses the use of HPSI stop and throttle to help control pressurizer level with a stuck open safety valve.

Choice A:
Distractor: Plausible because this would be correct if HPSI Stop and throttle criteria were all met.

Choice B:
Distractor: Plausible because this would help control pressurizer level, but HPSI stop and throttle criteria are not met and even if they were, all charging pumps should not be tripped.

Choice C:
Distractor. Plausible because an additional HPSI pump should be started but incorrect because letdown should not be restored.

Choice D:
Correct answer. HPSI stop and throttle criteria are not met due to inadequate subcooling, EOP-03 directs ensuring 2 HPSI pumps are running and performing a plant cooldown.

KA#:	000008 AA1.06	Bank Ref #:	N/A
LP# / Objective:	0718-13 03.01	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-03	Handout:	NONE

2.0 FLOATING STEPS

A. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. If emergency boration is required then at least one charging pump must remain running.

2. As natural circulation develops, the expected rise in T_H will reduce subcooling. This may jeopardize HPSI Stop and Throttle Criteria.

3. Reducing SI flow should be approached cautiously.

4. The purpose of HPSI stop and throttle is to prevent an over pressurization of the RCS and a solid PZR, however, maintaining RCS inventory is more important than pressure control.

****

1. Verify **ALL** of the following stop and throttle criteria are satisfied:

   - RCS subcooling is greater than or equal to 20°F

   - PZR level is greater than or equal to 10% and not lowering

   - At least one S/G is available for RCS heat removal

   - RVLMS indicates level is at or above the top of the Hot Leg (43%, ERF "I" display)

INSTRUCTIONS	CONTINGENCY ACTIONS
✘8. **IF** SIAS has actuated, **THEN** maximize Safety Injection and Charging flow to the RCS by operating **ALL** of the following available pumps: • Either HPSI Pumps, SI-2A/B or SI-2B/C • LPSI Pumps, SI-1A/B • Charging Pumps, CH-1A/B/C	8.1 **IF** Safety Injection and Charging flow is **NOT** maximized, **THEN** restore Safety Injection and Charging flow by performing the following: a. Restore electrical power to valves and pumps. b. Ensure **ALL** of the following SI valves are open to align SI for injection: • HPSI Loop Injection Valves • LPSI Loop Injection Valves • HPSI Discharge Cross-Connect Valves • HPSI Header Isolation Valves • SI Pump Suction SIRWT Isolation Valves c. Start **ALL** of the following idle pumps: • HPSI Pumps, SI-2A/B/C • LPSI Pumps, SI-1A/B • Charging Pumps, CH-1A/B/C d. Verify SI flow is acceptable PER Attachment 3, Safety Injection Flow vs. Pressurizer Pressure.

INSTRUCTIONS | CONTINGENCY ACTIONS

22. Place **ALL** Containment Spray Pumps, SI-3A/B/C in "PULL TO LOCK".

NOTE

Voiding of the RCS is indicated by the inability to depressurize to SDC entry pressure. Attachment 14, RCS Void Elimination, provides guidance to correct this condition.

****

CAUTION

1) When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr. When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.

2) No more than three RCP's shall be in operation when RCS temperature is less than 500°F.

****

23. Commence a Steam Generator cooldown by manually operating the Steam Dump and Bypass Valves.

Time: ____________

(continue)

23.1 **IF** manual operation of Steam Dump and Bypass Valves is **NOT** available, **THEN** commence a Steam Generator cooldown by performing step a or b:

a. Steam the S/Gs as follows:

1) Ensure MS-164, "MAIN STEAM LINE "A" STEAM DUMP TO ATMOSPHERE ISOLATION VALVE", is open (Room 81).

(continue)

QUESTION NUMBER: 003

If the primary system temperatures all drop below saturation temperature corresponding to the steam generator pressures during a loss of coolant accident, it indicates that:

A. The ECCS flow is inadequate to remove decay heat.

B. Reflux boiling is taking place in the hot legs.

C.✓ The break flow is adequate to remove decay heat.

D. The safety injection tanks are injecting into the RCS.

Question # 3 Rev 0

KA #: 000009 EK2.03 Tier 1 Group 1: Small Break LOCA
Knowledge of the interrelations between the small break LOCA and the following: S/Gs
Importance 3.0 / 3.3

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
EXPLAIN how the decay heat removal capacity of the break affects plant response.

Question Pedigree
Bank question used on 2002 NRC Exam. Distractor B changed from source question. Not counted as modified question.

K/A Fit:
Question addresses when S/G's are no longer needed for heat removal in a LOCA.

Choice A:
Distractor: Plausible if Applicant believes primary system saturation pressure is less than S/G pressure due to inadequate ECCS flow. Incorrect because decay heat is being removed by the break.

Choice B:
Distractor: Plausible if Applicant believes reflux boiling improves heat transfer and lowers primary temperatures. Incorrect because primary temperature must be greater than secondary temperature for heat transfer to the steam generators to occur.

Choice C:
Correct answer. If primary temperatures are less than secondary temperatures, there will be no heat transfer to the S/G's. The break is removing all decay heat.

Choice D:
Distractor: Plausible if Applicant believes that SIT injection is causing primary temperature to drop. Incorrect, because primary temperatures fall below secondary temperature at a pressure much higher than 250 psig where the SIT's inject.

KA#:	000009 EK2.03	Bank Ref #:	07-15-23 027
LP# / Objective:	0715-23 01.02	Exam Level:	RO-5
Cognitive Level:	HIGH	Source:	BANK
Reference:	SHB 0715-23	Handout:	NONE

OUTLINE OF INSTRUCTION | COMMENTS

II. B. 1.

b. In a large break LOCA, pressurizer pressure will decrease very rapidly. It may decrease to match containment pressure in 30 seconds or less.

c. For most breaks, the depressurization rate will slow down considerably once saturated conditions are reached in the RCS.

d. If the break is too small to remove decay heat, the RCS pressure will not fall below the steam generator pressure. This is because the steam generator is needed to remove heat. If the break is large enough to remove decay heat, the primary pressure will drop below the steam generator pressure.

e. For a very small break LOCA, a pressure balance may develop where the break flow is equal to the charging plus HPSI flow. In this case, some amount of subcooling will remain. If charging or HPSI flow is reduced, the RCS pressure will lower to a new balance point where break flow again equals charging plus HPSI flow. The amount of subcooling will then be reduced.

2. Pressurizer Level

a. Pressurizer level is NOT a good indicator of RCS inventory in a Loss of Coolant Accident.

b. Anytime subcooling has been lost anywhere in the RCS except the pressurizer, steam voids may exist in the RCS. When steam voids are forming or growing, pressurizer level may be increasing while RCS inventory is decreasing.

QUESTIONS REPORT
for ILO EXAM BANK

07-15-23 027

If the primary system temperatures all drop below saturation temperature coresponding to the steam generator pressures during a loss of coolant accident, it indicates that:

A. The ECCS flow is inadequate to remove decay heat.

B. HPSI "Stop and Throttle" should be performed..

C.✓ The break flow is adequate to remove decay heat..

D. The safety injection tanks are injecting into the RCS.

QUESTION NUMBER: 004

5.5 to 6 hours following a large loss of coolant accident which resulted in RAS, the HPSI injection flow is split between the cold legs and the pressurizer auxiliary spray.

Why is this done?

A. To minimize the temperature difference between the pressurizer and the RCS loops.

B. To maintain the RCS pressure low to minimize break flow.

C.✓ To prevent boron precipitation on the fuel assemblies.

D. To equalize boron concentration between the pressurizer and the loops.

Question # 4 Rev 0

KA #: 000011 EK3.12 Tier 1 Group 1:Large Break LOCA
Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: Actions contained in EOP for emergency LOCA (large break)
Importance 4.4 / 4.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
EXPLAIN how boric acid precipitation in the reactor vessel is prevented following a Loss of Coolant Accident.

Question Pedigree
Bank Question used on 2005 NRC exam.

K/A Fit:
A large break LOCA will require simultaneous hot and cold leg injection 5.5 to 6 hours after LOCA initiation to prevent boron precipitation on the fuel assemblies. Hot leg injection uses pressurizes auxilary spray at Fort Calhoun.

Choice A:
Distractor: Plausible because auxiliary spray will minimize the temperature difference between the pressurizer and the loops. Incorrect because this is not the purpose.

Choice B:
Distractor: Plausible because auxiliary spray can be used to lower pressure. Incorrect because this is not the purpose.

Choice C:
Correct answer: Simultaneous hot and cold leg injection, using auxiliary spray is performed to prevent boron precipitation on the fuel assemblies.

Choice D:
Distractor: Plausible because pressurizer spray is maximized during boration and dilution to equalize boron concentration between the pressurizer and the RCS loops. Incorrect because this is not the purpose following a LOCA.

KA#:	000011 EK3.12	Bank Ref #:	07-15-23 001
LP# / Objective:	0715-23 02.07	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	BANK
Reference:	TBD-EOP-03	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

Simultaneous hot and cold leg injection must be implemented between five and one-half and six hours after a LOCA with RCS pressure less than 140 psia if SDC can not be established.

****

✗ 43. **IF BOTH** of the following criteria are satisfied:

- RCS Pressure is less than 140 psia
- Five and one-half hours may elapse before SDC can be established

THEN IMPLEMENT Attachment 9, Simultaneous Hot and Cold Leg Injection.

Time: ___________

EPG Step: 40

Deviation:

1) The EOP specifies establishment of simultaneous hot and cold leg injection between 5.5 and 6 hours into the event if normal shutdown cooling system operation is not available. The EPG uses a band of 2-4 hours and lists conditions instead of stating that if SDC operation can not be established.
2) The simultaneous hot leg injection path that is discussed in Attachment 9 uses the auxiliary Pressurizer spray line and the Pressurizer surge line.

When an unisolated LOCA event exists, simultaneous hot leg and cold leg injection should be implemented if the plant cannot be placed on shutdown cooling within six hours of the LOCA initiation and RCS pressure is less than 120 psig, 140 psia is used as a conservative value to meet this condition [R4]. The procedure is implemented at five and one-half hours to provide adequate time to align simultaneous hot/cold leg injection before the six hour time limit.

Injecting to each side of the Reactor Vessel at an injection rate of 147 gpm to the cold legs and 134 gpm to the hot legs, ensures that fluid from the Reactor Vessel (where the boric acid is being concentrated) flows out of the break regardless of the break location and is replenished with a dilute solution of borated water from the other side of the Reactor Vessel [R3].

INSTRUCTIONS | CONTINGENCY ACTIONS

The action is taken between 5.5 and 6 hours after the LOCA in order to ensure that the buildup of boric acid is terminated well before the potential for boric acid precipitation occurs. Once the RCS is refilled, the boric acid is dispersed throughout the RCS via natural circulation. If entry into shutdown cooling system operation is anticipated before the 5.5 hour limit, then the realignment to hot/cold leg injection is unnecessary. This justifies deviation 1.

Hot leg injection path that is used during simultaneous hot and cold leg injection is via containment sump to available HPSI pump configuration to auxiliary Pressurizer spray line to Pressurizer to Pressurizer surge line to RCS hot leg. This justifies deviation 2.

✘ 44. **WHEN ALL** of the following SDC entry conditions are established:

- PZR level is greater than or equal to 45% and constant or rising
- RCS subcooling is greater than or equal to 20°F
- RCS pressure is less than or equal to 300 psia.
- RCS T_C less than 350°F

THEN initiate SDC operation PER **ONE** of the following Attachments:

- Attachment 4, SDC Without RAS
- Attachment 7, SDC with RAS
- Attachment 8, Cooled SI Flow with RAS

Time: ____________

44.1 **IF** SDC can **NOT** be established, **THEN** GO TO Step 45.

EPG Step: 45, 46

Deviation:

1) For unisolated LOCA entry into Shutdown Cooling, the EPG criterion on RCS activity level is not included.
2) The EOP specifies a temperature band less than 350°F for RCS T_C. The EPG specifies using T_H.
3) The EOP is not exited to implement a long term heat removal method that utilizes the Shutdown Cooling Heat Exchangers. Rather several Attachments that look at several different configurations are used. The EPG directs exiting to an operating instruction.

QUESTION NUMBER: 005

Annunciator CB-1/2/3, A6, "REACTOR COOLANT PUMP RC-3A VIBRATION HI" came into alarm when Reactor Coolant Pump RC-3A was started.

What action must be taken to reset the high vibration alarm once the vibration is below the alarm setpoint?

A. Reset the alarm on PC-69 located in the plant computer room (Room 73).

B. Reset the alarm on PC-69 located in the upper electrical penetration room (Room 57).

C. Reset the alarm at AI-270 located in the plant computer room (Room 73)

D.✓ Reset the alarm at AI-270 located in the upper electrical penetration room (Room 57)

Question # 5 Rev 0

KA #: 000015 2.1.30 Tier 1 Group 1
Reactor Coolant Pump Malfunctions
Ability to locate and operate components, including local controls.
Importance 4.4 / 4.0

CFR Number: 55.41(b)(3)
Mechanical components and design features of the reactor primary system.

Fort Calhoun Objective:
DISCUSS the prerequisites and precautions for operating an RCP.

Question Pedigree
Bank question used on the 2001-1 NRC exam. Reworded stem and changed 3 distractors to include location. Not counted as modified question.

K/A Fit:
AI-270 is a local panel where RCP vibration alarms can be reset.

Choice A:
Distractor: Plausible because RCP vibration alarms can be acknowleged on PC-69 and the plant computer room is a plausible location. Incorrect location and alarm can not be reset on PC-66.

Choice B:
Distractor: Plausible because PC-69 is located in room 57 and can be used to acknowledge RCP vibration alarms. Incorrect because the alarms can not be reset on PC-66.

Choice C:
Distractor: Plausible because the alarm can be reset on AI-270. Incorrect because AI-270 is not located in room 73.

Choice D:
Correct Answer: The alarm can be reset on AI-270. AI-270 is located in room 57.

KA#:	000015 2.1.30	Bank Ref #:	07-11-20 031
LP# / Objective:	0711-20 03.02A	Exam Level:	RO-3
Cognitive Level:	LOW	Source:	BANK
Reference:	ARP CB-1,2,3/A6	Handout:	NONE

Panel: **CB-1/2/3**	Annunciator: **A6**	Window: **A-4**

REACTOR COOLANT PUMP A VIBRATION HIGH

SAFETY RELATED

REACTOR COOLANT PUMP RC-3A VIBRATION HI

Tech Spec References: None

Initiating Device AI-270A Setpoint REF OI-RC-13 Power AI-42B

OPERATOR ACTIONS

NOTE: This annunciator may be deactivated if RC-3A vibrations are monitored every three hours.

NOTE: OI-RC-13, Attachment 3, "Alarm Response and Monitoring/Trending of RCP Vibrations", should be referenced in conjunction with this ARP guidance.

1. Dispatch an operator to AI-270 (Room 57) to report vibration indications and to determine the validity of the vibration alarm.

   1.1 If the vibration alarm is due to an electronic problem in the monitoring system, the dual vibration unit that caused the alarm will be in "Bypass" (AI-270A) and/or the "OK" light will be out.

   1.2 Vibration indications can be validated by:

      1.2.1 Verify the redundant probe for the monitoring point has a similar trend in vibration level.

      1.2.2 Corresponding changes in vibration levels on other monitoring points for RC-3A. One point in alarm with no corresponding change in vibration levels of other points indicates a monitoring equipment problem.

(continue)

PROBABLE CAUSES

- Pump Startup/Shutdown
- Loose pump or motor parts
- Plant Transient/Trip
- Worn or damaged bearings

REFERENCES

OI-RC-9 OI-RC-13 D-23866-210-111 Sh 1 10473

Panel: **CB-1/2/3**	Annunciator: **A6**	Window: **A-4**

REACTOR COOLANT PUMP A VIBRATION HIGH Page 2 of 3

SAFETY RELATED

OPERATOR ACTIONS (continued)

1.2.3 Points monitored by velocity transducers (seal flange and motor casing) are for diagnostic purposes only. Response to an alarm monitored by velocity transducers should be validated by a review of the other points monitored by proximity probes which indicate actual shaft displacement.

2. IF an alarm is proven to be valid and has reached the alert setpoint during power operation, THEN perform the following:

NOTE: RCP vibration alarm setpoints have been set to monitor RCPs during normal RCS temperature and pressure with 4 RCP's running. Alarms can be expected during RCP starting, stopping, during plant transients (plant trip) or other changes in RCS temperature and pressure.

2.1 Reset the alarm on AI-270A (Room 57).

2.2 If the alarm does not clear, the responsible System Engineer should be contacted to begin diagnostic efforts and trending.

2.3 Reference OI-RC-13 for vibration Baseline, Action Level and Shutdown Limit values.

3. IF an alarm is proven to be valid and has reached the danger setpoint during power operation, THEN perform the following:

3.1 Reset the alarm on AI-270A (Room 57).

(continue)

2.4.2 Design/Specifications

A. The following detectors are installed on each reactor coolant pump:

Table 1 - Reactor Coolant Pump Detectors	
Detector	**Quantity**
Seal Pressure	3
Seal Temperature	1
Controlled bleed off flow	1
Controlled bleedoff temperature	1
Motor oil level	2
Motor bearing temperature	4
Motor stator temperature (one in service)	6
Oil lift pressure	1
Pump vibration (AI-270)	10
Gasket bleedoff pressure	1
Pump rotation	4
Low pressure oil flow	1 (except RC-3B)
Oil pump suction filter pressure	1
Main impeller flow	1
Motor amps	1

B. The reactor coolant pump seal system is monitored by three pressure detectors and two temperature detectors. The three full-pressure seals (lower, middle, and upper) are mounted in series, with the middle and upper seals having a pressure detector that senses the pressure below the seal Additionally, bleed off temperature and pressure detectors are provided.

C. The reactor coolant pump motor is provided with stator winding temperature detectors, bearing temperature detectors, and oil reservoir level detectors which provide alarms to alert the operators to any abnormal conditions.

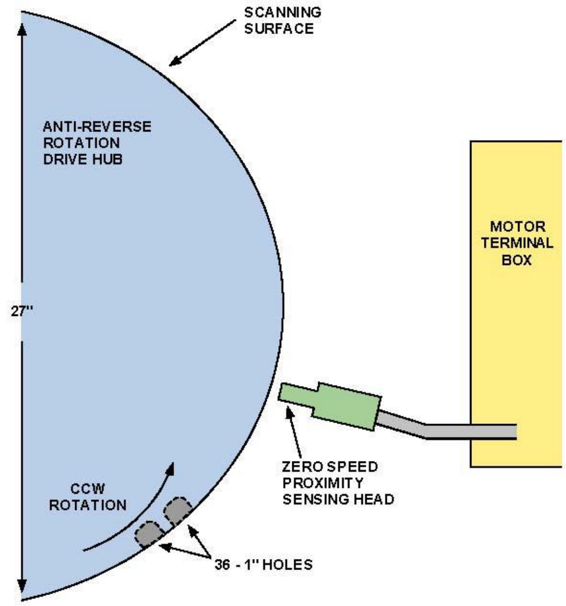
D. The Reactor Coolant Pump (RCP) Vibration Monitoring System trends, logs, and analyzes vibration on the four reactor coolant pumps (RC-3A/RC-3B/RC-3C/RC-3D) and their associated motors. This system allows for predictive and preventive maintenance to be performed in order to avoid possible equipment damage. The system consists of vibration and speed detection devices, signal conditioners, vibration monitoring instrumentation panels, and a system computer and printer.

2.4.2 (continued)

E. The vibration detection equipment is located on the pumps and pump motors. The detectors consist of proximity and velocity probes. The proximity probes monitor shaft vibration at each bearing location. The velocity probes measure overall pump casing vibration displacement. The vibration data is monitored on panels AI-270A/AI-270B/AI-270C/ AI-270D located in the upper electrical penetration area, Room 57.

F. Reactor coolant pump speed detection is also performed by a proximity probe which monitors holes drilled in the speed ring on the motor shaft for input to a tachometer.

Figure 2-10 - RCP Speed Sensors



G. The system computer monitors the system, records measured values, and manipulates the data to provide trending as well as graphs and plots of data manipulations to allow a visual display of the vibration analysis. The system monitor and system computer are also located in the upper electrical penetration area, Room 57.

2.4.3M (continued)

12. Each RCP oil reservoir level detector actuates a computer alarm at high and low levels. For RC-3A, the upper oil reservoir is monitored and has a normal range of 75% to 95%, with alarm setpoints that may vary after each calibration. The computer range for this value is 0 to 100%, corresponding to 0 to 15 inches of oil. The lower oil reservoir of RC-3A has a normal range of 80% to 110% and alarm setpoints that vary. The computer range for this value is 0 to 100%, corresponding to 0 to 5 inches of oil.

13. The RCP Vibration Monitoring System is programmed with setpoints to alarm when vibration limits are exceeded or when set speed limits are met. The alarms are local at panel AI-270. The system also provides input to the REACTOR COOLANT PUMP RC-3A (RC-3B, RC-3C, RC-3D) VIBRATION HI alarm on CB-1,2,3, panel A6. To reset an alarm, the vibration level must be below the setpoints and the alarm must be reset locally at the respective AI-270 panel. Instructions on how to reset the alarm and other operating information is detailed in procedure OI-RC-13, Operation of the RCP Vibration Monitoring System.

14. The RCP Oil Lift System pressure switch (PCS-3109 for RC-3A) also actuates the REACTOR COOLANT PUMP RC-3A (RC-3B, RC-3C, RC-3D) OIL LIFT PRESS HI alarm at 750 psig on CB-1,2,3, panel A6.

15. When the oil lift pumps suction filter needs cleaning, a differential pressure switch actuates a computer alarm. The filter has a lever on the side that points to a "Needs Cleaning" label. When flow through the cooler is less than 10 gpm, low-pressure switch FAS-3182 actuates the LUBE OIL LOW FLOW alarm on the computer.

16. Each reactor coolant pump has a tachometer located at AI-270, Room 57, which provides direct indication of RCP rotational speed. The RCP speed tachometer provides contacts which will illuminate the green zero speed light on CB-1,2,3 when the rotation speed approaches zero rpm.

17. Reactor coolant pump RC-3A's reverse rotation switch OCS-3112 is actuated when a change in the direction of the oil being flung off the thrust runner is sensed. A flow of 3.5 gpm causes the switch to actuate the REACTOR COOLANT PUMP RC-3A (RC-3B, RC-3C, RC-3D) REVERSE ROTATION alarm on CB-1,2,3, panel A6 (RC-3B setpoint is greater than 5 rpm in the reverse direction).

QUESTIONS REPORT
for ILO EXAM BANK

07-11-20 031

Annunciator CB-1/2/3, A6, "REACTOR COOLANT PUMP RC-3A VIBRATION HI" is in alarm. What action, if any, must be taken to reset the high vibration alarm once the vibration is below the alarm setpoint?

A. The alarm will automatically reset.

B. The alarm can be manually reset at CB-1/2/3.

C.✓ The alarm can be reset at AI-270.

D. The alarm can be reset at an ERFCS terminal.

QUESTION NUMBER: 006

The following Plant Conditions exist:

- The plant is operating at full power
- Letdown control valves are in manual
- Charging Pump CH-1A is in operation
- Charging Pumps CH-1B and CH-1C are in the After-Stop position
- The standby selector switch is in the CH-1B/1C position
- Pressurizer level control channel LRC-101X is selected as the controlling channel and is in CASCADE
- Charging flow is 32 gpm and slowly lowering due to a pump malfunction

Assuming no operator actions are taken and the pressurizer level continues to lower, when will CH-1B start?

A. When the "charging flow lo" alarm comes in.

B.✓ When the level error is -2.9%

C. When the level error is -3.3%

D. When the level error is -6.0%

Question # 6 Rev 0

KA #: 000022 AK1.03 Tier 1 Group 1
Loss of Reactor Coolant Makeup
Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level
Importance 3.0 / 3.4

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Fort Calhoun Objective:
EXPLAIN the interlocks and control functions associated with RCS Instrumentation.

Question Pedigree
New question.

K/A Fit:
Question addresses how additional charging pumps increase RCS makeup flow as pressurizer level lowers following a pump malfunction.

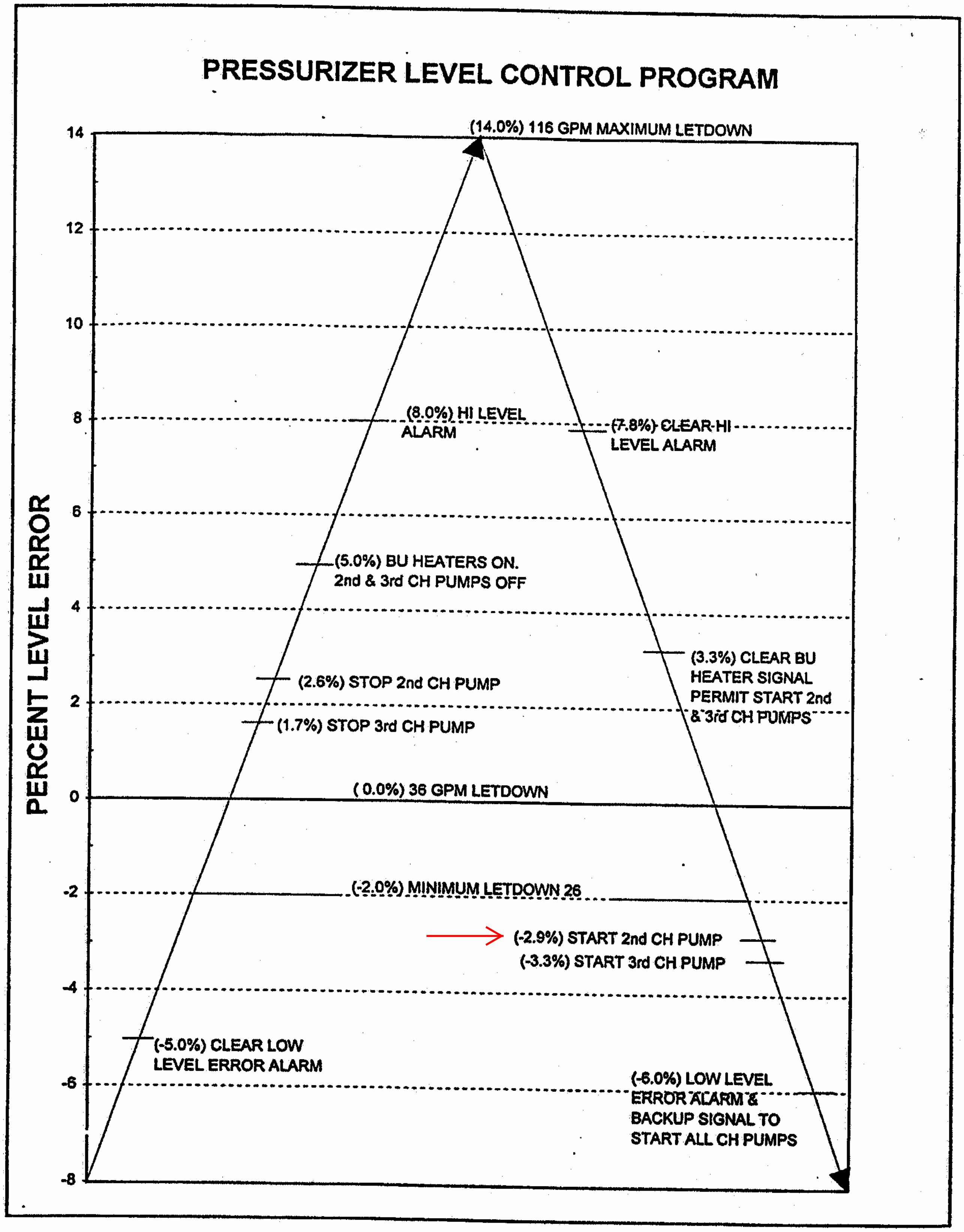
Choice A:
Distractor: Plausible if Applicant believes 2nd pump starts at the charging flow low setpoint. Incorrect, because 2nd pump starts due to level deviation.

Choice B:
Correct answer: CH-1B is selected as the first standby charging pump. The first standby pumps starts with a level deviation of -2.9%.

Choice C:
Distractor: Plausible because the standby charging pumps will autostart based on level deviation. Incorrect because this is where the second backup pump, CH-1C, will start.

Choice D:
Distractor: Plausible because the standby charging pumps will autostart based on level deviation. Incorrect because this is where a backup start signal will occur to start all charging pumps.

KA#:	000022 AK1.03	Bank Ref #:	N/A
LP# / Objective:	0711-20 04.04	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	NEW
Reference:	STM 37	Handout:	NONE


PRESSURIZER LEVEL CONTROL PROGRAM
PERCENT LEVEL ERROR
14
12
10
8
6
4
2
0
-2
-4
-6
-8
(14.0%) 116 GPM MAXIMUM LETDOWN
(8.0%) HI LEVEL ALARM
(7.8%) CLEAR HI LEVEL ALARM
(5.0%) BU HEATERS ON. 2nd & 3rd CH PUMPS OFF
(3.3%) CLEAR BU HEATER SIGNAL PERMIT START 2nd & 3rd CH PUMPS
(2.6%) STOP 2nd CH PUMP
(1.7%) STOP 3rd CH PUMP
(0.0%) 36 GPM LETDOWN
(-2.0%) MINIMUM LETDOWN 26
(-2.9%) START 2nd CH PUMP
(-3.3%) START 3rd CH PUMP
(-5.0%) CLEAR LOW LEVEL ERROR ALARM
(-6.0%) LOW LEVEL ERROR ALARM & BACKUP SIGNAL TO START ALL CH PUMPS

QUESTION NUMBER: 007

The following plant conditions exist:

- The plant operated for 11 months at full power before shutting down to replace a reactor coolant pump seal.
- The reactor has been shutdown for 5 days
- The RCS is currently at Mid-Loop for the RCP seal maintenance
- The plant is on shutdown cooling with RCS Temperature at 126°F
- Core Exit Thermocouple and Heated Junction Thermocouple indications are NOT available

How long would it take for the RCS to begin to boil if shutdown cooling was lost and not restored?

A. 9 -11 minutes

B✔ 16 - 18 minutes

C. 22 - 24 minutes

D. 37 - 39 minutes

Question # 7 Rev 0

KA #: 000025 AK1.01 Tier 1 Group 1: Loss of Residual Heat Removal System
Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation
Importance 3.9 / 4.3

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Describe how the plant responds to a Loss of Shutdown Cooling in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

Question Pedigree
New question

K/A Fit:
Question addresses the consequences of a loss of shutdown cooling while in cold shutdown, mid-loop conditions.

Choice A:
Distractor: This is the correct answer for 1 day after shutdown but incorrect because the stem condition is 5 days after shutdown.

Choice B:
Correct Answer: This is the correct answer for conditions given in the stem per AOP-19 Attachment B.

Choice C:
Distractor: This is the correct answer for 10 days after shutdown but incorrect because the stem condition is 5 days after shutdown.

Choice D:
Distractor: This is the correct answer for 30 days after shutdown but incorrect because the stem condition is 5 days after shutdown.

KA#:	000025 AK1.01	Bank Ref #:	N/A
LP# / Objective:	0717-19 01.02	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	AOP-19 ATT B	Handout:	AOP-19 ATT B

Attachment B

Time to Boil Determination Worksheet

1. Time Shutdown Cooling was lost: __________

2. Last known RCS/SDCS temperature: __________°F
from instrument number:______

3. Record the following information and inform the Shift Manager on 10 minute intervals.

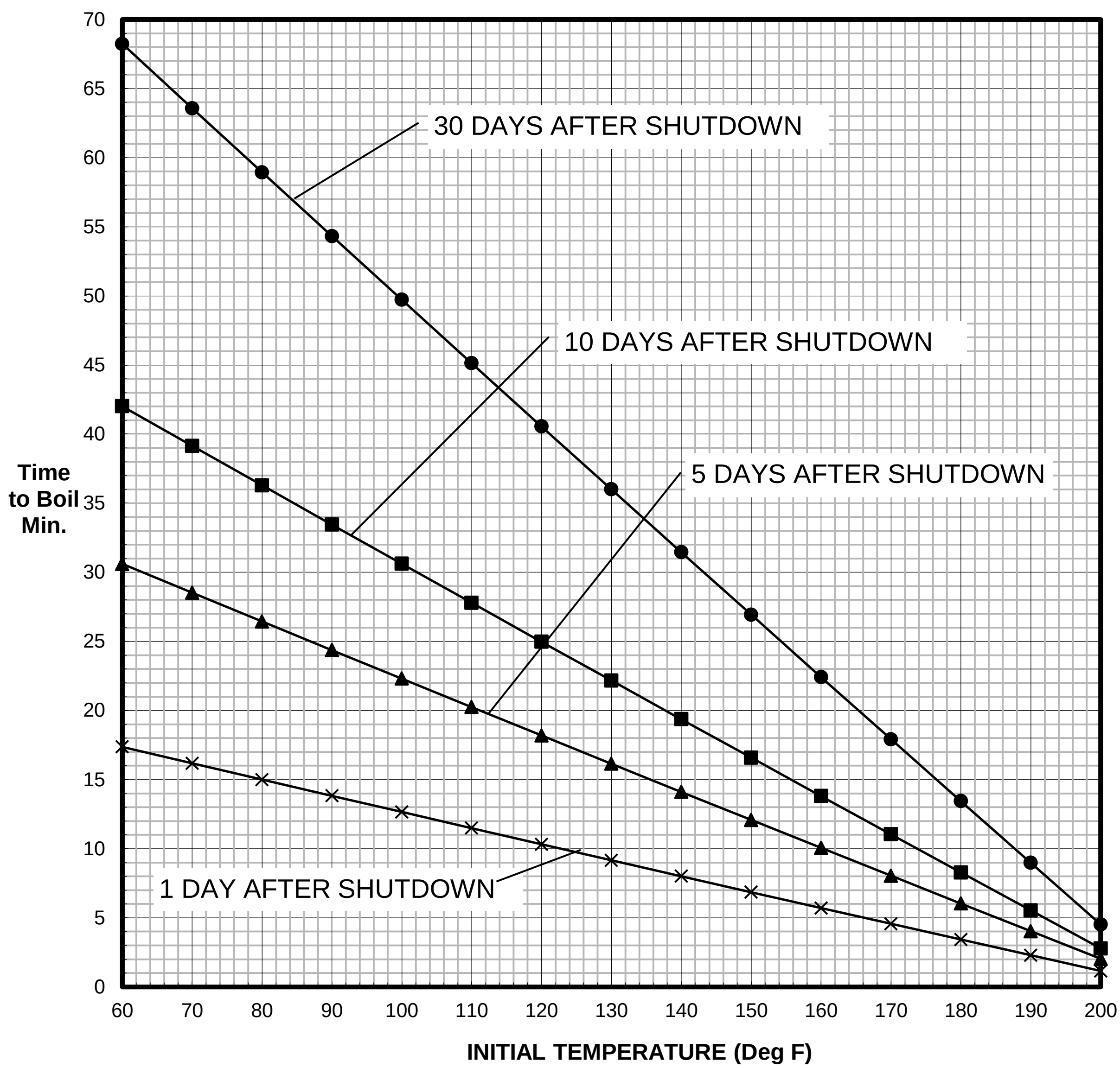
TIME	CET/HJTC °F	HEATUP RATE	TIME TO BOIL

Alternate Method: $T_b = T_a + T_0 - T_c$
Where: T_b is the remaining time to boil
T_a is the approximate time to boil from the appropriate curve
T_o is the time SDC was lost
T_c is the current time

Attachment B

Time to Boil Determination Worksheet

Time to Boil (RCS at Mid Loop)

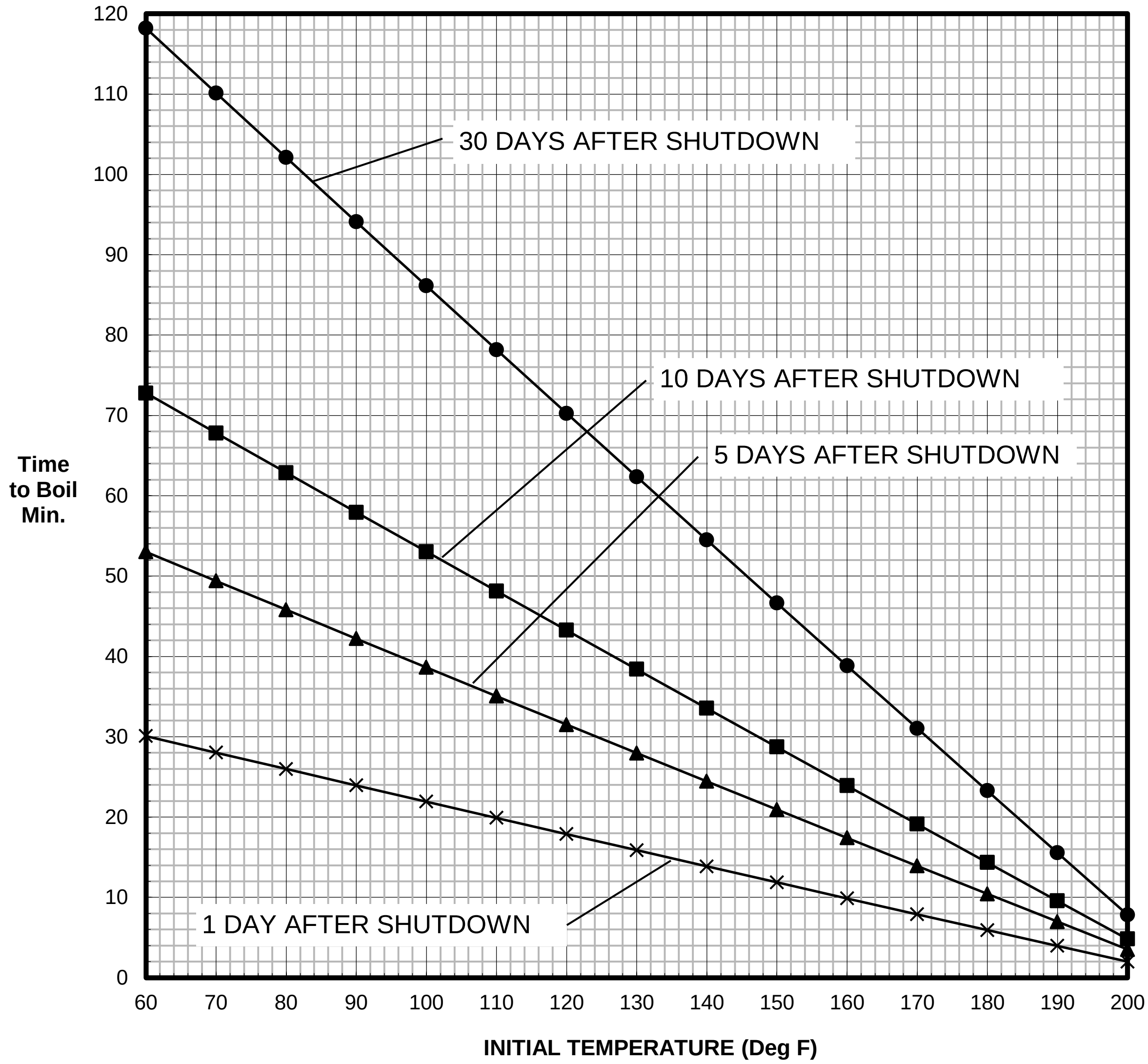


Attachment B

Time to Boil Determination Worksheet

Time to Boil (RCS at Reactor Vessel Flange

Steam Generators Unavailable)

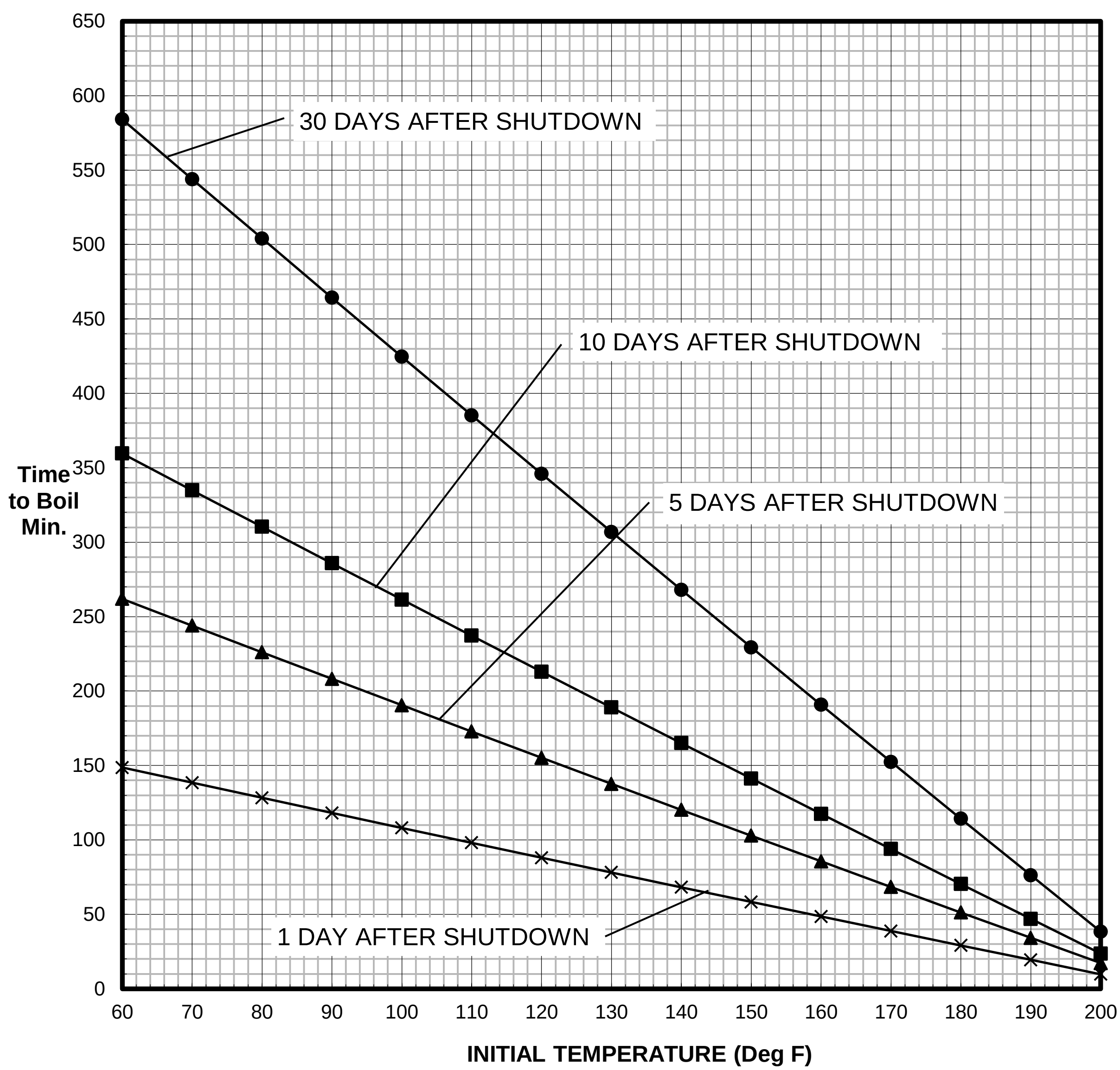


Attachment B

Time to Boil Determination Worksheet

Time to Boil (RCS Open, RV Head on, RCS Level > 1013',

Steam Generator U-tubes Filled, One Steam Generator in Wet Layup)

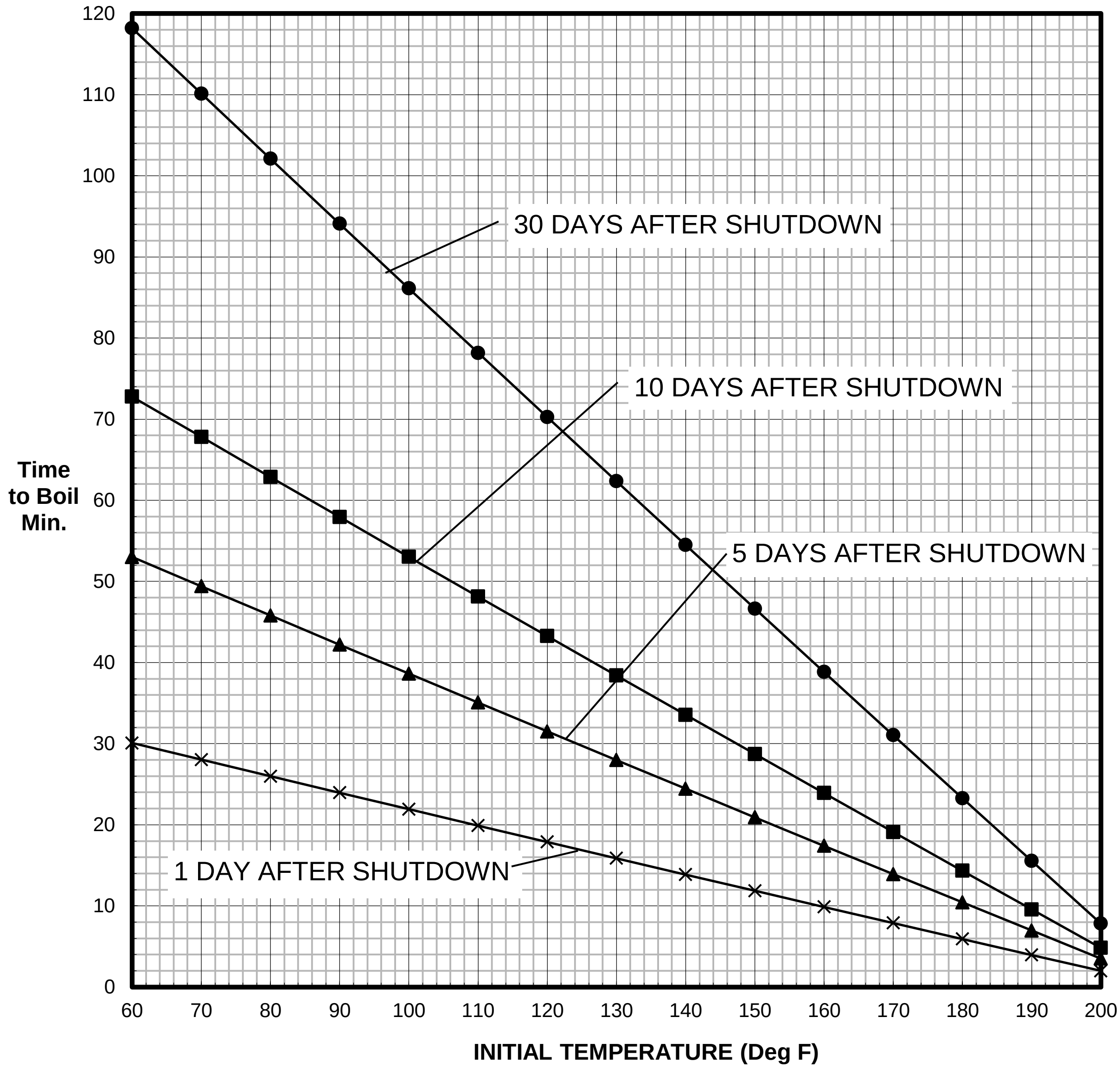


Attachment B

Time to Boil Determination Worksheet

Time to Boil (RCS at Normal Operating Volume

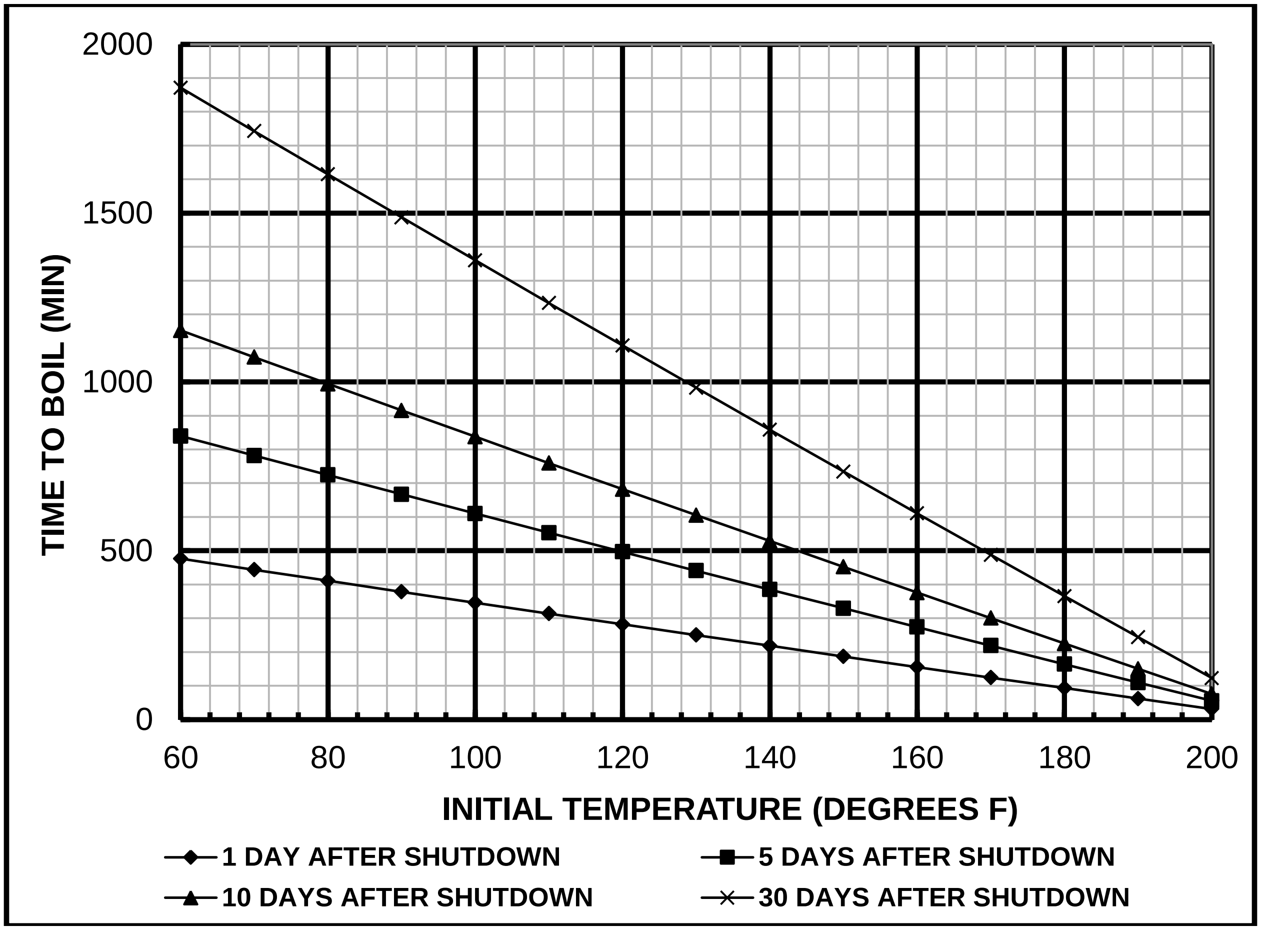
Steam Generators Unavailable)



Attachment B

Time to Boil Determination Worksheet

Time to Boil (Refueling Cavity Flooded)



End of Attachment B

QUESTION NUMBER: 008

CCW flow to Containment Cooler, VA-1A automatically isolated due to low CCW flow following a CIAS actuation. CCW Pump discharge pressure and flow are normal.

What must be done to reestablish flow to the containment cooler?

A. The Control switch for HCV-400A/C must be placed in the "ISOL" position momentarily and then returned to the "NOR" position.

B. The Control switch for HCV-400A/C must be placed in the "CIRC" position momentarily and then returned to the "NOR" position.

C. The Control switch for HCV-400A/C must be held in the "ISOL" position until the low flow signal has reset.

D.✓ The Control switch for HCV-400A/C must be held in the "CIRC" position until the low flow signal has reset.

Question # 8 Rev 0

KA #: 000026 AA1.02 Tier 1 Group 1: Loss of Component Cooling Water
Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: Loads on the CCWS in the control room
Importance 3.2 / 3.3

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
EXPLAIN the operation of controls associated with the CCW System valves operated from the Control Room.

Question Pedigree
Bank question used on 2007 NRC exam.

K/A Fit:
Question addresses operation of CCW valves to restore flow to a containment cooler following a low flow isolation.

Choice A:
Distractor: Plausible because this would be a possible logic combination. Incorrect because the control switch must be placed in the CIRC position until the low flow signal resets.

Choice B:
Distractor: Plausible because the control switch must be placed in the CIRC position. Incorrect because it must be held in that position until the low flow signal resets.

Choice C:
Distractor: Plausible because the control switch must be held until the low flow signal resets. Incorrect because it must be held in the CIRC position.

Choice D:
Correct Answer: The control switch must be held in the CIRC position until the low flow signal resets.

KA#:	000026 AA1.02	Bank Ref #:	07-11-06 063
LP# / Objective:	0711-06 01.02	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	BANK
Reference:	STM 08 2.188	Handout:	NONE

••Location

2.184 The containment air cooling and filtering unit CCW isolation valves HCV-400A/B/C/D through HCV-403A/B/C/D are on the 1025 foot elevation of the Auxiliary Building at the east end of Room 69, southwest of the CCW pumps.

••Power Supplies

2.185 Control power is supplied from opposite 125 VDC buses to ensure isolation of CCW to a component is possible coincident with a loss of one DC bus.

••Instrumentation and Controls

2.186 The following is a detailed description of the Instrumentation and Controls of the Containment air cooling and filtering unit CCW isolation valves.

•••Local

2.187 Not applicable

•••Remote

2.188 Each cooling/filtering unit has two control switches that are on the vertical section of CB-1/2/3. Each control switch operates one pair of CCW isolation valves.

2.189 The control switch positions for the A/C sets of valves are ISOL, NOR, and CIRC. The control switches for the B/D sets of valves also have three positions: CLOSE, NOR, and OPEN. The switches spring return to center (NORMAL). Placing a switch in ISOL will close the A/C set of valves if CIAS is reset. Placing the switch in CIRC opens the inlet valve and enables the controller for modulation of the outlet valve. The B/D valves are either opened or closed; the outlet valves are not modulated.

2.190 Holding the control switch in CIRC will override a low-flow signal and allow the operator to reopen the valves. This low-flow signal is only enabled if a CIAS is initiated. The low-flow signal is designed to allow isolation of CCW to a cooling and filtering unit that might develop a leak or rupture in the cooling coils.

•••Interlocks

2.191 The inlet and outlet valves open on a CIAS. A flow signal closes the valves if a CIAS is present and there is no flow out of the cooler after a variable time delay.

•••Alarms and Indications

2.192 The CC WATER FROM COIL VA-1A (1B) NO FLOW alarm (two windows) actuates if the flow sensed by FC-416A (FC-417A) for VA-1A (VA-1B) is less than the setpoint of approximately zero

QUESTION NUMBER: 009

The reactor tripped 20 minutes ago. The following conditions are observed:

- "PRESSURIZER PRESSURE OFF NORMAL HI-LO" channel X and Y are in alarm.
- "PRESSURIZER LEVEL HI-LO" channel X and Y are in alarm.
- PRC-103X (controlling channel) indicates 2020 psia and stable
- PRC-103Y indicates 2150 psia and rising
- All backup heaters in AUTO and energized
- LRC-101X indicates 40% and rising
- LRC-101Y (controlling channel) indicates 41% and rising
- Letdown flow is 26 gpm
- All charging pump are running
- T-cold indicates 533°F, T-hot indicates 534°F, both are stable

Select the probable cause and the action that should be taken to restore RCS pressure:

A. Low level on LRC-101X is maintaining the B/U heaters on, place the pressurizer heater cutout channel select switch in channel Y.

B. The bistable for the B/U heaters needs to be reset, place the control switches for all B/U heaters to reset and back to auto.

C. LRC-101Y has malfunctioned causing the B/U heaters to remain on, place LRC-101X in service.

D✓ PRC 103X has malfunctioned causing the B/U heaters to remain on, place PRC-103Y in service.

Question # 9 Rev 0

KA #: 000027 AK2.03 Tier 1 Group 1:Pressurizer Pressure Control System Malfunction
Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners
Importance 2.6 / 2.8

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
Given a current copy of ARP, EXPLAIN the alarms associated with the RCS Instrumentation System and the required actions.

Question Pedigree
The source question is a bank question that was used on the 2004 NRC exam. The question was modified by changing the conditions in the stem to make another choice correct.

K/A Fit:
Question addresses a pressure control malfunction of a pressurizer pressure controller.

Choice A:
Distractor: Plausible because a low level on either channel will actuate an interlock. Incorrect because the setpoint is 32% and the interlock deenergizes the heaters.

Choice B:
Distractor: Plausible because there are pressurizer level bistables that can be reset. Incorrect because the bistables affect the charging pumps not the heaters.

Choice C:
Distractor: Plausible because LRC-101Y is the controlling channel and a level deviation of 5% for program (program is 48%) will cause heaters to turn on. Incorrect because the level deviation must be 5% above programmed level.

Choice D:
Correct Answer: PRC-103X has failed at 2020 psia causing the heaters to remain on. The correct pressure is indicated by the non-controlling channel, PRC-103Y.

KA#:	000027 AK2.03	Bank Ref #:	07-11-20 008
LP# / Objective:	0711-20 05.04	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	STM 37	Handout:	NONE

Panel: **CB-1/2/3**	Annunciator: **A4**	Window: **A-4**

PRESSURIZER PRESSURE DEVIATION

Page 1 of 2

SAFETY RELATED

PRESSURIZER
PRESSURE
OFF NORMAL
HI-LO
CHANNEL X

Tech Spec References: 2.10.4(5)(a)(ii)

Initiating Device PR-103X

Setpoint HI > 2145 psia
Setpoint LO < 2080 psia

Power PQ-103X/AI-40A

OPERATOR ACTIONS

1. Check RCS Pressure using all available indications.

2. IF actual pressure is low, THEN perform the following:

   2.1 Apply Technical Specification LCO Action Statement if pressure is less than or equal to 2075 psia.

   2.2 Check Pressurizer Heaters are energized.

   2.3 Ensure the Pressurizer Spray Valves are closed.

   2.4 Ensure the Pressurizer Heater Control Switches are in AUTO.

   2.5 Check Pressurizer Level for decreasing trend (LR-101X, LR-101Y, CB-2).

   2.6 Check VCT Level for level trend (YR-4100, CB-2).

(continue)

PROBABLE CAUSES

- Plant Power transient
- Pressurizer Spray Valve open
- Pressurizer Heaters on
- Pressurizer Spray Valve closed
- Turbine Load/Reactor Power Mismatch

REFERENCES

136B3081 Sh 21 06245	161F561 Sh 3 09478	161F561 Sh 19 09503
EM-103 01567	IC-CP-01-0103X	E-23866-210-110 10475

Panel: **CB-1/2/3**	Annunciator: **A4**	Window: **A-4**

PRESSURIZER PRESSURE DEVIATION Page 2 of 2

SAFETY RELATED

OPERATOR ACTIONS (continued)

3. IF actual pressure is high, THEN perform the following:

    3.1 Ensure the Pressurizer Spray Valves are open and controlling Pressurizer Pressure.

    3.2 Ensure the Pressurizer Heaters are not energized.

4. Monitor Pressurizer Pressure and the operation of PC-103X.

    4.1 IF PC-103X is not controlling pressure, THEN shift pressure control to PC-103Y (HC-103, CB-3).

    4.2 If necessary, take manual control of Pressurizer Pressure.

Panel: **CB-1/2/3**	Annunciator: **A4**	Window: **A-8**

PRESSURIZER LEVEL DEVIATION
FROM PROGRAMMED LEVEL

Page 1 of 2

SAFETY RELATED

PRESSURIZER
LEVEL
HI-LO
CHANNEL X

Tech Spec References: 2.1.4

Initiating Device LA-101X
LCA-101X

Setpoint HI 8.0% > program
Setpoint LO 6.0% < program

Power LQ-101X / AI-40A

OPERATOR ACTIONS

1. Check Pressurizer Level on LR-101X and Y (CB-3).

2. IF Pressurizer Level is Low, THEN perform the following:

   2.1 Ensure all available Charging Pumps are running and supplying flow to the RCS (FIA-236, CB-2).

   2.2 Ensure Letdown Flow is at minimum (26 gpm).

   2.3 Check VCT Level for trend (YR-4100, CB-2).

   2.4 Check the position of LCV-101-1 and LCV-101-2.

      2.4.1 IF LCV-101-1 or -2 are not at minimum position, THEN place HC-101 Pressurizer Level Chan Selector Switch to the Alternate Channel.

(continue)

PROBABLE CAUSES

- Pressurizer Level Channel failed High or Low
- Turbine Load/Reactor Power Mismatch
- Improper functioning of the Letdown Level Control Valves
- Loss of Makeup capabilities
- Charging Pump fail to start
- Leakage in the Reactor Coolant or Chemical and Volume Control System

REFERENCES

136B3081 Sh 23 06247	161F561 Sh 3 09478	161F561 Sh 27 09512
EM-101 10227	IC-ST-RC-0004	E-23866-210-110 10475

Panel: **CB-1/2/3**	Annunciator: **A4**	Window: **A-8**

PRESSURIZED LEVEL DEVIATION
FROM PROGRAMMED LEVEL

Page 2 of 2

SAFETY RELATED

OPERATOR ACTIONS (continued)

2.5 Monitor T_{avg} versus Pressurizer Level for indication of leakage.

2.5.1 IF low level is due to system leak, THEN GO TO AOP-22.

3. IF Pressurizer Level is High, THEN perform the following:

3.1 Check VCT Level for trend (YR-4100, CB-2).

3.2 Ensure only one Charging Pump is running.

3.3 Ensure Letdown Flow is at maximum.

3.4 Ensure Pressurizer Backup Heaters are Energized.

3.5 Check the position of LCV-101-1 and LCV-101-2.

3.5.1 IF LCV-101-1 or -2 is not fully open, THEN place HC-101 Pressurizer Level Chan Selector Switch to the Alternate Channel.

4. IF Auto PZR Level Control does not control Level, THEN take manual control.

5. IF LI-101X has failed, THEN notify I&C and refer to Technical Specification 2.15.5.

5.1.1 Place HC-101-1, Pzr Heater Cutout Channel Select Switch, to Chan Y.

channel fails, the switch is used to select the controlling (operable) channel (X or Y). **NOTE:** When control via one channel is selected, the opposite channel is defeated.

2.125 The manual/automatic transfer station and bias station, HIC-101, receives the signal from LC-101X/Y through an isolator module (IY-101) with a time delay and a signal converter (IY-101-1) to control the position of the letdown throttle valves. The output of HIC-101 is adjusted with the bias dial so that pressurizer level may be controlled at the program setpoint; whether one, two, or three charging pumps are in operation. A detailed description of HIC-101 is in the STM for CVCS.

2.126 Signal limiter LM-101 limits the letdown flow to between 26 and 116 gpm. Both the manual and auto signals from HIC-101 are limited by LM-101X. The limiter bypass switch allows letdown flow to vary from no flow to 126 gpm.

2.127 Selector switch HC-101-3 selects one or both letdown throttle valves to accept a signal from the level control system. If both valves are in service, a letdown flow rate greater than 126 gpm is possible.

2.128 The PRESSURIZER LEVEL HI-LO CHANNEL X (Y) alarm actuates at +8 percent or -6 percent level deviation. High deviation starts the backup heaters, sends a backup signal to stop the standby charging pumps and increases letdown flow. The low setpoint starts all charging pumps and decreases letdown flow. The PRESSURIZER LEVEL LO-LO CHANNEL X (Y) alarm actuates at a level of 32% and de-energizes all pressurizer heaters.

2.129 There are four banks of backup heaters, and each bank is comprised of two or three groups of heaters. Each bank has a control switch on CB-1/2/3 with four positions:

- OFF, de-energizes the heaters.
- RESET, allows heater reset after a trip.
- AUTO, allows heater control by the selected pressure controller and level controller.
- ON, energizes the heaters.

2.130 The backup heater control switches must be placed in RESET momentarily:

- After power to the heaters is lost.
- To energize one group of heaters in each bank during 480 volt load shed, after a time delay, or to operate that group of heaters in each bank after the load shed signal has cleared (if that group was not operated during the load shed condition). The other one or two groups of heaters in each bank are locked out during 480

volt load shed and cannot be operated until the load shed signal has cleared, at which time they will return to their pre-load shed condition.

- To operate bank four backup heaters after the transfer switch is returned to REMOTE following LOCAL control at alternate shutdown panel AI-185.

2.131 Each backup heater control switch has four indicating lights. The green light indicates the heaters in that bank are de-energized. The amber light indicates that the control switch for that bank is in AUTO. The red light indicates that all groups of heaters in that bank are energized. The small red light indicates that only the group that is not locked out during 480 volt load shed is energized within that bank.

2.132 The ON/OFF control switches on MCC-4C1 for each group of bank four backup heaters control the heaters if the transfer switch on AI-185 is in LOCAL. This switch bypasses all normal control signals, including the low-low level heater cutout.

2.133 The proportional heater control switch on CB-1/2/3 has two positions: OFF and AUTO. The heaters remain de-energized when selected to OFF. AUTO energizes the heaters through the silicon controlled rectifier (SCR proportional controller).

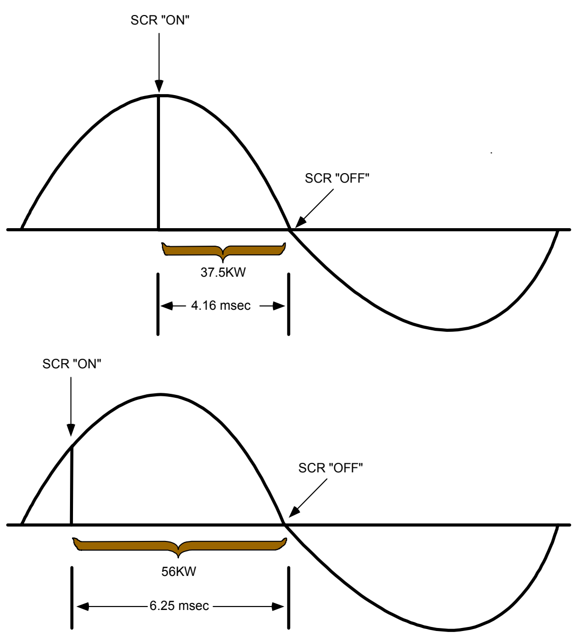


FIGURE 2-11: PROPORTIONAL HEATER SCR POWER TIMING

2.134 Two single-pen pressure recorder controllers, PRC-103X and PRC-103Y, are located on CB-1/2/3. Channel selector switch HC-103 selects which channel controls the heaters and spray valves. The setpoint, nominally 2100 psia, is adjustable by depressing the respective setpoint pushbutton (up/down) on each pressure controller. The selected controller generates an analog signal that modulates the proportional heater power and spray valve opening. A HI-LO PRESSURE alarm actuates at 2145 psia (HI) or 2080 psia (LO).

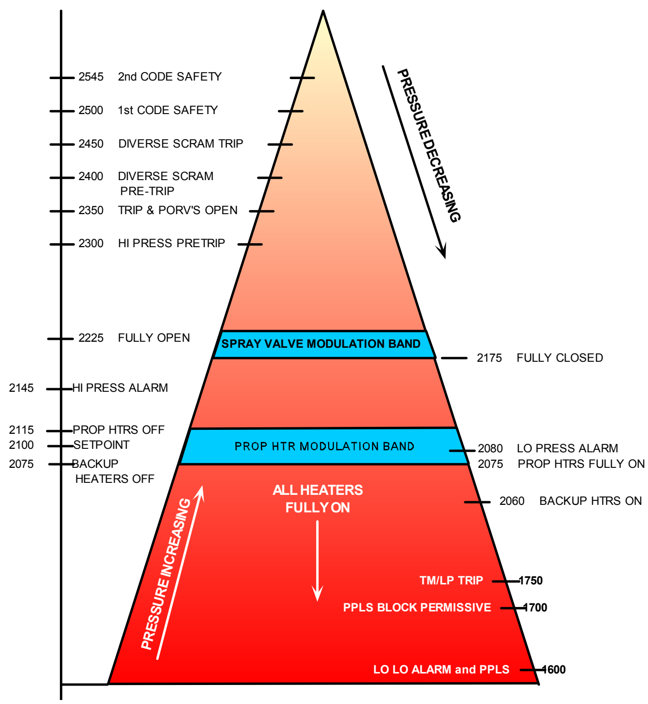


FIGURE 2-12: RCS PRESSURE CONTROL WITH 2100 PSIA SETPOINT

2.135 Low-range pressure channel P-118 provides Control Room indication on CB-1/2/3 from zero to 1600 psia. The low-range pressure channel actuates a close signal to the shutdown cooling suction valves if pressure exceeds 300 psia. P-118 must be energized to allow opening of HCV-347 and HCV-348. A loss of power override switch is provided to allow shutdown cooling operation during a loss of power from AI-40B. The redundant shutdown cooling valve interlock is provided by P-115, which requires power to provide the interlock, so no override switch is needed.

QUESTIONS REPORT
for ILO EXAM BANK

07-11-20 008

The reactor tripped 20 minutes ago. The following conditions are observed:

- "PRESSURIZER PRESSURE OFF NORMAL HI-LO" channel X and Y are in alarm.
- PRC-103X (controlling channel) indicates 2160 psia and stable
- All backup heaters in AUTO and energized
- LRC-101Y (controlling channel) indicates 60% and stable
- LRC-101X indicates 43% and increasing slowly
- LI-106 indicates 28%
- Letdown flow is 26 gpm
- One charging pump is running
- Tcold indicates 533°F, Thot indicates 534°F, both are stable

Select the probable cause and the action that should be taken to restore RCS pressure:

A. Low level on LRC-101X is maintaining the B/U heaters on, place the pressurizer heater cutout channel select switch in channel Y.

B. The bistable for the B/U heaters needs to be reset, place the control switches for all B/U heaters to reset and back to auto.

C.✓ LRC-101Y has malfunctioned causing the B/U heaters to remain on, place LRC-101X in service.

D. PRC 103X has malfunctioned causing the B/U heaters to remain on, place PRC-103Y in service.

QUESTION NUMBER: 010

The plant is operating at full power near the end of an operating cycle when a loss of feedwater without scram (ATWS) event occurs.

Which one of the following reactivity mechanisms causes a POSITIVE reactivity change during this event?

A.✔ Fuel temperature coefficient.

B. Moderator temperature coefficient.

C. Void coefficient.

D. Boron concentration change.

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 10 Rev 0

KA #: 000029 EK1.01 Tier 1 Group 1: Anticipated Transient Without Scram (ATWS)
Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior
Importance 2.8 / 3.1

CFR Number: 55.41(b)(1)
Fundamentals of Reactor Theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Fort Calhoun Objective:
EXPLAIN the primary and the secondary plant response to a loss of feedwater ATWS.

Question Pedigree
Bank question used in the 2007 NRC exam.

K/A Fit:
Question addresses reactivity change due to temperature changes during a loss of feedwater ATWS.

Choice A:
Correct answer. In a loss of feedwater ATWS, the moderator will heat up significantly providing negative reactivity feedback and a reduction in reactor power. The fuel temperature will decrease due to the power reduction causing a positive reactivity change.

Choice B:
Plausible because the change in moderator temperature will cause a reactivity change but incorrect because the reactivity change will be negative.

Choice C:
Plausible because a change in moderator voiding will cause a reactivity change but incorrect because the reactivity change will be negative.

Choice D:
Plausible, because an ATWS may result in automatic emergency boration due to safeguards actuation due to high containment pressure. Incorrect because the reactivity change will be negative.

KA#:	000029 EK1.01	Bank Ref #:	07-15-17 002
LP# / Objective:	0715-17 01.06	Exam Level:	RO-1
Cognitive Level:	HIGH	Source:	2007 NRC EXAM
Reference:	LP 0715-17	Handout:	NONE

OUTLINE OF INSTRUCTION	COMMENTS

II. A. 5. (continued)

The loss of feedwater without scram is the classic ATWS event since it produces the highest pressure in the RCS. Peak RCS pressure is allowed to exceed 2750 psia for this event and indeed it does!

A loss of feedwater without scram produces a very large mismatch between core heat production and secondary heat removal.

Primary temperatures and pressures increase rapidly. The pressurizer quickly fills with water and the PORVs and safety valves open and remain open for an extended period of time.

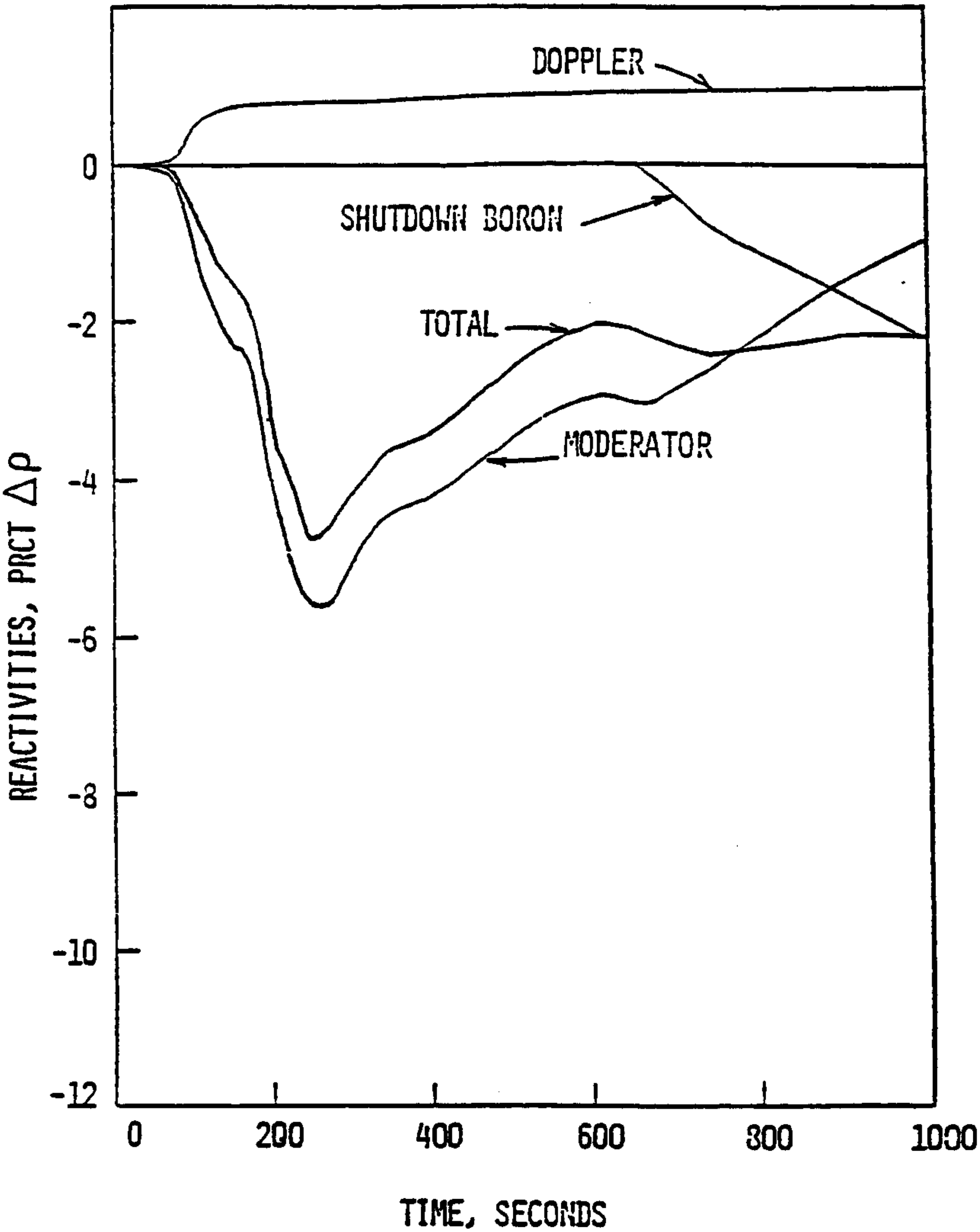
As the temperature of the water in the core increases, reactor power will decrease due to the negative MTC. The negative reactivity added by the MTC is partially offset by positive reactivity added by the doppler coefficient as the power decreases.

Eventually the power is reduced to the point that the rate of volumetric expansion of the primary fluid equals the volumetric flow out the PORVs and the pressurizer safety valves.

This is when the system pressure reaches its peak of over 4000 psia. Combustion Engineering has stated that at this pressure, the reactor vessel head bolts will stretch and water will be relieved between the reactor vessel and the head.

Figures 6 - 10 show the primary plant response to this event. Note that this is a generic CE analysis and is not specific to Fort Calhoun. — *Comments: Figures 6-10*

An excessive heat removal event occurring at End of Cycle will result in a greater reactivity addition than one at Beginning of cycle because the moderator temperature coefficient is more negative at EOC.



REACTIVITY COMPONENT TRANSIENTS DURING
3410 MWT COMPLETE LOSS OF FEEDWATER

Figure 10
07-15-17

QUESTION NUMBER: 011

Given the following plant conditions:

- The plant is operating at 20% power
- A high radiation alarm has been received on RM-054B
- Counts are rising on RM-057 but are below the alarm setpoint
- Charging flow is 40 gpm
- Letdown flow is 30 gpm
- Pressurizer pressure is 2080 psia and steady
- Pressurizer level is 48% and steady
- The are no containment radiation alarms
- Containment pressure is steady
- Containment sump level is steady

Which one of the following actions will occur automatically with these plant conditions?

A. Blowdown flow from Steam Generator, RC-2B will be isolated. Blowdown flow will continue from RC-2A.

B✓ Blowdown flow from both Steam Generators will be isolated.

C. RCV-978, 6th Stage Extraction Isolation Valve, will close.

D. A Ventilation Isolation Actuation Signal (VIAS) will be generated

Question # 11 Rev 0

KA #: 000038 2.4.31 Tier 1 Group 1: Steam Generator Tube Rupture
Knowledge of annunciator alarms, indications, or response procedures.
Importance 4.2 / 4.1

CFR Number: 55.41(b)(11)
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Fort Calhoun Objective:
EXPLAIN the response of secondary system parameters.

Question Pedigree
Bank question that was used on the 2009 NRC exam.

K/A Fit:
Question addresses expected system response to a high blowdown radiation alarm during a steam generator tube rupture.

Choice A:
Distractor: Plausible because high radiation on RM-054B will isolate blowdown from RC-2B. Incorrect because blowdown flow from RC-2A will also be isolated.

Choice B:
Correct answer: A high radiation alarm on RM-054B will close one containment isolation valve on each steam generator.

Choice C:
Distractor: Plausible because RCV-978 will close on high radiation. Incorrect because RCV-978 closes for a high radiation alarm on RM-057, condenser offgas, and RM-057 is below it's alarm setpoint

Choice D:
Plausible because several high radiation alarms result in a VIAS. Incorrect because RM-054B is not one of them.

KA#:	000038 2.4.31	Bank Ref #:	07-15-33 025
LP# / Objective:	0715-33 01.02	Exam Level:	RO-11
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM-33	Handout:	NONE

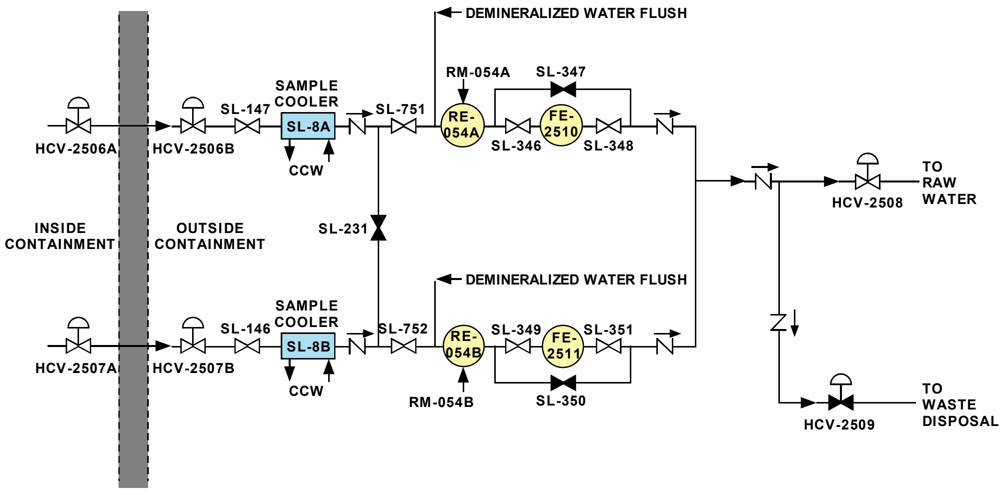


FIGURE 2-41: STEAM GENERATOR BLOWDOWN SAMPLING FLOW

••Location

2.246 The local ratemeters for both channels (ADM-610) are located just outside of Room 60 in Corridor 26.

••Power Supplies

2.247 Instrument power for RM-054A ratemeters is provided by 120 VAC instrument bus AI-40C.

2.248 Instrument power for RM-054B ratemeters is provided from 120 VAC instrument bus AI-40D.

••Instrumentation and Control

2.249 The local ratemeters for both channels (ADM-610), located just outside of Room 60 in Corridor 26, output to the Control Room ratemeters on AI-33A.

2.250 The Control Room ratemeters (ADM-600) provide an output signal to the ERF computer, radiation monitor recorder RR-049A, Control Room annunciator panel A33C and the four containment blowdown isolation valves.

2.251 A high alarm on RM-054A closes the inside containment blowdown isolation valves (HCV-1387A and HCV-1388A) for both steam generators. A high alarm on RM-054B closes the outside containment blowdown isolation valves (HCV-1387B and HCV-1388B) for both steam generators.

QUESTION NUMBER: 012

Given the following plant conditions:

- A large steam line break occurred inside containment
- PPLS, CPHS, SGLS and CSAS actuation all occurred
- Two containment spray pumps are running
- Containment pressure is 41 psig and lowering
- All containment cooling units and cooling and filtering units are operating

What actions are directed by EOP-05, "Uncontrolled Heat Extraction," and EOP/AOP Floating Step F, “Containment Spray Termination,” and why are these actions directed?

A.✓ Stop one containment spray pump to conserve water in the SIRWT.

B. Stop both containment spray pumps to minimize water damage to equipment inside containment.

C. Start a third containment spray pump to reduce containment pressure faster.

D. Continue to operate two containment spray pumps because containment pressure is still above 3 psig.

Question # 12 Rev 0

KA #: CE-E05 EK2.02 Tier 1 Group 1
Excess Steam Demand: Knowledge of the interrelations between the (Excess Steam Demand) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.
Importance 3.7 / 4.2

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
STATE from memory the condition(s) that must be met prior to termination of Containment Spray.

Question Pedigree
New question.

K/A Fit:
Question addresses operation of a heat removal system (containment spray) during an excessive steam demand event.

Choice A:
Correct answer: With containment pressure above 30 psig and below 60 psig, EOP-05 and Floating Step F direct the operators to ensure only one containment spray pump is operating. TDB-EOP-05 states that one of the objectives of this action is to reduce the demand on the SIRWT.

Choice B:
Plausible because both of the referenced procedures contain instructions for termination of all containment spray and a caution in floating step F states that containment spray may affect equipment inside containment. Incorrect because containment pressure must be less than 30 psig to stop all containment spray.

Choice C:
Plausible because floating step F directs starting a containment spray pump after spray termination if containment pressure can not be maintained below 40 psig. Incorrect because EOP-05 contains a caution that states "Do NOT run SI-3B and SI-3C at the same time."

Choice D:
Plausible because floating step H directs resetting the CPHS lockout relays when containment pressure is below 3 psig. Incorrect because floating step F directs stopping all containment spray when containment pressure is less than 30 psig.

KA#:	CE-E05 EK2.02	Bank Ref #:	N/A
LP# / Objective:	0718-15 03.09	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	FLOATING STEP F	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

Throughout the cooldown and depressurization, the operator should verify that the pressurizer pressure is being maintained within the maximum P-T limits. If the maximum Pressure-Temperature limits are being violated, then the operators should take actions to attempt to restore the RCS to within the P-T limits. The operator should perform the following actions as in such a situation:

- Operate main or auxiliary spray and letdown
- Throttle or stop HPSI pumps

Throttle or stop charging flow and manually control letdown flow.

NOTES

Stopping SI-3A or SI-3B will result in closure of one spray valve, HCV-344 or HCV-345 by interlock which will extend the time to RAS.

✖24. **IF** Containment Spray has been initiated
AND
ALL of the following conditions are satisfied:

- Two CS pumps are operating
- Containment pressure is less than 60 psig and **NOT** rising
- At least one VA-3A/B, Containment Air Cooler Filter System in service
- At least one VA-7C/D, Containment Air Cooler in service

THEN perform the following:

a. Ensure only **ONE** CS pump is operating.

b. Ensure only **ONE** of the following valves is open:

- HCV-344
- HCV-345

c. Ensure total CS flow is at least 2300 gpm.

24.1 **IF** Containment pressure can not be maintained less than 60 psig, **THEN** restore full CS flow by starting available CS Pumps, either SI-3A/B or SI-3A/C.

INSTRUCTIONS	CONTINGENCY ACTIONS
✗25. **IF** CS pump(s) are operating, **AND ALL** of the following conditions are satisfied: • Containment pressure is less than 30 psig and stable or lowering • Containment Spray is **NOT** required for Containment cooling • At least on VA-3A/B, Containment Air Cooler filter System in service • At least one VA-7C/D, Containment Air Cooler **THEN** stop Containment Spray by performing Floating Step F, Containment Spray Termination.	25.1 **IF** containment pressure can not be maintained less than 40 psig, **THEN** initiate Containment Spray by performing the following: a. Start **ONE** CS Pump: • SI-3A • SI-3B b. Open **ONE** Containment Spray Valve: • HCV-344 • HCV-345 c. Ensure total CS flow is at least 2300 gpm.

EPG Step: 23

Deviation
1) The EOP specifies containment design pressure is met. The EPG uses CSAS initiation pressure.
2) The EOP uses a two step approach to securing Containment Spray. The EPG utilizes a single step.
3) The EOP utilizes Floating Steps to perform Containment Spray Termination.
4) A note is used to warn operators of pump run out, The EPG does not have this note.

The note prior to step 24 is plant specific and is included to ensure the actuation of the interlock is not unexpected by the operators. This justifies deviation 3.

The intent of this step is to secure unneeded Containment Spray Pumps as early as practical after it has been confirmed that they have performed their safety function of pressure control. The overall objective is to:

- Reduce the demand on the SIRWT
- Reduce the flow rate to the sump during Containment recirculation
- Reduce the pressure differential across the sump screens if there is a buildup of debris

INSTRUCTIONS | CONTINGENCY ACTIONS

The Containment Spray System is actuated automatically to mitigate an Uncontrolled Heat Extraction by a combination of CSAS and SGLS. This automatic action is addressed earlier in this section, along with other steps being taken to mitigate a potential UHE. Once minimum safeguards are verified and Containment pressure is verified within design limits, one train of Spray is secured. The design basis for this even only requires one train of CACFS/CAC in-service with containment spray to mitigate the event. Analysis shows that the Containment Vent Fans will maintain Containment pressure and temperature control. Failure of half of the Containment Vent Fan trains does not challenge the Containment pressure and temperature safety functions. This step allows operations to control the spray system to conserve SIRWT water. This meets the intent of the EPG. This justifies deviation 1.

Containment Spray is used as a backup to the Containment Air Cooling and Filtering System for beyond-design basis failure of the CACFS during a LOCA. The steps to secure Containment Spray ensure the energy levels in containment are low enough not to cause equipment damage. The securing of Containment Spray will allow less wetting of containment components, and debris transport. Early termination is important to prevent RAS from initiating unnecessarily. The use of two steps for Containment Spray termination provides closer control of the Spray System during restoration. Containment Spray is secured after the affected Steam Generator has blown down and containment pressure has lowered below 30 psig in order to conserve SIRWT water and preclude a RAS condition. Containment pressure and temperature is expected to rise after Containment Spray termination; therefore, spray may have to be cycled on and off to control Containment pressure [(R10)]. This justifies deviation 2.

Floating steps are used to perform complicated actions that can be accomplished at any time during the event. The detail for restoration is placed in the floating steps to allow this. This justifies deviation 3.

Analysis shows that the Containment Vent Fans will maintain Containment pressure and temperature control. Failure of half of the Containment Vent Fan trains does not challenge the Containment pressure and temperature safety functions.

✘ 26. **IF** offsite power has been lost
AND resources permit,
THEN restore power to plant distribution and station loads by performing **ANY** or all of the following attachments:

- Attachments 17, Restoring Off-Site Power to Bus 1A3
- Attachment 18, Restoring Off-Site Power to Bus 1A4
- Attachment 21, Energizing Buses 1A1 and 1A2 from Off-Site Power

INSTRUCTIONS

✘ 5. (continued)

(continue)

CONTINGENCY ACTIONS

5.1 (continued)

3) **IF** two Containment Spray Pumps are operating, **THEN** open **BOTH** of the following Containment Spray Valves:

- HCV-344
- HCV-345

4) Ensure that total Containment Spray flow is greater than or equal to 2300 gpm.

****

CAUTION

Do **NOT** run SI-3B and SI-3C at the same time.

****

a.2 **IF** no Containment Spray Pumps are running **OR** total Containment Spray flow is less than 2300 gpm, **THEN** restore Containment Spray flow by performing the following:

(continue)

2.0 FLOATING STEPS

F. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTES

1. Stopping SI-3A or SI-3B will result in closure of one spray valve, HCV-344 or HCV-345 by interlock which will extend the time to RAS.

2. A small rise in Containment temperature and pressure is an expected response during the initiation of RAS.

****

CAUTION

Containment Spray may affect proper operation of RCPs, non-qualified equipment, Containment Sump, and instrumentation inside the Containment. When the termination criterion is satisfied, Containment Spray should be promptly secured.
****

1. **IF** Containment Spray has been initiated **AND** **ALL** of the following conditions are satisfied:

   - Two CS pumps are operating
   - Containment pressure is less than 60 psig and **NOT** rising
   - At least one VA-3A/B inservice
   - At least one VA-7C/D inservice

   THEN perform the following:

   a. Ensure only **ONE** CS pump is operating.

   b. Ensure only **ONE** of the following valves is open:

      - HCV-344
      - HCV-345

1.1 **IF** Containment pressure can not be maintained less than 60 psig, **THEN** restore full CS flow by starting available CS Pumps, either SI-3A/B or SI-3A/C

2.0 FLOATING STEPS

F. CONTAINMENT SPRAY TERMINATION

INSTRUCTIONS

c. Ensure total CS flow is at least 2300 gpm.

NOTE

Terminating Containment Spray prior to resetting actuation relays will require increased monitoring of containment parameters.

2. **IF** CS pump(s) are operating, **AND ALL** of the following conditions are satisfied:

- Containment pressure is less than 30 psig and stable or lowering
- Containment Spray is **NOT** required for Containment cooling
- At least one VA-3A/B inservice
- At least one VA-7C/D inservice

THEN terminate Containment Spray flow by performing the following:

a. Place the hand controller(s) for the open valve(s) in "OPEN":

- HIC-344
- HIC-345

b. Place all CS pumps, SI-3A/B/C in "PULL TO LOCK"

c. Close **BOTH** Containment Spray Valves:

- HCV-344
- HCV-345

CONTINGENCY ACTIONS

2.1 **IF** containment pressure can not be maintained less than 40 psig, **THEN** initiate Containment Spray by performing the following:

a. Start **ONE** CS Pump:

- SI-3A
- SI-3B

b. Open **ONE** Containment Spray Valve:

- HCV-344
- HCV-345

c. Ensure total CS flow is at least 2300 gpm.

c.1 **IF** either Containment Spray Valve fails to close, **THEN** manually close the open valve(s) using the handwheel (Room 59).

2.0 FLOATING STEPS

H. RESET OF ENGINEERED SAFEGUARDS

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

Do not perform this floating step if CSAS relays are tripped to prevent damage to VIAS relays.

****

2. **IF** CPHS has actuated
   AND Containment pressure is less than
   or equal to 3.0 psig,
   THEN reset CPHS by performing the
   following steps:

NOTE

Resetting CPHS Lockout Relays may reset SGIS. HCV-1105 and HCV-1106 may reopen.

a. Reset **BOTH** of the following relays:

- 86A/CPHS
- 86B/CPHS

b. Reset **BOTH** of the following relays:

- 86A1/CPHS
- 86B1/CPHS

(continue)

QUESTION NUMBER: 013

Given the following plant conditions:

- A loss of all feedwater event has occurred
- The conditions requiring Once-through-cooling have been met.
- All 4160 volt buses are energized
- All Reactor Coolant pumps have been tripped
- HCV-150 and HCV-151," PORV Block Valves," are open
- All pressurizer heaters have been deenergized
- PPLS has been initiated using the test switches
- All HPSI loop injection valves are open

Which one of the following describes the remaining steps required to establish once-through cooling per EOP-20, HR-4?

A. Ensure two HPSI pumps are running, open one PORV

B. Ensure three HPSI pumps are running, open one PORV

C.✓ Ensure two HPSI pumps are running, open both PORVs

D. Ensure three HPSI pumps are running, open both PORVs

Question # 13 Rev 0

KA #: 000054 2.1.20 Tier 1 Group 1: Loss of Main Feedwater
Ability to interpret and execute procedure steps.
Importance 4.6 / 4.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
EXPLAIN the process of once through cooling.

Question Pedigree
New question.

K/A Fit:
Question addresses executing procedure steps required to initiate once-through-cooling following a total loss of feedwater.

Choice A:
Distractor: Plausible because using only one PORV will conserve RCS inventory. Incorrect because two PORVs are needed for adequate heat removal.

Choice B:
Distractor: Plausible because using only one PORV will conserve RCS inventory. Incorrect because two PORVs are needed for adequate heat removal.

Choice C:
Correct answer. Two PORVs are needed for adequate heat removal and the procedures direct using two HPSI pumps.

Choice D:
Distractor: Plausible because two PORVs are needed for adequate heat removal and three HPSI pumps would increase injection flow. Incorrect because the procedures direct using two HPSI pumps.

KA#:	000054 2.1.20	Bank Ref #:	N/A
LP# / Objective:	0715-17 02.04	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	EOP-20. HR-4	Handout:	NONE

16.0 RCS AND CORE HEAT REMOVAL

SAFETY FUNCTION: RCS and Core Heat Removal

SUCCESS PATH: Once-Through-Cooling: HR-4

RESOURCE TREE: Tree E

HR-4

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTE

The action in this success path are intended to establish once-through-cooling in which both PORVs are open and at least two HPSI pumps are running. This is the desired mode of operation, however, other configurations may also be successful.

****

CAUTIONS

1. This is a heat removal path of last resort, it should only be utilized when S/G heat removal is not possible.

2. Opening PORVs may cause RCS pressure to drop.

3. Do not allow Diesel Generator loads to exceed power and current rating limits.

****

INSTRUCTIONS	CONTINGENCY ACTIONS
✗1. **IF** both Vital 4160 V Buses are energized, **THEN** establish once-through-cooling by performing the following: a. Stop all RCPs. (continue)	1.1 **IF** once-through-cooling can **NOT** be established because a Vital 4160 V bus is deenergized, **THEN** determine the appropriate method of recovery by performing step a, b, c or d: (continue)

16.0 RCS AND CORE HEAT REMOVAL

HR-4

INSTRUCTIONS	CONTINGENCY ACTIONS
✘1. (continued)	1.1 (continued)
b. Deenergize all PZR Heaters.	a. **IF** 1A3 is deenergized, **AND BOTH** of the PORV Block Valves are open: • HCV-150 • HCV-151 **THEN** GO TO Step 2.
c. Initiate PPLS by placing the following switches in “TEST”: • 86A/PPLS TEST SWITCH • 86B/PPLS TEST SWITCH	
d. Ensure **ALL** of the available pumps have started: • Either HPSI Pumps, SI-2A/B or SI-2B/C • Charging Pumps, CH-1A/B/C	b. **IF** 1A4 is deenergized, **AND BOTH** of the PORV Block Valves are open: • HCV-150 • HCV-151 **THEN** GO TO Step 12.
e. Ensure all of the HPSI Loop Injection Valves are open.	
f. Ensure **BOTH** of the PORV Block Valves are open: • HCV-150 • HCV-151	
(continue)	(continue)

16.0 RCS AND CORE HEAT REMOVAL

INSTRUCTIONS	CONTINGENCY ACTIONS
✖1. (continued)	1.1 (continued)
g. Open the PORVs. Time: ____________ h. GO TO Step 60.	**c. IF** 1A3 is deenergized, **AND ANY** of the PORV Block Valves are closed: • HCV-150 • HCV-151 **THEN** GO TO Step 21. d. **IF** 1A4 is deenergized, **AND ANY** of the PORV Block Valves are closed: • HCV-150 • HCV-151 **THEN** GO TO Step 42.

QUESTION NUMBER: 014

AC-3B is the running CCW pump when a Loss of Offsite Power occurs.
Diesel Generator D-1 fails to start.
Diesel Generator D-2 operates as designed.
The event does not result in SIAS actuation.

What operator action is required to restore CCW?

A. Verify AC-3B automatically starts.

B.✓ Manually start AC-3B

C. Verify AC-3A and AC-3C automatically start

D. Manually start AC-3A or AC-3C

Question # 14 Rev 0

KA #: 000056 AA1.09 Tier 1 Group 1: Loss of Off-Site Power
Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: CCW pump
Importance 3.3 / 3.3

CFR Number: 55.41(b)(8)
Components, capacity, and functions of emergency systems.

Fort Calhoun Objective:
EXPLAIN standby operation of CCW pumps in terms of switch positions and automatic actions.

Question Pedigree
Modified question. The source question was used on the 2002 NRC exam.

K/A Fit:
The question addresses required operation to restore a CCW pump following a loss of offsite power.

Choice A:
Distractor: Plausible because AC-3B was the running pump. Incorrect because the control switch will be in after start and will not receive an autostart signal.

Choice B:
Correct Answer: AC-3B will not receive an autostart signal and must be manually started. Note: Fort Calhoun sequencers will not operate without a SIAS actuation.

Choice C:
Distractor: Plausible because AC-3A and AC-3C would automatically start if AC-3B trips. Incorrect because there will not be any power to them with bus 1A3 deenergized due to the D-1 failure, unless 480 volt buses were manually crosstied.

Choice D:
Plausible because they could be started if there were power to bus 1A3. Incorrect because the stem states the D-1 failed to start.

KA#:	000056 AA1.09	Bank Ref #:	07-11-06 061
LP# / Objective:	0711-06 01.04	Exam Level:	RO-8
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	STM 8	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

3. IMPLEMENT the Emergency Plan.

Time: ____________

NOTE

Floating Step BB, Minimizing DC Loads, requires operator action within 15 minutes of loss of either battery charger.

✘ 4. Monitor the Floating Steps.

✘ 5. Verify either Vital 4160 V Bus is energized.

5.1 **IF** both Vital 4160 V Buses are deenergized,
THEN GO TO EOP-07, Station Blackout.

✘ 6. Attempt to start **ALL** of the following equipment:

- **ONE** CCW Pump, AC-3A/B/C
- **ONE** Raw Water Pump, AC-10A/B/C/D
- **ONE** Bearing Water Pump, AC-9A/B
- **ONE** Air Compressor, CA-1A/B/C
- **ONE** Charging Pump, CH-1A/B/C
- **ONE** AFW Pump, FW-6, FW-10, or FW-54

••Design/Specifications

2.4 CCW pumps are single-stage, horizontal centrifugal pumps. CCW pumps are driven by 250 hp motors powered from the 480 VAC buses. Flowrate with one pump running during normal operation is 5400 gpm. After a design basis accident, with two pumps assumed running, each pump has a design flow of 3750 gpm. In reality, all three pumps will start if available.

2.5 During normal operation, one pump provides approximately 150 ft. of total developed head (about 65 psid). With a nominal 40 psig nitrogen overpressure in the CCW surge tank, this results in a normal system pressure of approximately 90 psig. After a design basis accident, the three pumps provide approximately 200 ft. total developed head (about 86 psid).

2.6 CCW temperature is maintained between 70¬×F and 90¬×F. If raw water temperature is low, a portion of the CCW is bypassed around the heat exchangers, or CCW flow is established through a CCW heat exchanger with the raw water secured. After a design basis accident, CCW temperatures may be as high as 120¬×F.

2.7 Design pressure of the CCW pumps is 150 psig with a design temperature of 200¬×F.

••Location

2.8 The CCW pumps AC-3A/B/C are on the 1025 foot elevation of the Auxiliary Building at the east end of Room 69, just west of the surge tank.

••Power Supplies

2.9 The 480 VAC power supplies for the CCW pumps are as follows:

- AC-3A from bus 1B3B
- AC-3B from bus 1B4A
- AC-3C from bus 1B3C-4C

••Instrumentation and Control

2.10 The following is a detailed description of the Instrumentation and Controls for the Component Cooling Water Pumps.

•••Local

2.11 There are three breaker status lights (green, amber and red) at the respective motor control center (MCC) in the switchgear room for each CCW pump breaker. The status lights are located just above the 69-permissive control switch. The green and red lights indicate breaker position (green-open and red-closed), while the amber light indicates breaker availability. The amber light will be lit if control power is available to the closing circuit and the breaker is racked in.

2.12 All three breaker status lights will turn off if the 69-permissive switch at the MCC is placed in PULL OUT or if the breaker is racked out.

•••Remote

2.13 CCW pumps are operated from the control room on CB-1/2/3.

2.14 Each CCW pump has an ammeter, pistol-grip control switch and breaker status lights (green, white and red) on CB-1/2/3. Control switch positions are PULL TO LOCK, STOP and START. After operation, the control switch spring returns to the center position (AFTER STOP/START). The breaker status lights on CB-1/2/3 provide the following indications:

- Green light - is lit if the breaker is open and control power is available. The green light turns off if the 69-premissive switch is in PULL TO LOCK because it removes control power from the closing circuit. NOTE: control power will be energized at the breaker until the control power fuses are pulled.
- White light - is lit if there is disagreement between the control switch position and the local breaker position. For example, if the breaker trips open and the switch is in AFTER CLOSE.
- Red light - is lit if the breaker is closed. The red light is also used to verify trip circuit continuity during calibration testing of the breaker in the TEST position. This ensures that the trip circuit, if needed, will function properly.

2.15 The CCW pump is disabled by turning its control switch to the PULL-TO-LOCK position and pulling out (indenting) the handle. The PULL-TO-LOCK position actuates an alarm and turns off the breaker status lights at CB-1/2/3 and the respective MCC.

2.16 The CCW pump is aligned for standby operation (automatic start) by ensuring that the breaker is racked in, the 69-permissive switch is in AFTER CLOSE (red flag), the control switch on CB-1/2/3 is in AFTER STOP (green flag), and the green status light is lit.

•••Interlocks

2.17 The standby pumps automatically start if the running pump trips. Safeguards load sequencers start all CCW pumps.

2.18 Pressure switches PCS-412 and PCS-413 are located on the CCW pump discharge header in Room 69 and provide an interlock with containment isolation valves HCV-438A/B/C/D that supply CCW to the CEDM coolers and the Reactor Coolant Pump seal and lube oil coolers.

Figure 1-11 - 480 VAC Distribution

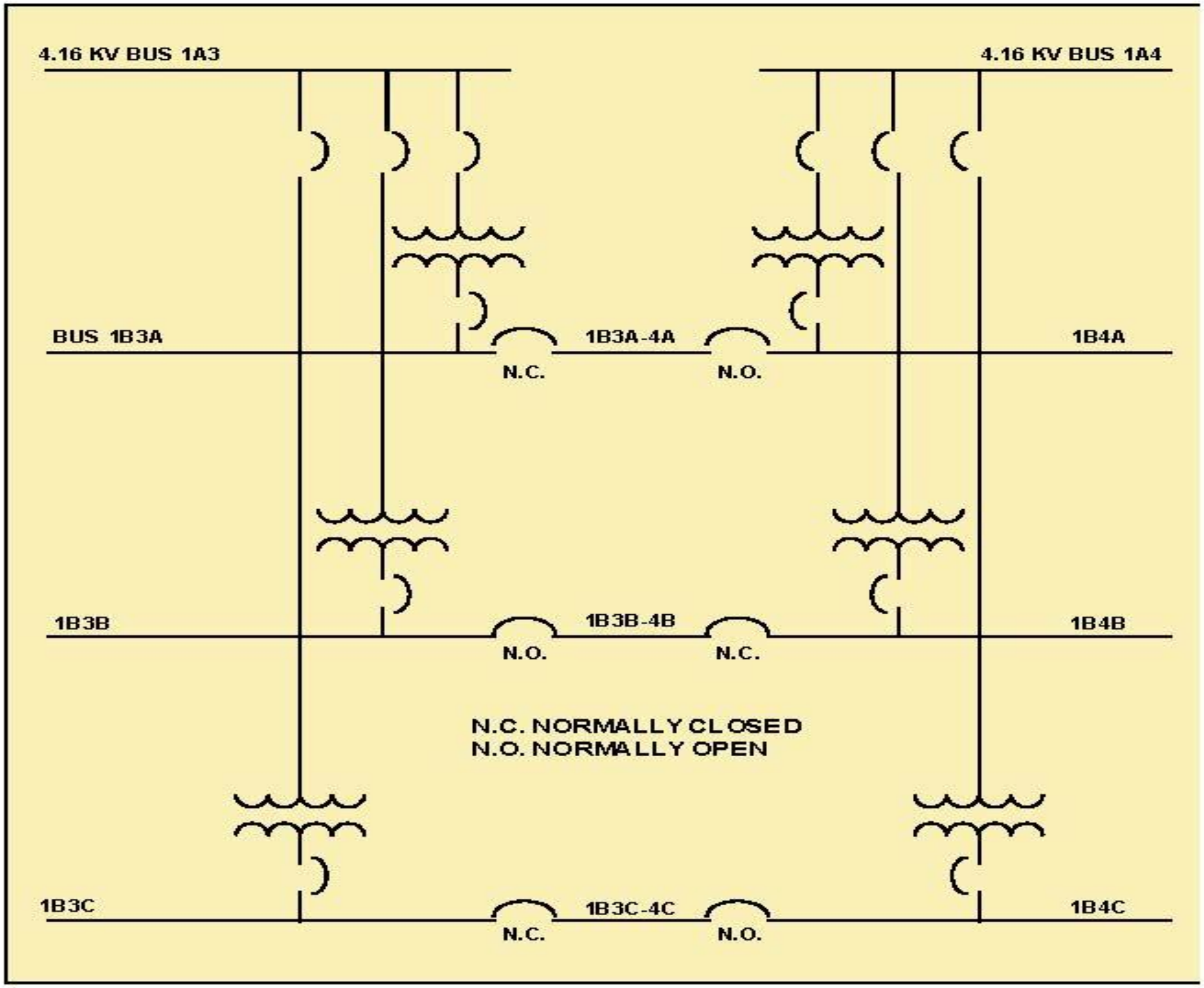
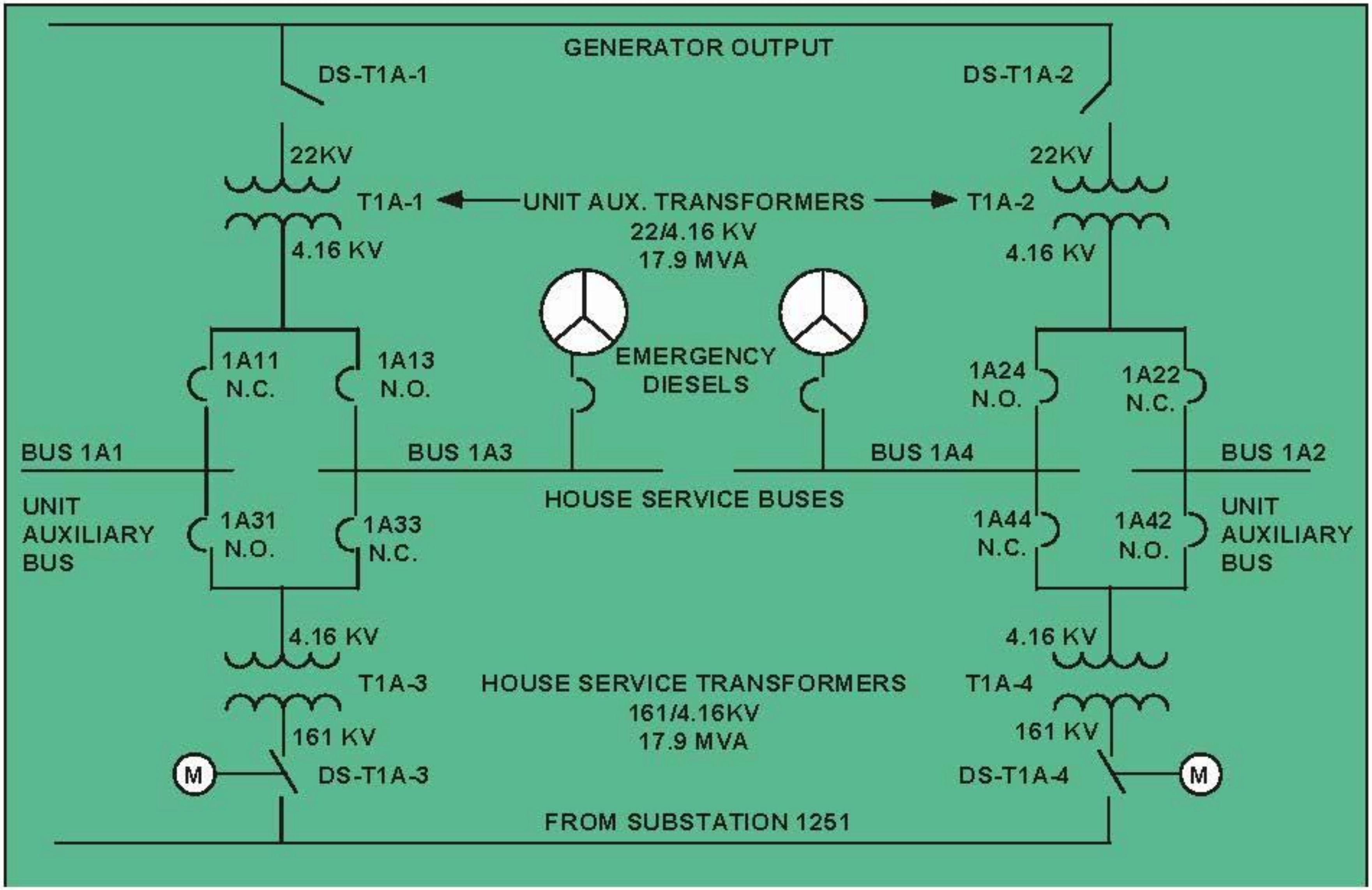


Figure 1-8 - Station Supply/4160 Distribution



QUESTIONS REPORT
for ILO EXAM BANK

07-11-06 061

AC-3B is the running CCW pump when a loss of offsite power occurs. Both Diesel generators operate as designed. The event does not result in SIAS actuation. Assuming no operator action, what will be the status of the CCW pumps 60 seconds after the diesels start?

A. Only AC-3B will be running

B. AC-3A and AC-3C will be running

C. All three CCW pumps will be running

D.✓ No CCW pumps will be running

QUESTION NUMBER: 015

Given the following plant conditions:

- The plant was operating at full power
- Power was lost to Instrument bus AI-40A
- AOP-16, LOSS OF INSTRUMENT BUS POWER, was entered

AOP-16, Section II, "Loss of Instrument Bus AI-40A", Step 14 states "Consider closing BOTH of the PORV (Power-Operated Relief Valves) Block Valves"

What is the purpose of this step?

A. An additional vital instrument bus failure will result in opening ONE of the PORVs.

B.✔ An additional vital instrument bus failure will result in opening BOTH of the PORVs.

C. An additional vital instrument bus failure will result in the inability to close ONE of the PORV Block Valves.

D. An additional vital instrument bus failure will result in the inability to close BOTH of the PORV Block Valves.

Question # 15 Rev 0

KA #: 000057 AK3.01 Tier 1 Group 1: Loss of Vital AC Electrical Instrument Bus
Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus
Importance 4.1 / 4.4

CFR Number 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
Describe how the plant responds to a loss of instrument bus power in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
Question addresses reason for procedural action taken due to a loss of an instrument bus.

Choice A:
Distractor: Plausible because Applicant could confuse with a failure of a pressure instrument while LTOP is in service which would result in opening one PORV. Incorrect because 2/4 logic opens both PORVs.

Choice B:
Correct answer:Deenergizing 2 vital instrument buses results in a 2/4 logic that opens both PORVs.

Choice C:
Distractor: Plausible if the Applicant does not fully understand the PORV logic. Incorrect because a loss of two vital instrument buses will not affect operation of the PORV block valves.

Choice D:
Distractor: Plausible if the Applicant does not fully understand the PORV logic. Incorrect because a loss of two vital instrument buses will not affect operation of the PORV block valves.

KA#:	000057 AK3.01	Bank Ref #:	07-17-16
LP# / Objective:	0717-16 01.02	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	TBD-AOP-16	Handout:	NONE

Section II - Loss of Instrument Bus AI-40A

INSTRUCTIONS	CONTINGENCY ACTIONS

12. Consider closing **BOTH** of the PORV Block Valves:

- HCV-150
- HCV-151

One additional channel trip is needed to actuate the PORVs. The operator is given a note explaining this fact and an instruction to consider closing the PORV Block Valves. This will alert the operator to the status of the PORV system.

NOTE

Upon loss of Instrument Bus A, **ALL** of the following instrumentation or equipment associated with the **Core Heat Removal Safety Function** are inoperable:

- "SUBCOOLED MARGIN MONITOR A-168"
- "RC LOOP TEMPERATURES LOOP 1A "T-COLD" A/TI-112C"
- "RC LOOP TEMPERATURES LOOP 1 "T-HOT" A/TI-112H"
- "RC LOOP TEMPERATURES LOOP 2A "T-COLD" A/TI-122C"
- "RC LOOP TEMPERATURES LOOP 2 "T-HOT" A/TI-122H"
- "SHTDN HT EXCH AC-4A OUTLET VALVE CNTRLR HCV-484"

13. Verify at least one RCP is running.

14.1 **IF** all RCPs are stopped **AND** SDC is in service, **THEN** perform the following:

a. **IF** AC-4A, Shutdown Heat Exchanger is in service, **THEN** perform the following:

1) Ensure HCV-485 is closed.

2) Place HCV-481 control switch in "OPEN".

3) Place HCV-480 control switch in "CLOSE".

4) Throttle HCV-485 to maintain RCS temperature.

- When the pretrip setpoint is reached, an annunciator will alarm on the main control board and a white light will be energized on the respective trip unit.
- When the trip setpoint is reached, a another annunciator will alarm on the main control board and a red light will be energized on the respective trip unit. The three red trip relay lights will be energized.

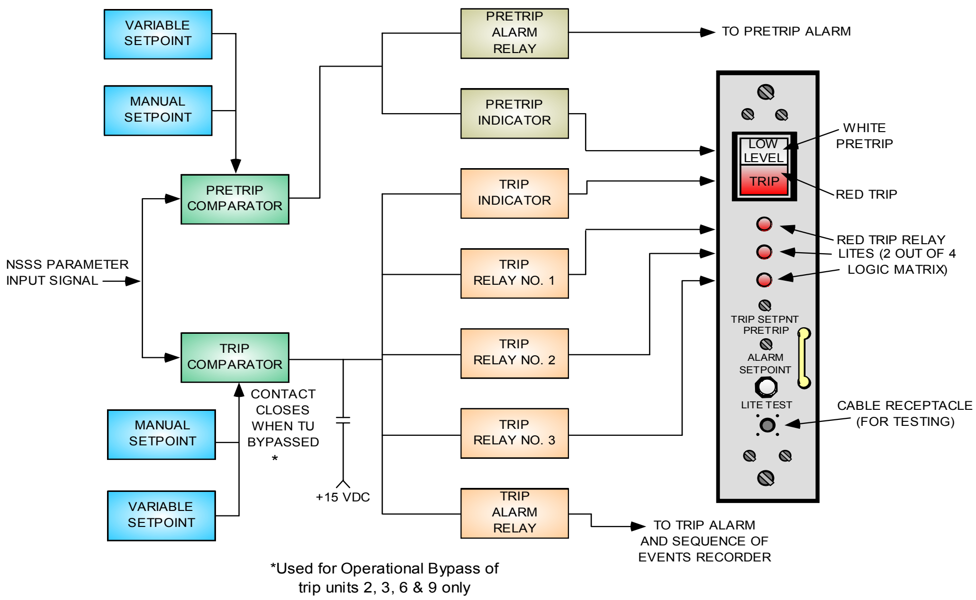


FIGURE 2-3: TYPICAL BISTABLE TRIP UNIT BLOCK DIAGRAM

••Failure Modes

2.27 The trip units are reliable, qualified components that are required to generate a reactor trip signal when parameters are exceeded, and prevent a reactor trip signal when spurious faults are experienced. The removal of any trip unit from the cubicle or loss of power to any trip unit will place that module in a tripped condition. Any one channel of each trip parameter may be placed in bypass for testing or maintenance.

High Power Level Trip Unit TU-1

2.28 The following is a detailed description of the high power level trip unit, TU-1.

••Function

2.29 The high power level trip protects the core against fast reactivity excursions that are too fast for the high pressure or TM/LP trips.

2.133 If a channel is put in BYPASS, it also bypasses PORV opening. For example, if "A" channel high pressure is bypassed, it would take 2 of the other 3 (B, C, D) high pressure channels going into trip to open the PORVs.

2.134 The high pressure bistable comparators have a fixed pretrip at 2300 psia and trip at 2350 psia.

•••Interlocks

2.135 Two out of four channels at 2350 psia initiates a reactor trip and opens the two pressurizer power operated relief valves.

•••Alarms and Indications

2.136 The HIGH PRESSURIZER PRESSURE CHANNEL PRETRIP alarm on CB-4 actuates at 2300 psia (1 of 4 channels). The HIGH PRESSURIZER PRESSURE CHANNEL TRIP alarm actuates at 2350 psia on CB-4. The REACTOR TRIP alarm on CB-4 will also actuate at 2350 psia (2 of 4 channels).

••Failure Modes

2.137 Loss of power to a channel produces a PORV open signal that cannot be bypassed, which results in a one-of-three logic for the remaining channels.

Thermal Margin/Low Pressure (TM/LP) Trip Unit TU-9

2.138 The following is a detailed description of the thermal margin/low pressure (TM/LP) trip unit, TU-9.

••Function

2.139 The TM/LP trip ensures operation is within the DNBR safety limit stated in the Technical Specifications. This limit states that reactor power level shall not exceed the allowable limit for a given pressurizer pressure and cold leg temperature as shown on Figure 1-1 in the Technical Specifications (repeated below as the Thermal Margin/Low Pressure Safety curves, Figure 2-17). This figure represents the values of reactor thermal power, RCS pressure, and cold leg temperature, where the value of DNBR is at the limiting value for the fuel.

2.140 To prevent exceeding the limits of Figure 2-17, a more restrictive set of curves is used for the TM/LP trip, as shown in the Thermal Margin/Low Pressure Limited Safety System Setting (LSSS) curve, shown in Figure 2-18. **NOTE**: TM/LP LSSS limits are subject to change after each refueling cycle. The TM/LP limits of Figure 2-18 are therefore only for illustration purposes. Refer to the COLR (TDB VI), Figure 1 for current values.

QUESTION NUMBER: 016

Why does AOP-16, Section VIII, "Loss of DC Bus 1," direct you to use HCV-1040, Atmospheric Dump Valve, for heat removal?

A. Control Power has been lost to MSIVs and MSIV bypass valves on both steam generators and they have have failed closed.

B. Control Power is lost to MS-291, MS-292 and the steam dump and bypass valves.

C.✓ Control Power is lost to MS-291 and the steam dump and bypass valves. MSIV, HCV-1042A, has failed closed.

D. Control Power is lost to MS-292 and the steam dump and bypass valves. MSIV, HCV-1041A, has failed closed.

Question # 16 Rev 0

KA #: 000058 AK3.02 Tier 1 Group 1:Loss of DC Power
Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Actions contained in EOP for loss of dc power
Importance 4.0 / 4.2

CFR Number 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
Describe the major recovery actions of this AOP.

Question Pedigree
New question.

K/A Fit:
Question addresses reason for steaming RC-2A using HCV-1040 following a loss of DC bus number 1 per AOP-16, Section VIII.

Choice A:
Distractor: Plausible if the Applicant believes both MSIVs will close on loss of DC bus 1 as they will on a loss of DC bus #2 and does not realize HCV-1040 is downstream of the MSIVs. Incorrect because MSIV HCV-1041A does not fail closed with a loss of DC bus #1.

Choice B:
Distractor: Plausible because Control power has been lost to MS-291 and the steam dump and bypass valves. Incorrect because control power is still available to MS-292.

Choice C:
Correct answer: Control power has been lost to MS-291 and the steam dump and bypass valves. S/G RC-2A can be steamed through HCV-1040 because MSIV HCV-1041A is still open.

Choice D:
Distractor: Plausible because control power will be lost to one S/G air operated safety valve, one MSIV and all steam dump and bypass valves. Incorrect because MS-292 and HCV-1041A still have control power.

KA#:	000058 AK3.02	Bank Ref #:	N/A
LP# / Objective:	0717-16 01.03	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	TBD-AOP-16	Handout:	NONE

Section VIII - Loss of DC Bus 1

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Upon loss of DC Bus 1, **ALL** of the following instrumentation or equipment associated with the **RCS Heat Removal Safety Function** is affected as follows:

- Control power to FW-4A, Feed Pump, is lost
- Control power to FW-5A, Heater Drain Pump, is lost
- Control power to FW-2A, Condensate Pump, is lost
- Control power to MS-291, Air Assisted Main Steam Safety Valve, is lost
- Power to Steam Dump and Bypass Valves is lost
- Control power is lost and SGIS closes HCV-1105, FCV-1101 Bypass Valve
- Control power to AFW Isolation Valves, HCV-1107A and HCV-1108A, is lost
- SGIS closes HCV-1042A, MSIV
- SGIS closes HCV-1103, Feed Reg Block Valves
- SGIS closes, HCV-1385, Feed Header Isolation Valve

21. Feed S/Gs with AFW to the AFW Nozzles to maintain S/G levels 35-85% NR (73-94% WR):

    a. a. Place the 43/FW Switch in "OFF".

21.1 **IF** a feedpath using AFW is **NOT** available,
THEN feed S/Gs to maintain S/G levels 35-85% NR (73-94% WR) by performing the following:

a. Verify at least one of the following are running:

- FW-4A/B/C
- FW-54

b. **IF** a SGIS has actuated,
THEN override SGIS by placing the following affected valve's SGIS Override Switches in "OVERRIDE":

- OR/HC-1386
- OR/HC-1385
- OR/HC-1105
- OR/HC-1106

c. Open the Feed Header Isolation Valves:

- HCV-1386
- HCV-1385

Section VIII - Loss of DC Bus 1

INSTRUCTIONS

b. Start at least one of the following AFW Pumps:

- FW-10
- FW-6

c. Verify HCV-1384, FW/AFW Header Cross-Connect Valve is closed.

d. Open **BOTH** of the following AFW Isolation Valves:

- HCV-1107A
- HCV-1108A

e. Manually control **BOTH** of the following AFW Isolation Valves:

- HCV-1107B
- HCV-1108B

CONTINGENCY ACTIONS

d. Manually control **EITHER** of the following valves:

- HCV-1105
- HCV-1106

e. **(LOCAL)** Manually control **EITHER** of the following Valves (Room 81):

- HCV-1105
- HCV-1106

With the loss of control power to normal feed paths to RC-2A the option to feed the Steam Generators through the Auxiliary Feedwater Nozzles is a preferred flow path. The valves HCV-1107A and HCV-1108A lose control power and fail open, this will require open verification using flow indication. Feedwater addition to both steam generators using the normal feed path is provided as a contingency, dependent on equipment availability. Local and remote operation of both valves is provided to allow flexibility in flow path selection.

The Steam Generator design and subsequent testing has removed the water hammer issue while feeding via the AFW nozzles. However, normal precautions should be used while feeding.

---

22. Steam RC-2A to control RCS temperature by performing the following:

a. Open MS-164, "MAIN STEAM LINE "A" STEAM DUMP TO ATMOSPHERE ISOLATION VALVE" (Room 81).

22.1 **IF** HCV-1040 is inoperable, **THEN** control RCS temperature using MS-292, Air Assisted Main Steam Safety Valve.

Section VIII - Loss of DC Bus 1

INSTRUCTIONS | CONTINGENCY ACTIONS

b. Ensure MS-407, "ATMOSPHERIC DUMP VALVE HCV-1040 2nd ISOLATION", is open (Room 81).

c. Open HCV-1040, Atmospheric Dump Valve.

Loss of control power to MS-291, PCV-910, and HIC-909 requires the use of HCV-1040 or MS-292 to control RCS Temperature.

NOTE

Upon loss of DC Bus 1, **ALL** of the following equipment associated with the **Containment Integrity Safety Function** is affected as follows:

- PCV-742E and PCV-742G, RM-050/051 Sample Valves, fail closed
- RM-050 and RM-051, Containment Atmosphere Radiation Monitors, are inoperable

23. Terminate all radioactive releases.

All radioactive releases are stopped until the status of effluent monitoring can be determined.

---

24. Ensure the RM-050/RM-051 Sample Pump is stopped.

RM-050/51 Sample Pump is verified stopped to prevent possible damage, since its sample lines are isolated.

---

25. Confirm Containment integrity by performing the following:

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

    b. Check for no Containment Area Radiation Monitor alarms.

QUESTION NUMBER: 017

What effect will a total loss of instrument air header pressure have on SIRWT level indication?

Assume that there is no actual change in SIRWT level and that the loss of pressure is of short enough duration that local air accumulators maintain pressure.

A. Low level will not be indicated, low level alarms will not be received, STLS actuation will not occur.

B.✔ Low level will be indicated, low level alarms will be received, STLS actuation will not occur.

C. Low level will be indicated, low level alarms will not be received, STLS actuation will not occur.

D. Low level will be indicated, low level alarms will be received, STLS actuation will occur.

Question # 17 Rev 0

KA #: 000065 AA2.08 Tier 1 Group 1: Loss of Instrument Air
Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment
Importance 2.9 / 3.3

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
Describe how the plant responds to a loss of instrument air in terms of how specific equipment is affected and how it affects overall plant operation and reliability.

Question Pedigree
Bank question used on 2005 NRC exam.

K/A Fit:
Question addresses failure modes of SIRWT air operated instrumentation (bubblers) following a loss of instrument air.

Choice A:
Distractor: Plausible if Applicant believes that all SIRWT level instruments have accumulators. Incorrect because only LC-383A/B/C/D, used for STLS actuation, have accumulators.

Choice B:
Correct answer. LT-381 and LT-382 provide level indication and alarm and do not have accumulators. LC-383A/B/C/D provide the STLS signal and they do have accumulators.

Choice C:
Distractor: Plausible if Applicant believes level indication and STLS actuation come from the same channels that are provided with accumulators. Incorrect because the level indication channels do not use accumulators.

Choice D:
Distractor: Plausible if the Applicant does not realize that the STLS instruments have accumulators. Incorrect because they do.

KA#:	000065 AA2.08	Bank Ref #:	07-17-17 002
LP# / Objective:	0717-17 01.02	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	BANK
Reference:	AOP-17	Handout:	NONE

Attachment B

Failure Position of Valves in Auxiliary Building - Controlled Area

NOTE

This attachment is not a complete listing of all air-operated valves in the controlled area of the Auxiliary Building.

1. CS, HPSI and LPSI Pumps suction and discharge valves all open.

2. SIRWT level indication (LT-381 and LT-382) and alarm (LIC-381 and LIC-382), in the Control Room are lost. An erroneous low level indication and alarm will result.

3. The SIRWT level controllers (LC-383A/B/C/D) which provide input to the RAS logic, are equipped with Air Accumulators and will continue to function for at least 12 hours.

4. Spent Fuel Pool level indication is lost.

5. Concentrated Boric Acid Tanks level indication is lost.

6. Blowdown Tank level control and indication are lost.

7. All Auxiliary Building ventilation dampers close with the exceptions listed below. These valves may be manually closed.

SI Pump Rooms	Inlet	HCV-804A
		HCV-805A
	Charcoal Filter Outlets	HCV-804C
		HCV-805C
Spent Regenerant Room	Inlet	HCV-803A
	Charcoal Filter Outlets	HCV-803C

8. The Vacuum Deaerator will overflow if DW-41 Pump is running.

QUESTION NUMBER: 018

Given the following plant conditions:

- The plant is operating at full power
- A grid disturbance has resulted in a loss of 161 KV to the plant
- Buses 1A3 and 1A4 fast transferred to 345 KV
- Main generator terminal voltage indicates 22,500 volts
- All Circulating Water Pumps are operating

Which one of the following actions is directed by AOP-31, “161 KV GRID MALFUNCTIONS,” section II, “All 4160 V Buses Fed From 22 KV”?

A.✓ Lower the Main generator terminal voltage using the voltage regulator

B. Raise the Main generator terminal voltage using the voltage regulator

C. Contact System Operations and request grid voltage be lowered

D. Contact System Operations and request grid voltage be raised

Question # 18 Rev 0

KA #: 000077 AA1.03 Tier 1 Group 1: Generator Voltage and Electric Grid Disturbances
Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Voltage regulator controls
Importance 3.8 / 3.7

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Describe the major recovery actions of this AOP.

Question Pedigree
New question.

K/A Fit:
Addresses operation of the main generator voltage regulator following a loss of 161 KV due to a grid disturbance.

Choice A:
Correct answer: AOP-31, Section II directs adjusting main generator terminal voltage less than 22,000 volts.

Choice B:
Distractor: Plausible because most electrical buses run at slightly higher voltages than designated. Incorrect because procedure directs lowering voltage below 22,000 volts.

Choice C:
Distractor: Plausible because AOP-31 section II contains direction to notify system operations and AOP-31, Section I does contain direction to request system operations adjust grid voltage for a 161 KV grid instability. Incorrect because procedure directs lowering main generator terminal voltage using the voltage regulator and 161 KV has been lost.

Choice D:
Distractor: Plausible because most electrical buses run at slightly higher voltages than designated, AOP-31 section II contains direction to notify system operations and AOP-31, Section I does contain direction to request system operations adjust grid voltage for a 161 KV grid instability. Incorrect because procedure directs lowering main generator terminal voltage using the voltage regulator and 161 KV has been lost.

KA#:	000077 AA1.03	Bank Ref #:	N/A
LP# / Objective:	0717-31 01.03	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	TBD-AOP-31	Handout:	NONE

Section II - All 4160 V Buses Fed From 22 KV

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS

1. **IF** Reactor power is greater than or equal to 50%,
   THEN ensure **ALL** of the following conditions are satisfied:
   - At least two Condensate Pumps are operating
   - At least two Feed Pumps are operating
   - At least two Heater Drain Pumps are operating

CONTINGENCY ACTIONS

1.1 **IF** Reactor power is less than 50%,
THEN align the Feedwater System by performing the following steps:

****

CAUTION

To protect Bus 1A1 in the event of a fault, FW-2A and FW-4A should not both be left running when the Feedwater System is realigned.

****

a. Ensure **ALL** of the following conditions are satisfied:
   - **ONE** Condensate Pump, FW-2A/B/C, is operating
   - **ONE** Feed Pump, FW-4A/B/C, is operating
   - **ONE** Heater Drain Pump, FW-5A/B/C, is operating

b. GO TO Step 4.

Section II - All 4160 V Buses Fed From 22 KV

INSTRUCTIONS

2. Adjust Main Generator terminal voltage less than 22,000 volts by performing the following steps:

   a. Notify System Operations of the need to adjust voltage.

   b. Adjust the Voltage Regulator.

   c. Verify terminal voltage is less than 22,000 volts.

CONTINGENCY ACTIONS

2.1 **IF** terminal voltage is greater than or equal to 22,000 volts,
AND plant conditions permit operation of only two Circ Water Pumps,
THEN stop **ONE** of the following Circ Water Pumps:

- CW-1B
- CW-1C

2.2 **IF** terminal voltage is greater than 22,500 volts,
THEN commence a power reduction to achieve less than 50% power within 24 hours PER OP-4, Load Change and Normal Power Operation.

Section I - 161 KV Grid Instability

INSTRUCTIONS	CONTINGENCY ACTIONS
10. **IF** 161 KV Grid voltage exceeds 168.6 KV **OR** a Urgent High voltage ERFCS alarm is received on **ANY** 4160 V bus, **THEN** verify voltages on 4160 V Buses are less than 4390 V.	10.1 **IF ANY** Bus voltages are greater than 4390 V for greater than one hour, **THEN** reduce voltage by performing any of the following steps: a. Contact System Operations and request Grid voltage be lowered. b. Start additional plant equipment powered from the affected 4160 V Buses.

NOTE

With high bus voltage, the operation of the LPSI and RCP motors above 4400 V for an extended period of time must be evaluated.

11. **IF ANY** 4160 V Bus voltage exceeds 4390 V for more than 24 hours, **THEN** contact System Engineering for bus load evaluation.	11.1 **IF** a plant shutdown is required, **THEN** commence a plant shutdown PER OP-4, Load Changes and Normal Power Operation. 11.2 **IF** plant conditions allow, **THEN** secure plant equipment as recommended by System Engineering.

12. GO TO Section 5, EXIT CONDITIONS.

End of Section 4.0

QUESTION NUMBER: 019

Given the following initial conditions:

- The plant was taken to hot shutdown during the middle of an operating cycle to repair feedwater heater tube leaks.
- The repair took 4 days
- The reactor was restarted and power raised to 8%
- The mode selector switch was in Manual Sequential
- The turbine was being warmed up

Current plant conditions:

- Reactor power level is 10% and increasing steadily
- RCS T-ave is increasing slowly
- Pressurizer level is following its programmed level
- Pressurizer pressure is slowly rising
- Containment pressure and temperature are normal

Which one of the following events will result in the current plant conditions?

A. A condenser bypass valve has failed closed

B.✔ A continuous CEA withdrawal is occurring

C. A turbine control valve is failing open

D. Reactor Xenon-135 concentration is changing

Question # 19 Rev 0

KA #: 000001 AA2.04 Tier 1 Group 2: Continuous Rod Withdrawal
Ability to determine and interpret the following as they apply to the Continuous Rod Withdrawal: Reactor power and its trend
Importance 4.2 / 4.3

CFR Number: 55.41(b)(1)
Fundamentals of Reactor Theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Fort Calhoun Objective:
A CEA withdrawal from full power

Question Pedigree
New question.

K/A Fit:
Question requires Applicant to diagnose a continuous rod withdrawal from reactor power and its trend and primary system response.

Choice A:
Distractor: Plausible because RCS temperature and pressurizer level and pressure will increase for a reduction in secondary heat removal such as a condenser bypass valve failing closed. Incorrect because a reduction in secondary heat removal will cause reactor power to lower.

Choice B:
Correct answer: A continuous rod withdrawal will cause reactor power, RCS temperature, pressurizer level and pressure to all increase.

Choice C:
Distractor: Plausible because a turbine control valve failing open would produce an increase in reactor power. Incorrect because it would cause RCS temperature, pressurizer pressure and pressurizer level to all decrease.

Choice D:
Distractor: Plausible because changes in xenon concentration can result in positive or negative reactivity changes depending on time after shutdown. Incorrect because the stem states that the repair took 4 days and the reactor will be xenon free.

KA#:	000001 AA2.04	Bank Ref #:	N/A
LP# / Objective:	0715-32 02.01	Exam Level:	RO-1
Cognitive Level:	HIGH	Source:	NEW
Reference:	LP 07-15-32	Handout:	NONE

OUTLINE OF INSTRUCTION	COMMENTS

II. C.

6. To make the analysis conservative, the event was assumed to occur at BOC with a slightly positive moderator temperature coefficient. The rods were assumed to be inserted to the PDIL. Initial power was assumed to be 102%. Credit was taken for only the variable overpower trip. — Used 112% setpoint

7. The power response is shown in figure 2. As we discussed in the theory section, the power increases in a linear manner until the overpower reactor trip setpoint is reached. It then falls off rapidly due to the reactor trip. The RCS temperature, pressurizer level and pressurizer pressure increase in response to the power increase, and then decrease rapidly due to the trip.

8. As shown in the USAR analysis; the DNBR stays above it's limit and the peak LHR and the peak RCS pressure stay below their limits.

9. At Fort Calhoun, the required operator response to a rod withdrawal event is covered by AOP-02. The operator is instructed to place the mode selector switch in the off position. If the rods continue to withdraw, the operator is then instructed to manually trip the reactor.

10. Once the reactor trip occurs, the event is over and EOP-00 is entered. If there are no further events and the control systems function as designed, all of the safety functions will be satisfied.

D. CEA Bank Withdrawal from a Zero Power Condition (Startup Accident)

1. This event is also discussed in section 14.2 of the USAR.

QUESTION NUMBER: 020

During refueling operations, a new fuel assembly was inserted into the core. The following counts were recorded prior to insertion of the assembly: (Channels A, B & C are connected to scaler-timers)

A = 118 cps B = 132 cps C = 112 cps D = 107 cps

After insertion of the assembly, the following counts were observed:

A = 240 cps B = 0 cps C = 225 cps D = 189 cps

Which one of the following actions should be taken next?

A.✓ Initiate Emergency Boration

B. Move the "B" scaler-timer to channel "D"

C. Withdraw the fuel assembly that was just loaded into the core

D. Continue loading fuel into the core

Question # 20 Rev 0

KA #: 000024 2.1.07 Tier 1 Group 2: Emergency Boration
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Importance 4.4 / 4.7

CFR Number: 55.41(b)(1)
Fundamentals of Reactor Theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Fort Calhoun Objective:
Describe the major recovery actions of this AOP.

Question Pedigree
Modified question. The source question was used in NRC exam 2001-2.

K/A Fit:
The question addresses the need to initiate emergency boration based on the response of nuclear instrumentation during refueling.

Choice A:
Correct answer: Channels A and C have doubled meeting the entry condition for AOP-03, Emergency Boration.

Choice B:
Distractor: Plausible because one of the channels with a scaler-timer has failed. Incorrect because emergency boration is required.

Choice C:
Distractor: Plausible because it would be reversing the reactivity change. Incorrect because emergency boration is required.

Choice D:
Distractor: Plausible If Applicant doesn't realize that counts have doubled on two channels. Incorrect because emergency boration is required.

KA#:	000024 2.1.07	Bank Ref #:	07-11-13 005
LP# / Objective:	0717-03 01.03	Exam Level:	RO-1
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	AOP-03	Handout:	NONE

1.0 PURPOSE

This procedure provides guidance in the event Emergency Boration of the RCS is required.

2.0 ENTRY CONDITIONS

The following conditions require the initiation of Emergency Boration using this procedure:

A. Uncontrolled Plant Cooldown.

B. Two or more CEAs inserted below the Emergency Boration Limit of the COLR Power Dependent Insertion Limit Curve.

C. Reactivity anomaly in a positive direction with all CEA's inserted.

D. Unexpected rise in count rate or unexpected doubling of count rate on two or more channels during refueling.

E. Failure of more than one Regulating or Shutdown CEA to insert following a Reactor trip.*

* Guidance for Emergency Boration with these conditions is provided in more appropriate procedures.

QUESTIONS REPORT
for ILO EXAM BANK

07-11-13 005

During refueling operations, a new fuel assembly was inserted into the core. The following counts were recorded prior to insertion of the assembly: (Channels A, B & C are connected to scaler-timers

A = 262 cps   B = 290 cps   C = 308 cps   D = 228 cps

After insertion of the assembly, the following counts were observed:

A = 270 cps   B = 0 cps   C = 312 cps   D = 232 cps

Can the next fuel assembly be inserted into the core?

A.✓ Yes, only two channels of counts are required to proceed.

B. Yes, but only with written permission from the Plant Manager.

C. No, the bundle should be withdrawn to observe the affect on countrate.

D. No, since the base counts were taken on three channels, a new base count should be taken on two channels.

QUESTION NUMBER: 021

Given the following conditions:

- Both level control channels (LC-101X and LC-101Y) are in cascade
- Level control channel selected to Channel Y
- Heater low level cutout selected to X/Y
- Pressurizer pressure is 2100 psia.

With no operator action, which of the following describes the response of the Pressurizer Level Control System if the reference leg of LI-101Y develops a leak?

A. Backup charging pumps (CH-1B and CH-1C) start, heaters energize, letdown flow decreases.

B. Backup charging pumps (CH-1B and CH-1C) start, heaters cutout, letdown flow decreases.

C.✓ Normal running charging pump continues to run, any running backup charging pumps stop, backup heaters energize, letdown flow increases.

D. Normal running charging pump stops, backup heaters do not change state, letdown flow increases.

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 21 Rev 0

KA #: 000028 AK1.01 Tier 1 Group 2: Pressurizer Level Control Malfunction
Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leak abnormalities
Importance 2.8 / 3.1

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective
EXPLAIN the interlocks and control functions associated with RCS Instrumentation.

Question Pedigree
Bank question used on 1999 NRC exam.

K/A Fit:
Question addresses how pressurizer level control responds to a reference leg leak

Choice A:
Distractor: Plausible if the Applicant believes level indication fails low and doesn't fully understand pressurizer heater operation. Incorrect because level fails high.

Choice B:
Distractor: Plausible if the Applicant believes level indication fails low. Incorrect because level fails high.

Choice C:
Correct answer. Pressurizer level indication fails high causing backup charging pumps to start and letdown flow to increase. Backup heaters energize due to high level deviation.

Choice D:
Distractor: Plausible because pressurizer level indication does fail high causing letdown flow to increase. Incorrect because one charging pump will continue to operate and heater will energize due to high level deviation.

KA#:	000028 AK1.01	Bank Ref #:	07-11-20 025
LP# / Objective:	0711-20 04.04	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	BANK
Reference:	GFE C07	Handout:	NONE

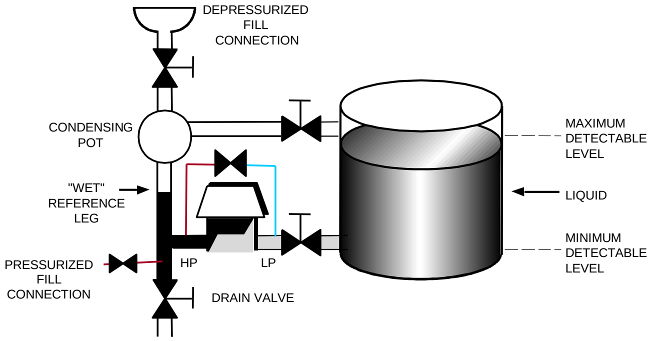


Figure 7-22 Closed Tank with Reference Leg Flashing

ENVIRONMENTAL EFFECTS ON OPERATION

As previously discussed, variations in temperature and pressure can affect the reference leg density and the accuracy of the level measurement. These same density variations can be caused by the ambient conditions to which the level instruments are exposed, specifically the containment building.

Level measuring differential pressure cells are not directly affected by changes in ambient pressure. Any pressure variations in the environment are felt on both sides of the open vessel D/P level instrument and cancel each other. The dry and wet reference leg level detectors are only exposed to system pressure and therefore are not affected.

Temperature variations can significantly affect the level measurement accuracies. An increase in ambient temperature will cause the density of the wet reference leg to decrease. This will result in a lower D/P sensed by the D/P cell, and indicated level will be greater than actual level. The opposite effect produces a lower indicated level when the ambient temperature decreases.

As previously discussed, radiation levels near the D/P cell affect the detector integrity. A high radiation environment can permanently embrittle the detector cell, causing the cell to lose its elasticity and altering its characteristics, as well as degrade sensitive electronics.

FAILURE INDICATIONS

For level detectors with a wet reference leg connected to the "high pressure" side of a D/P cell, the following failure modes exist. A break in the variable leg of the D/P cell creates a higher D/P being sensed by the D/P cell, resulting in the level instrument indicating a low level. Conversely, a reference leg break creates a lower D/P sensed across the D/P cell, resulting in an indicated level higher than actual level. A ruptured diaphragm or an open equalizing valve in a D/P cell allows the high pressure and low pressure chambers to equalize and produces a minimum D/P signal. The minimum D/P signal will result in a maximum level indication. Reference leg inventory losses usually occur during a rapid pressure decrease, resulting in flashing part of the reference leg water to steam. When this occurs, the pressure on the reference leg side of the D/P cell decreases, causing the D/P to decrease and indicated level to increase. For applications where the dry reference leg is connected to the "low pressure" side of the D/P cell, the failure modes will be directly opposite of those described above for ΔP. See Table 7-3 for failure indications.

•••Fail Position on Loss of Air or Power

2.94 Subcooled margin monitor SMM-A (B) is rendered inoperable on a loss of instrument bus AI-40A (B) or DC bus No.1 (No. 2).

Pressurizer Level, Temperature, and Pressure

2.95 The following is a detailed description of the pressurizer level, temperature, and pressure instrumentation.

••Function

2.96 Pressurizer level instrumentation functions to maintain pressurizer level at the programmed setpoint by regulating the letdown flow rate and starting or stopping the backup charging pumps, as required.

2.97 Pressurizer pressure instrumentation is used to:

- Maintain pressure at 2100 psia during normal operation.
- Energize heaters during a decreasing pressure transient.
- De-energize heaters and open spray valves during an increasing pressure transient.
- Generate alarms to warn of abnormal pressures.
- Supply pressure signals to the RPS, ESC, SMM, RRS, DSS, QSPDS, and the ERF computer.
- Provide overpressure protection during cooldown operation.

••Design/Specification

2.98 The Pressurizer Level Control System is designed to maintain pressurizer level between 48 percent at 535°F to 60 percent at 560°F. The Pressurizer Pressure Control System is designed to maintain pressurizer pressure at 2100 psia.

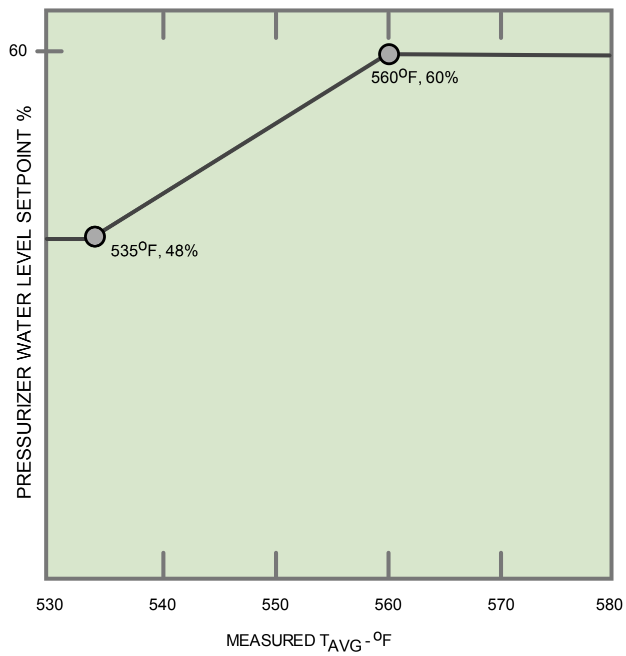


FIGURE 2-6: PRESSURIZER LEVEL SETPOINT

2.99 The pressurizer level sensors are differential pressure (ΔP) cells. The sensors measure the pressure difference between a reference leg and the variable leg, which corresponds to pressurizer level. The reference leg is maintained full by a condensing pot exposed to containment ambient temperature.

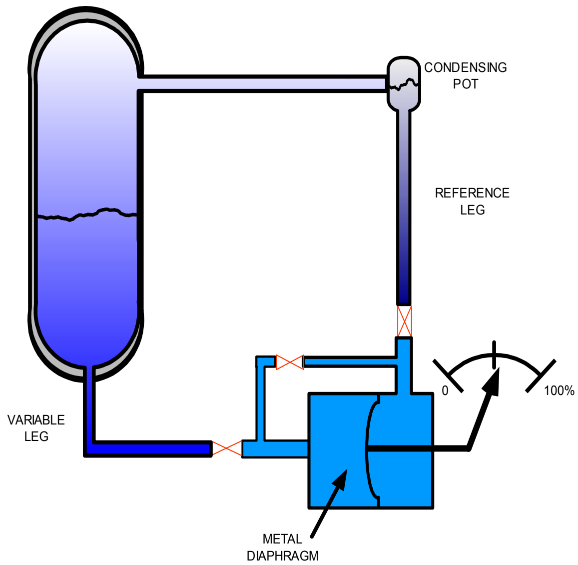
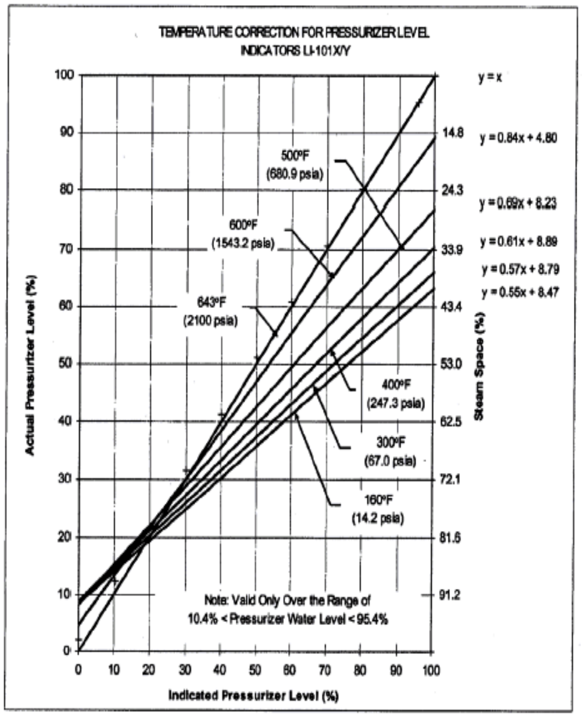


FIGURE 2-7: DIFFERENTIAL PRESSURE LEVEL SENSOR

2.100 Pressurizer level control channels LI-101X and LI-101Y are calibrated for 643°F. LI-106 is calibrated for 120°F. Temperature correction curves are used to determine actual level if the pressurizer is not at the calibration temperature. Reference leg temperature (which is determined by containment atmosphere temperature) also affects indicated pressurizer level. Therefore, a family of curves is provided, based on containment atmosphere temperature, to correlate indicated pressurizer level to actual level. Figure 2-8 shows a typical pressurizer level correction curve. If pressurizer temperature is below the calibration temperature, indicated level is higher than actual level.



Commitment ID # 790101

FIGURE 2-8: LI-101 X & Y TEMPERATURE CORRECTION (TDB.111.1.A)

2.101 The control channels function to maintain pressurizer level at the setpoint. The setpoint is developed by the RRS. RRS A provides the setpoint for channel X, and RRS B provides the setpoint for channel Y. The level control functions respond to deviation from setpoint. **NOTE**: The T_{ave} signal from the RRS is hard wired and, therefore, not selectable with the A/B selector switch on CB-4.

2.102 There are six control channels of pressurizer pressure: four are safety channels and two are control channels. The control channels provide input to the ERF computer and pressurizer heater controls (automatic control rod positioning has been disabled). The safety channels provide input for RPS trips (high-pressure and TM/LP) and for PPLS. The only instrumentation on the pressurizer that is safety related is the pressurizer pressure instrumentation.

2.103 The four pressurizer pressure safety channels, A/B/C/D/PT-102, each have a transmitter and amplifier in a current loop with three voltage dropping resistors. The pressurizer pressure sigma meters have an internal voltage dropping resistor. The voltage dropped across the resistors is used for the high pressurizer pressure trip, TM/LP trip and PPLS.

2.104 The safety channels, A/B/C/D/PT-120, provide input for the Diverse Scram System (DSS). The pretrip is set at 2400 psia, and the trip is set at 2450 psia. The instrument range is from 1900 to 2900 psia. A/PI-120 provides pressure indication on AI-66A, and B/PI-120 provides pressure indication on AI-66B.

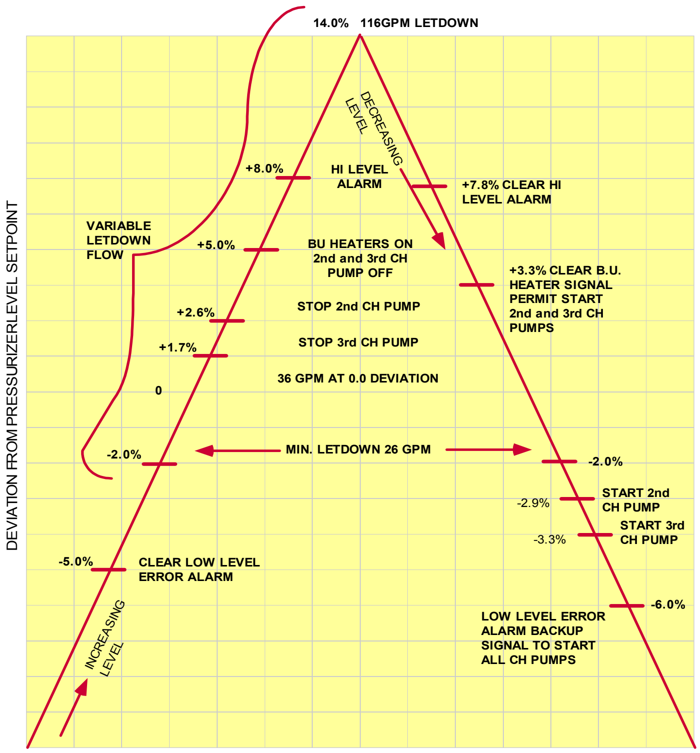


FIGURE 2-9: LEVEL CONTROL CHRISTMAS TREE

2.105 Safety channels A/B/C/D/PT-102 provide the following interlocks/controls. At 1700 psia decreasing, the pressurizer pressure safety channels allow a block of the PPLS signal. A PPLS is generated by safety channels at 1600 psia decreasing if not previously bypassed and actuates the LO-LO PRESSURE alarm. Safety channels also provide a TM/LP trip at a variable setpoint, a HIGH PRESSURE PRETRIP alarm at 2300 psia, and a high pressure trip and PORV actuation at 2350 psia.

2.106 Each safety channel pressure (A/B/C/D/PIA-102) is displayed on CB-1/2/3 from 1500 to 2500 psia. A signal from each channel is supplied to the computer. Meters A/B/C/D/PIA-102X display the TM/LP setpoints generated by the RPS.

QUESTION NUMBER: 022

During a large steam line break event, the CVCS is designed to initially supply _____ borated water to the RCS from the _____ for reactivity control.

Assume all systems work as designed.

A. 80 gpm, Boric Acid Storage Tanks

B.✓ 120 gpm, Boric Acid Storage Tanks

C. 80 gpm, Safety Injection and Refueling Water Storage Tank

D. 120 gpm, Safety Injection and Refueling Water Storage Tank

Question # 22 Rev 0

KA #: CE-A11 AK1.01 Tier 1 Group 2: RCS Overcooling
Knowledge of the operational implications of the following concepts as they apply to the (RCS Overcooling) Components, capacity, and function of emergency systems.
Importance 3.1 / 3.3

CFR Number: 55.41(b)(8)
Components, capacity, and functions of emergency systems.

Fort Calhoun Objective:
EXPLAIN the response of the CVCS to signals from the Engineered Safeguards Control System.

Question Pedigree
New question

K/A Fit:
Question addresses the charging flow capacity and function of the Boric Acid Storage Tanks following safeguards initiation due to an overcooling event (steam line break).

Choice A:
Distractor: Plausible if the Applicant believes that 2 of 3 charging pumps start on a SIAS just like 2 of 3 HPSI pumps start. Incorrect because all three charging pumps start.

Choice B:
Correct answer. SIAS starts all 3 charging pumps and opens valves from the suction of the Boric Acid Storage tanks to the suction of the charging pumps.

Choice C:
Distractor: Plausible because the SIRWT can be aligned to the suction of the charging pumps and the Applicant believes that 2 of 3 charging pumps start on a SIAS just like 2 of 3 HPSI pumps start. Incorrect because the initial supply of borated water is from the Boric Acid Storage Tank.

Choice D:
Distractor: Plausible because the SIRWT can be aligned to the suction of the charging pumps. Incorrect because the initial supply of borated water is from the Boric Acid Storage Tank.

KA#:	CE-A11 AK1.01	Bank Ref #:	N/A
LP# / Objective:	0711-02 01.04	Exam Level:	RO-8
Cognitive Level:	LOW	Source:	NEW
Reference:	EOP-05	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

✗ 6. **IF** RCS pressure is less than or equal to 1600 psia,
THEN verify Engineered Safeguards are actuated by performing the following steps:

a. Ensure Emergency Boration is in progress.

b. Ensure SI flow is acceptable PER Attachment 3, Safety Injection Flow vs. Pressurizer Pressure.

EPG Step: 4, 5, 21a

Deviation:
1) The EOP step anticipates automatic containment isolation based on a low RCS pressure while the EPG step bases it on CIAS.
2) The EOP requires Emergency Boration in progress, the EPG references the starting of the Charging Pumps.
3) The EPG provides specific guidance in the step for restoration, the EOP utilizes a floating step.

Plant specific Engineered Safeguards initiating signals are checked. Plant specific SIAS is actuated by either a Pressurizer Pressure Low Signal (PPLS) at 1600 psia or a Containment Pressure High Signal (CPHS) at 5 psig. Many functions performed by the EPG SIAS are directly or indirectly performed by the plant specific PPLS or CPHS signals. EPG SIAS may be referred to in the EOP as PPLS or SIAS.

The plant specific automatic Containment Isolation circuitry is designed to anticipate that a low primary pressure is the result of a primary system break inside containment (i.e., a primary break that will result in a direct release to the containment atmosphere). Hence, low RCS pressure due to a UHE will result in a Containment Isolation. This justifies deviation 1.

Emergency Boration provides the admission of concentrated Boric Acid directly to the suction of the charging pumps via forced flow or gravity feed from the concentrated Boric Acid Storage Tanks. This addition is prudent to counteract the positive reactivity inserted during a UHE due to the lowering moderator temperature. It is therefore prudent to ensure Emergency Boration is taking place. This justifies deviation 2.

Floating steps are used to perform complicated actions that can be accomplished at any time during the event. The detail for restoration is placed in the floating steps to allow this. This justifies deviation 3.

2.351 The "Second" standby charging pump will auto start when pressurizer level decreases to 3.3% below programmed level as long as the following conditions are met:

- The charging pump has been selected as Standby via the mode selector switch.
- The charging pump control switch is in the AFTER-STOP position.

2.352 The "Second" standby charging pump will auto stop when pressurizer level increases to 1.7% above programmed level.

2.353 The "First" standby charging pump will auto stop when pressurizer level increases to 2.6% above programmed level.

2.354 In the event pressurizer level decreases to 6% below programmed level, all charging pumps that are not currently running will receive a start signal regardless of their standby status. Once pressurizer level increases to 5% above programmed level, the standby charging pumps will stop. The charging pump that was previously running will be unaffected.

2.355 The charging pumps will also start on a Pressurizer Pressure Low Signal (PPLS) or Containment Pressure High Signal (CPHS). This will cause the auto starts due to pressurizer level to be blocked.

•••Alarms and Indications

2.356 The following charging pump alarms are located on CB-1/2/3:

- The CHARGING PUMP LUBE OIL PRESSURE LO alarm actuates via pressure control switch PCS-234 (-237, -240) if lube oil pump discharge pressure is less than 9 psig and the respective charging pump motor supply breaker is closed.
- The CHARGING FLOW LO alarm actuates via FIA-236 if the charging flow is less than 30 gpm, which could be indicative of a charging pump trip.
- The CHARGING PUMPS TRIP alarm is a common alarm for all three pumps that actuates via a low lube oil pressure control switch (9 psig), a low suction pressure control switch (10 psia), a bus undervoltage, or a breaker overcurrent.

2.357 The following charging pump alarms are located on AI-30A(B):

- The CHARGING PUMP CH-1A (-1B or -1C) OFF NORMAL alarm actuates if the respective control switch on CB-1/2/3 is placed in the PULL-TO-LOCK position, the "69" switch is not in the AFTER-CLOSE position, or the pump breaker is not in the racked in position.
- The CHARGING PUMP CH-1A (-1B or -1C) AUTO START FAIL alarm actuates if the respective charging pump breaker fails to close within one second of receiving the signal to close

QUESTION NUMBER: 023

The following plant conditions exist:
- A plant startup is in progress
- The plant is in MODE 2
- Steam Dumps/Turbine Bypass Valves are controlling RCS Temperature
- A maintenance error results in all running Circulating Water Pumps being stopped

Which of the following describes the expected response of the Steam Dumps and Turbine Bypass Valves with no operator actions?

A. Any open valves will close immediately.

B✓ Any open valves will close when Condenser Vacuum degrades to 19 inches.

C. The Steam Dumps will close immediately; the Turbine Bypass Valve will close when Condenser Vacuum degrades to 19 inches.

D. The Turbine Bypass Valve will close immediately; the Steam Dumps will close when Condenser Vacuum degrades to 19 inches.

Question # 23 Rev 0

KA #: 000051 AK3.01 Tier 1 Group 2: Loss of Condenser Vacuum
Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum
Importance 2.8 / 3.1

CFR Number: 55.41(b)(4)
Secondary coolant and auxiliary systems that affect the facility.

Fort Calhoun Objective
EXPLAIN the actions necessary to control main steam pressure using the steam dump and turbine bypass valves if the automatic pressure and temperature control outputs from the RRS are not available.

Question Pedigree
Bank question used on 1995 NRC exam.

K/A Fit:
Question address steam dump and bypass valve response to a loss of condenser vacuum.

Choice A:
Distractor: Plausible if Applicant believes there is an interlock between circulating water pumps and steam dump and bypass valves. Incorrect because there is no interlock.

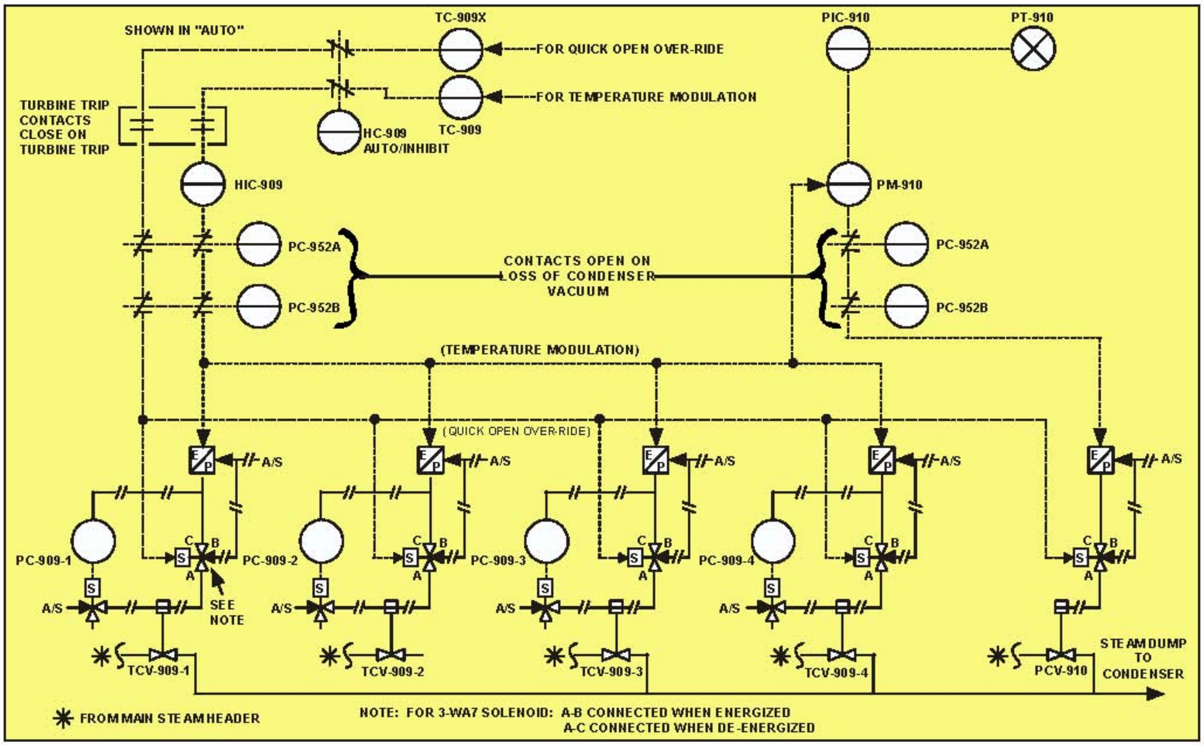
Choice B:
Correct answer. Steam dump and bypass valves will close when condenser vacuum degrades to 19 inches.

Choice C:
Distractor:Plausible combination of choices A and B. Incorrect because any open valve will close when condenser vacuum degrades to 19 inches.

Choice D:
Distractor:Plausible combination of choices A and B. Incorrect because any open valve will close when condenser vacuum degrades to 19 inches.

KA#:	000051 AK3.01	Bank Ref #:	07-12-31 001
LP# / Objective:	0712.-31 02.02	Exam Level:	RO-4
Cognitive Level:	LOW	Source:	BANK
Reference:	LP 0712.31	Handout:	NONE

Figure 2-24 - Turbine Dump and Bypass Valve Control



2.5.5Ab (continued)

9) Turbine bypass valve signal auctioneering unit pressure control setpoint may be adjusted on PIC-910 using the setpoint controller and is variable from 800 psia to 1000 psia. It generates an electrical pressure error signal that is proportional to the difference between the main steam pressure and the established pressure setpoint. The pressure error signal is supplied as one input to the signal auctioneering unit. The second input (programmed Tave from the RRS) is applied to the auctioneering unit through the main turbine trip contacts when the main turbine is tripped.

B. Interlocks

1. The auto signal from the RRS is overridden by a signal from the condenser on low vacuum. This signal prevents dump valves from opening in either MANUAL or AUTO if condenser vacuum drops below 19 inches of mercury.

2. The dump valves will not open on a reactor trip if RCS average temperature is less than 535°F. The opening signal from the RRS is overridden if auto-inhibit switch HC-909 is placed in INHIBIT.

QUESTION NUMBER: 024

With the plant in Mode 3, the following conditions exist:

- Circulating Water Pump, CW-1A is running
- Circulating Water Pumps, CW-1B & 1C are secured
- Raw water Pump, AC-10A is operating
- Waste Monitor Tank, WD-22B, is being released
- RM-055 is being used to monitor the release

If CW-1A trips, which of the following actions should be taken immediately to ensure the offsite doses determined by the release permit are not exceeded?

A. An additional Raw Water Pump should be started.

B. The RM-055 setpoint should be lowered per the release permit.

C.✓ The Waste Monitor Tank Release should be terminated.

D. The Waste Monitor Tank should be re-sampled.

Question # 24 Rev 0

KA #: 000059 AK3.04 Tier 1 Group 2: Accidental Liquid Radwaste Release
Knowledge of the reasons for the following responses as they apply to the Accidental Liquid Radwaste Release: Actions contained in EOP for accidental liquid radioactive-waste release
Importance 3.8 / 4.3

CFR Number: 55.41(b)(13)
Procedures and equipment available for handling and disposal of radioactive materials and effluents.

Fort Calhoun Objective:
Using the Offsite Dose Calculation Manual (ODCM) and OI's as reference, explain the minimum circulating water requirements during radioactive liquid effluent releases.

Question Pedigree
Bank question used on the 2002 NRC exam. Slight rewording of the stem to better fit the K/A. Replaced choice B and D, reordered choices. Not counted as modified question.

K/A Fit:
Addresses procedural actions to be taken if a circulating water trip could result in a liquid release to the river that exceeds the release permit.

Choice A:
Distractor: Plausible because two Raw water pumps are required for a release with no circulating pumps operating. Incorrect because AOP-10 directs terminating a liquid release if a circulating water pump trips.

Choice B:
Distractor: Plausible because a lower setpoint would be needed with less dilution flow. Incorrect because a release is not allowed with only one Raw Water pump operating and AOP-10 directs terminating a liquid release if a circulating water pump trips.

Choice C:
Correct answer. The release must be terminated with only one Raw Water Pump operating. AOP-10 directs terminating a liquid release if a circulating water pump trips.

Choice D:
Distracter. Plausible because this is an action that is taken before resuming a release if the radiation monitor failed. Incorrect because a release is not allowed with only one Raw Water pump operating and AOP-10 directs terminating a liquid release if a circulating water pump trips.

KA#:	000059 AK3.04	Bank Ref #:	07-11-03 017
LP# / Objective:	0711-03 01.06	Exam Level:	RO-13
Cognitive Level:	LOW	Source:	BANK
Reference:	AOP-10	Handout:	NONE

Fort Calhoun Station
Unit No. 1

AOP-10

LOSS OF CIRCULATING WATER

Change No.:	Convert
Reason for Change:	Convert document from WordPerfect to Word.
Initiator:	N/A
Preparer:	K. Horn
Issued:	05-02-06 3:00 pm

1.0 PURPOSE

This procedure provides guidance in the event of degraded operation or rupture of the Circulating Water System.

2.0 ENTRY CONDITIONS

System operation has become degraded or a system rupture has occurred which may be indicated by any of the following:

A. Decreasing pressure indications on "CONDENSER CIRC WATER PRESSURE INLET PI-1913A".

B. White light(s) on CW Pumps.

C. Circulating Water Pump discharge pressure is low.

D. Decreasing Condenser vacuum.

E. "TURBINE BLDG SUMP LEVEL HI" alarm (CB-10,11; A11).

F. "INTAKE BLDG SUMP LEVEL HI" alarm (CB-10,11; A11).

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. CAUTIONS

None

B. NOTES

None

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Terminate radioactive liquid waste releases.	
2. Ensure all CW Pumps are running.	
3. **IF** all CW flow has been lost, **THEN** GO TO Step 11.	
4. **IF** a system rupture has occurred, **THEN** GO TO Step 9.	

PRECAUTIONS (continued)

5. The Liquid Waste Disposal System is radioactively contaminated. Extra precautions should be exercised to avoid possible personnel or equipment contamination.

6. Personnel Corridor 4 may not be accessible during Post-Accident conditions due to High Radiation fields. An emergency hatch, a hoist to remove the hatch and a ladder have been installed in Corridor 52 for access to Panel AI-100 during Post-Accident conditions.

7. Only one Monitor Tank can be released at any one time.

8. A minimum of one Circulating or two Raw Water Pumps must be in operation to release a Monitor Tank.

REFERENCES/COMMITMENT DOCUMENTS

1. Procedures:
   - Fort Calhoun Station Off-Site Dose Calculation Manual (ODCM), Section 2.1.1
   - Chemistry Form FC-211, Waste Liquid Tank Release Permit

2. USAR:
   - 9.13, Sampling
   - 11.1.2, Liquid Wastes

3. Drawings

Drawings	File	Description
11405-M-7	44338	Waste Disposal System
11405-M-9	44339	Waste Disposal System
627-D-8053	10474	Aquachem Piping and Instrument Diagram Model RW 900 SP
11405-M-257	44336	Circulating Water

APPENDICES

OI-WDL-3-CL-A 28
OI-WDL-3-CL-B 30
OI-WDL-3-CL-C 33
OI-WDL-3-CL-D 35
OI-WDL-3-CL-E 38
OI-WDL-3-CL-F 40
OI-WDL-3-CL-G 43
OI-WDL-3-CL-H 45
OI-WDL-3-CL-I 48

QUESTIONS REPORT
for ILO EXAM BANK

07-11-03 017

With the plant in Mode 3, the following conditions exist:

- Circulating Water Pump, CW-1A is running
- Circulating Water Pumps, CW-1B & 1C are secured
- Raw water Pump, AC-10A is operating
- Waste Monitor Tank, WD-22B, is being released

Which of the following actions should be taken if CW-1A trips?

A. One of the other Circulating Water Pumps should be started immediately.

B.✔ The Waste Monitor Tank Release should be terminated immediately.

C. An additional Raw water Pump should be started immediately.

D. No immediate action is required as long as one Raw Water Pump remains in service.

QUESTION NUMBER: 025

The following plant conditions exist:

- The plant is operating at full power
- RM-052 is aligned to the auxiliary building stack
- The “RM-052 STACK/CNTMT NOBLE GAS HIGH RADIATION” annunciator is in alarm on panel A33C
- VIAS has actuated.
- Aux Building Supply Fan, VA-35A. is in operation
- Aux Building Exhaust Fan, VA-40A, is in operation
- Aux Building Exhaust Fan, VA-40B, is out of service

What actions should be taken to align Aux Building Ventilation per AOP-09, “HIGH RADIATION?”

A.✔ Stop VA-35A

B. Start VA-35B

C. Stop VA-40A

D. Start VA-40C

Question # 25 Rev 0

KA #: 000060 AA1.02 Tier 1 Group 2: Accidental Gaseous Radwaste Release
Ability to operate and / or monitor the following as they apply to the Accidental Gaseous Release: Ventilation system
Importance 2.9 / 3.1

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Given the caution statements and/or notes listed in this AOP, explain the reason for each.

Question Pedigree
Modified bank question.

K/A Fit:
Question addresses realigning the auxiliary building ventilation system following a high radiation alarm due to an accidental gaseous radwaste release.

Choice A:
Correct answer. AOP-09 directs that all supply fans be stopped and only one exhaust fan should be running.

Choice B:
Distractor: Plausible if Applicant believes two supply fans should be running. Incorrect because this is not in accordance with AOP-09.

Choice C:
Distractor: Plausible if Applicant believes all exhaust fans should be tripped. Incorrect because this is not in accordance with AOP-09.

Choice D:
Distractor: Plausible if Applicant believes two exhaust fans should be running to make the auxiliary building pressure more negative as is directed in AOP-08, "Fuel Handling Incident." Incorrect because this is not in accordance with AOP-09.

KA#:	000060 AA1.02	Bank Ref #:	07-17-09 019
LP# / Objective:	0717-09 01.05	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	AOP-09	Handout:	NONE

INSTRUCTIONS

8. Verify at least one of the Stack Radiation Monitors has power and the pump is energized (AI-33C).

9. **IF** any of the Stack Radiation Monitor radiation levels are rising, **THEN** minimize Auxiliary Building ventilation flow by performing the following steps (AI-44):

   a. Shutdown both Auxiliary Building Supply Fans, VA-35A/B.

(continue)

CONTINGENCY ACTIONS

8.1 **IF** Stack Radiation Monitors are not powered or pumps are not energized, **THEN** align a power source and energize the associated pump:

- RM-062 and RM-063 (MCC-4C2)
- RM-052 (MCC-3B1) (normal)
- RM-052 (MCC-4C2), Attachment 2 of OI-RM-1, Radiation Monitoring

9.1 **IF NONE** of the Stack Radiation Monitor radiation levels are rising, **THEN** GO TO AOP-21, Reactor Coolant System High Activity.

INSTRUCTIONS | CONTINGENCY ACTIONS

9. (continued)

b. Ensure **ONE** of the following Auxiliary Building Exhaust Fans is in operation:

- VA-40A
- VA-40B
- VA-40C

b.1 **IF NO** Auxiliary Building Exhaust Fans are operating **AND**, a Radwaste Building Fan is in service, **THEN** perform the following:

1) Request Security unlock doors 1007-1A and 1007-1B.

2) Open Auxiliary building to Radwaste Building door and West Radwaste Building Rollup door (Corr 26).

3) Monitor RM-043, “RWP BLDG VENT STACK GAS RADIATION MONITOR”.

10. Ensure the railroad siding rollup door is closed (Room 25).

11. **IF** VIAS has **NOT** occurred, **THEN** close **ALL** of the following supply and exhaust dampers (AI-44):

- HCV-805A/B, East SI Pump Room
- HCV-804A/B, West SI Pump Room
- HCV-803A/B, Spent Regen Chem Holdup Room

11.1 **IF** VIAS has occurred, **THEN** place **ALL** of the following switches in "PULL-TO-FILT" (AI-44):

- VA-26A
- VA-26B
- VA-27

QUESTIONS REPORT
for ILO EXAM BANK

07-17-09 019

In AOP-09, High Radiation, negative pressure in the Auxiliary Building is maintained by:

A.✔ Shutting of all supply fans and all but one exhaust fan.

B. Shutting off all but one supply fan and all exhaust fans.

C. Shutting off all but one supply fan and all but one exhaust fan.

D. shutting off all supply and exhaust fans.

QUESTION NUMBER: 026

The following RM-078 setpoints are provided in in the technical data book, TDB-IV.8.

Normal	<1.0 mrem
Warn/Alert	10 mrem
High	30 mrem

A radiation source is moved in Corridor 4 which causes the measured radiation level to rise to 45 mrem and then go back down to normal. What is the status of the alarm status lights on the RM-078 panel at AI-33 after the radiation levels have returned to normal? Assume no operator action has been taken.

A✔ The Alert and Alarm status lights are both lit and will stay lit until the Alarm ACK button is pushed.

B. The Alert status light has automatically reset and is off. The Alarm status light is lit and will stay lit until the Alarm ACK button is pushed.

C. The Alarm status light has automatically reset and is off. The Alert status light is lit and will stay lit until the Alarm ACK button is pushed.

D. The Alert and Alarm status lights have automatically reset and are off.

Question # 26 Rev 0

KA #: 000061 AK2.01 Tier 1 Group 2: Area Radiation Monitoring (ARM) System Alarms
Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location
Importance 2.5 2.6

CFR Number: 55.41(b)(11)
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Fort Calhoun Objective:
Front panel controls/ indications

Question Pedigree
Bank question used on the 2004 NRC exam.

K/A Fit:
The question addresses the interrelationship between RM-078 located in Auxiliary Building Corridor 4 and alarms at the Control Room radiation monitoring panel.

Choice A:
Correct answer: The radiation level at RM-078 went above both the alert and high alarms. Both the Alert and Alarm status lights will stay lit until the Alarm ACK button is pushed.

Choice B:
Distractor: Plausible if the Applicant believes that the alert status light automatically resets, but not the alarm status light. Incorrect because neither auto resets.

Choice C:
Distractor: Plausible if the Applicant believes that the alarm status light automatically resets, but not the alert status light. Incorrect because neither auto resets.

Choice D:
Distractor: Plausible if the Applicant believes that the alarm and alert both auto reset like the local monitor horn and flashing light. Incorrect because neither auto resets on the control room panel.

KA#:	000061 AK2.01	Bank Ref #:	07-12-03 057
LP# / Objective:	0712-03 03.03C	Exam Level:	RO-11
Cognitive Level:	LOW	Source:	BANK
Reference:	STM 33	Handout:	NONE

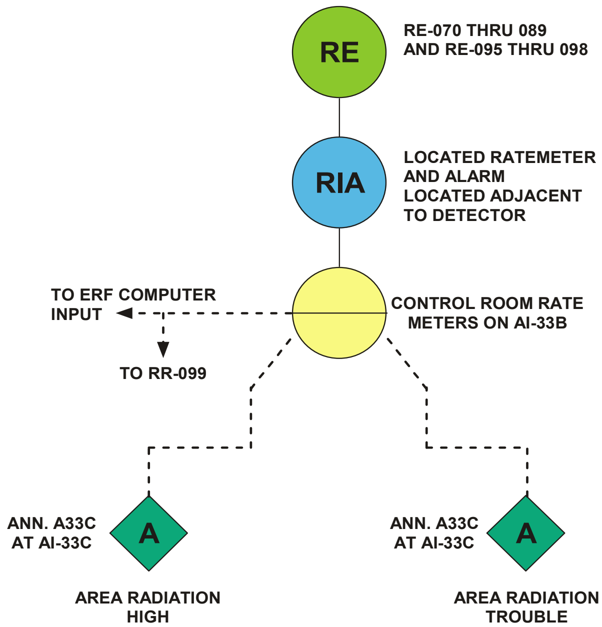


FIGURE 2-53: NORMAL RANGE AREA RADIATION MONITOR CHANNEL ARRANGEMENT

2.338 Each detector is provided with three alarms:

- A downscale alarm to indicate failure of the monitor, loss of power, or loss of detectable background radiation.
- An alert-level alarm to indicate increasing radiation.
- A high-level alarm to alert personnel for evacuation.

••Location

2.339 The read-out modules, independent channel power supplies, and multipoint recorder are installed on Control Room panel AI-33B.

••Power Supplies

2.340 Instrument bus AI-40D supplies 120 VAC power to the normal-range area monitors and recorder RR-099 on panel AI-33B.

••Instrumentation and Control

2.341 Each channel has two independently adjusted setpoints. The setpoints are established based on the individual detector local background radiation level. The ALERT alarm level indicates the dose rates in the area have increased to an abnormal level. The HIGH RADIATION alarm level indicates the area dose rate has reached the permissible limit for continued occupancy set forth in 10 CFR

20. The alarm setpoints are changed on certain monitors for prolonged shutdowns and outages due to changes in background radiation levels.

2.342 The Control Room ratemeters (Victoreen Model 946A) include the following indications and controls on their meter face:

- Main LED display with a five-digit capacity.
- A LED bargraph display with a range of 10^{-1} to 10^{7} mR/hr.
- Status Lamps for HIGH ALARM (red), WARN ALARM (amber), FAIL ALARM (red) and RANGE ALARM (red).
- WARN and HIGH pushbuttons used to display the high and warn level setpoints.
- CHECK SOURCE pushbutton used to actuate the electronic check source and its associated green LED.
- ALARM ACK. pushbutton that resets relay outputs and causes alarm lights to stop flashing after acknowledgement.
- An alternate action ON/OFF pushbutton for power control.

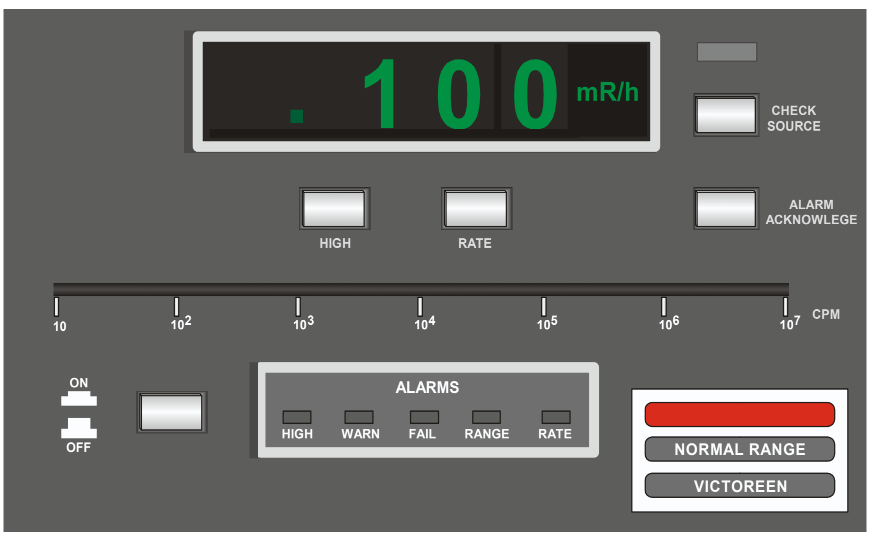


FIGURE 2-54: AREA RADIATION MONITOR NORMAL RANGE RATEMETER

•••Local

2.343 The local ratemeters provide personnel with continuous alarm and indication via a horn and light on each monitor. The horn and light are automatically reset when the radiation level lowers to less than the high alarm setpoint. **NOTE**: High alarm conditions cannot be canceled locally.

•••Remote

2.344 The Control Room ratemeters (Victoreen Model 946A) include the following controls on the meter face:

- WARN and HIGH pushbuttons used to display the high and warn level setpoints.
- CHECK SOURCE pushbutton used to actuate the electronic check source and its associated green LED.
- ALARM ACK. pushbutton that resets relay outputs and causes alarm lights to stop flashing after acknowledgement.
- An alternate action ON/OFF pushbutton for power control.

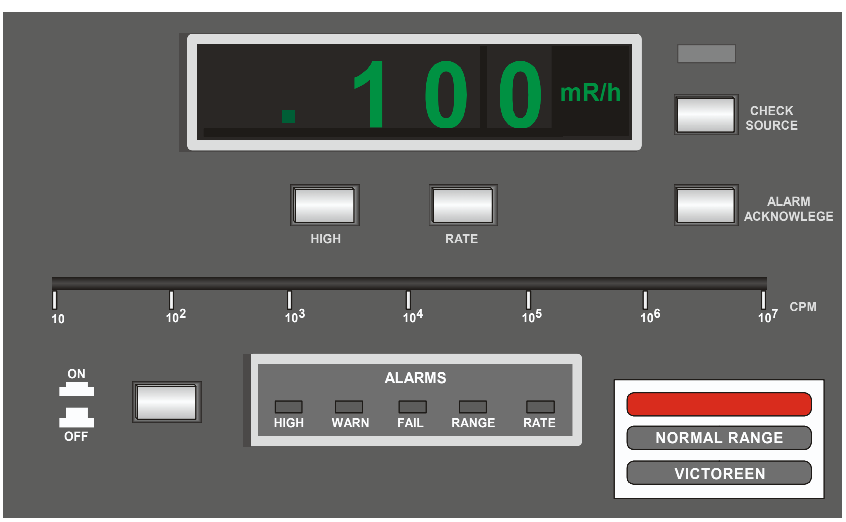
•••Interlocks

2.345 Not applicable.

•••Alarms and Indications

2.346 The Control Room ratemeters (Victoreen Model 946A) include the following indications on the meter face:

- Main LED display with a five-digit capacity.
- A LED bargraph display with a range of 10^{-1} to 10^{7} mR/hr.
- Status Lamps for HIGH ALARM (red), WARN ALARM (amber), FAIL ALARM (red) and RANGE ALARM (red).



QUESTION NUMBER: 027

An ATWS event has occurred. EOP-00 standard post-trip actions have been completed and the Reactivity Control Safety Function is still not satisfied.

After entering EOP-20, which one of the following actions will be performed that was NOT attempted using EOP-00?

A.✔ Borating from the safety injection and refueling water storage tank

B. Manually opening the clutch power supply breakers

C. Borating from the boric acid storage tanks

D. Manually actuating the DSS trip switches

Question # 27 Rev 0

KA #: CE-E09 EA2.02 Tier 1 Group 2:Functional Recovery
Ability to determine and interpret the following as they apply to the (Functional Recovery) Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.
Importance 3.5 / 4.0

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Given the Resource Assessment Trees, basically DESCRIBE the Method, Path and Acceptance Criteria for each success path.

Question Pedigree
Bank question used on the 2005 NRC exam

K/A Fit:
Question addresses procedural response to loss of a safety function.

Choice A:
Correct answer: Borating from the SIRWT is addressed in EOP-20 but not in EOP-00.

Choice B:
Distractor: Plausible because it is a method to attempt to satisfy the safety function. Incorrect becasue it is directed in both EOP-00 and EOP-20.

Choice C:
Distractor: Plausible because it is a method to attempt to satisfy the safety function. Incorrect becasue it is directed in both EOP-00 and EOP-20.

Choice D:
Distractor: Plausible because it is a method to attempt to satisfy the safety function. Incorrect becasue it is directed in both EOP-00 and EOP-20.

KA#:	CE-E09 EA2.02	Bank Ref #:	07-18-18 020
LP# / Objective:	0718-18 01.05	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	EOP-20	Handout:	NONE

5.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
Time of RX Trip:________	
1. Verify Reactivity Control is established by performing steps a and b:	1.1 **IF** the reactor did **NOT** trip, **THEN** establish Reactivity Control by performing step a, b, c or d:
a. Verify **ALL** of the following: • No more than one Regulating or Shutdown CEA is **NOT** inserted • Reactor power is lowering • Startup rate is negative	a. Manually trip the Reactor (CB-4). b. Manually trip the Reactor (AI-31). c. Place the DSS Manual Trip Switches in "TRIP" (AI-66A/B).
b. Monitor for an uncontrolled RCS cooldown.	d. Manually open the CEDM Clutch Power Supply Breakers (AI-57).
(continue)	(continue)

INSTRUCTIONS

1. (continued)

(continue)

CONTINGENCY ACTIONS

1.2 **IF** more than one CEA is **NOT** fully inserted,
OR an uncontrolled RCS cooldown is in progress**,**
THEN emergency borate by performing the following:

a. Ensure **BOTH** of the following valves are closed:

- FCV-269X, Demin Water Makeup Valve
- FCV-269Y, Boric Acid Makeup Valve

b. Open **ALL** of the following valves:

- HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
- HCV-265, CH-11A Gravity Feed Valve
- HCV-258, CH-11B Gravity Feed Valve

c. Start **ALL** of the following pumps:

- Boric Acid Pumps, CH-4A/B
- Charging Pumps, CH-1A/B/C

(continue)

INSTRUCTIONS

1. (continued)

CONTINGENCY ACTIONS

1.2 (continued)

d. Close LCV-218-2, VCT Outlet Valve.

e. Ensure **ALL** of the following valves are closed:

- LCV-218-3, Charging Pump Suction SIRWT Isolation Valve
- HCV-257, CH-4B Recirc Valve
- HCV-264, CH-4A Recirc Valve

f. Borate until adequate shutdown margin is established.

9.0 REACTIVITY CONTROL

SAFETY FUNCTION: Reactivity Control

SUCCESS PATH: Boration Using CVCS: RC-2

RESOURCE TREE: Tree A

RC-2

INSTRUCTIONS	CONTINGENCY ACTIONS
✘1. Maintain RCS temperature constant by controlled steaming using the Steam Dump and Bypass Valves.	1.1 **IF** the Steam Dump and Bypass Valves are **NOT** available, **THEN** control RCS temperature by performing step a or b: a. Steam the S/Gs as follows: 1) Ensure MS-164, "MAIN STEAM LINE "A" STEAM DUMP TO ATMOSPHERE ISOLATION VALVE", is open (Room 81). 2) Ensure MS-407, "ATMOSPHERIC DUMP VALVE HCV-1040 2nd ISOLATION ", is open (Room 81).
(continue)	(continue)

9.0 REACTIVITY CONTROL

RC-2

INSTRUCTIONS	CONTINGENCY ACTIONS
✘1. (continued)	1.1.a (continued) 3) Control HCV-1040, Atmospheric Dump Valve. b. Operate at least one of the following Air Assisted Main Steam Safety Valves: • MS-291 • MS-292
2. Verify a BAST is available for boration.	2.1 **IF** both BASTs are unavailable, **THEN** GO TO Step 4 to borate the RCS.

9.0 REACTIVITY CONTROL

RC-2

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

If 1A3 is not energized, manual action may be needed to establish boric acid flow.

✘3. Commence emergency boration using BASTs to achieve adequate Shutdown Margin by performing the following:

a. Ensure **BOTH** of the following valves are closed:

- FCV-269X, Demin Water Makeup Valve
- FCV-269Y, Boric Acid Makeup Valve

b. Open **ALL** of the following valves:

- HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
- HCV-265, CH-11A Gravity Feed Valve
- HCV-258, CH-11B Gravity Feed Valve

(continue)

9.0 REACTIVITY CONTROL

RC-2

INSTRUCTIONS	CONTINGENCY ACTIONS
✘3. (continued)	
c. Ensure **ALL** of the following Charging Isolation Valves are open: • HCV-247 • HCV-238 • HCV-248 • HCV-239	c.1 **IF** the Charging Header is inoperable, **THEN** open **ALL** of the following valves to charge via the HPSI Header: • HCV-308, Charging Pump HPSI Header Isolation Valve • HCV-2987, HPSI Header Isolation Valve • HPSI Loop Injection Valves
d. Start **ALL** of the following pumps: • Boric Acid Pumps, CH-4A/B • Charging Pumps, CH-1A/B/C	
e. Ensure **ALL** of the following valves are closed: • LCV-218-2, VCT Outlet Valve • LCV-218-3, Charging Pump Suction SIRWT Isolation Valve • HCV-257, CH-4B Recirc Valve • HCV-264, CH-4A Recirc Valve Time: ____________	
f. GO TO Step 5.	

9.0 REACTIVITY CONTROL

RC-2

INSTRUCTIONS | CONTINGENCY ACTIONS

✘4. Commence boration using the SIRWT to achieve adequate Shutdown Margin by performing the following:

a. Open LCV-218-3, Charging Pump Suction SIRWT Isolation Valve.

b. Stop both of the Boric Acid Pumps, CH-4A/B.

c. Ensure **ALL** of the following valves are closed:

- HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve
- HCV-265, CH-11A Gravity Feed Valve
- HCV-258, CH-11B Gravity Feed Valve
- LCV-218-2, VCT Outlet Valve

(continue)

9.0 <u>REACTIVITY CONTROL</u>

RC-2

INSTRUCTIONS	CONTINGENCY ACTIONS
✘4. (continued)	
d. <u>Ensure</u> **ALL** of the following Charging Isolation Valves are open: • HCV-247 • HCV-238 • HCV-248 • HCV-239	d.1 **IF** the Charging Header is inoperable, **THEN** <u>open</u> **ALL** of the following valves to charge via the HPSI Header: • HCV-308, Charging Pump HPSI Header Isolation Valve • HCV-2987, HPSI Header Isolation Valve • HPSI Loop Injection Valves
e. <u>Start</u> all of the Charging Pumps, CH-1A/B/C. Time: ____________	

QUESTION NUMBER: 028

The Reactor was operating at full power with a normal electrical alignment when a reactor trip occurred. Following the trip, fast transfer to 161 KV failed.

Which Reactor Coolant Pumps would continue to operate in this situation?

A. RC-3A and RC-3B

B. RC-3A and RC-3C

C.✓ RC-3C and RC-3D

D. RC-3B and RC-3D

Question # 28 Rev 0

KA #: 003000 K2.01 Tier 2 Group 1: Reactor Coolant Pump System
Knowledge of bus power supplies to the following: RCPs
Importance 3.1 3.1

CFR Number: 55.41(b)(3)
Mechanical components and design features of the reactor primary system.

Fort Calhoun Objective:
LIST the power supplies for the RCPs.

Question Pedigree
Bank question used on the 1997 NRC exam.

K/A Fit:
Question addresses the power supplies to the RCPs.

Choice A:
Distractor:Plausible if Applicant does not understand RCP power supplies or the normal electrical alignment. Incorrect because RC-3A and RCP-3B would lose power if fast transfer failed.

Choice B:
Distractor:Plausible if Applicant does not understand RCP power supplies or the normal electrical alignment. Incorrect because RC-3A would lose power if fast transfer failed.

Choice C:
Correct answer. RC-3C and RC-3D are normally supplied by 161 KV and will continue to run.

Choice D:
Distractor:Plausible if Applicant does not understand RCP power supplies or the normal electrical alignment. Incorrect because RCP-3B would lose power if fast transfer failed.

KA#:	003000 K2.01	Bank Ref #:	07-11-20 024
LP# / Objective:	0711-20 01.07C	Exam Level:	RO-3
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM 37	Handout:	NONE

2.3.2 (continued)

W. Water from the Component Cooling Water System cools the reactor coolant pump shaft seal and pump motor. Both the upper and lower oil reservoirs in the pump motor have oil-to-water heat exchangers for cooling the oil.

X. The seal area cooling water is further subdivided into two streams:

- Cooling for the thermal barrier area, the labyrinth passageway through which the controlled bleedoff enters the seal cavity area
- Cooling for the integral heat exchanger tubes through which the recirculating water passes

Y. Component cooling water enters the integral heat exchanger and flows through the annular region formed by the heat exchanger's double pipe tube arrangement. Recirculating water, contained in the inner pipe of the heat exchanger tubing, transfers heat energy to the cooler component cooling water.

Z. Each pump has its own high-pressure oil lift pump to lubricate the thrust bearings and upper guide bearings during reactor coolant pump startup and shutdown. This oil pressure separates the thrust shoes and lubricates the thrust bearing for pump startup. While the RCP is running, the thrust runner acts as the oil pump. It is sometimes referred to as the low-pressure oil pump. The oil lift pump should be in operation for at least two minutes prior to starting the RCP.

2.3.3 Location

A. The reactor coolant pumps are located on the 1013-foot elevation of the Containment Structure: RC pumps RC-3A and RC-3B are on the south side of the reactor vessel along with steam generator RC-2A, and RC pumps RC-3C and RC-3D are on the north side of the reactor vessel along with steam generator RC-2B.

2.3.4 Power Supplies

A. The reactor coolant pumps are powered from the 4.16 kV buses. They receive power from buses 1A1, 1A2, 1A3 or 1A4 for RCPs A, B, C and D, respectively. The oil lift pumps and motor heaters for the RCPs are powered from motor control centers 3B1, 4A1, 3A1 and 4C1 for RCPs A, B, C and D, respectively.

2.13.3 Location

A. The 4160V switchgear is located in the main switchgear rooms separated by a fire barrier.

B. The manual transfer switches for breaker control power are located at the front of the auxiliary power compartment of the switchgear.

2.13.4 Power Supplies

A. The normal power supply for buses 1A1 and 1A2 is the 22 kV System through unit auxiliary transformers T1A-1 and T1A-2. The normal power supply for buses 1A3 and 1A4 is the 161 kV System through engineered safeguards transformers T1A-3 and T1A-4. Buses 1A3 and 1A4 provide a limited source of emergency power from the diesel generators.

B. Control power for breaker operation is supplied from the 125V DC Distribution System. Buses 1A1 and 1A3 are normally supplied from DC Bus no. 1 and 1A2 and 1A4 are supplied from DC Bus No. 2. Alternate or emergency power is available from the opposite bus.

2.13.5 Instrumentation and Control

A. Each 4160V bus has voltmeters for each phase on the vertical section of CB-20.

B. Two of two loss-of-voltage signals will initiate a load shed. A load shed opens breakers on the affected bus to prevent overloading an incoming supply and to allow the operator to realign the system. A backup load shed signal is generated if the circuit breakers feeding the bus are open. Load shed can also be caused by the lockout relays on the breakers feeding the 4160V buses using the auxiliary relay contacts.

C. The 4160V supply breakers 1A11, 1A13, 1A22, and 1A24 will trip on any of the following conditions:

- Undervoltage on the supply transformer secondary
- OPLS (1A13, 1A24 only)
- Main Generator Lockout (86/G1)
- Main Transformer Lockout (86/GT1)
- Main Turbine Lockout (86/SVG1)
- 345 kV Breaker Failure Lockouts (86/BF-4 or 86/BF-5)

D. Each of the breakers will trip and lockout if a signal is received from its inverse time overcurrent (51) relay or lock-out (86) relay.

QUESTION NUMBER: 029

Given the following plant conditions:

- The reactor tripped from full power due to low steam generator level following the loss of all Main Feedwater Pumps
- Water Level in Steam Generator “A” is lower than the water level in Steam Generator “B”
- The water level in both steam generators is lowering at the same rate

Assuming no operator action is taken, how will the Auxiliary Feedwater System respond in this situation?

A. AFW will be supplied to S/G "A" only when narrow range level in S/G "A" falls to 32%

B✔ AFW will be supplied to S/G "A" only when wide range level in S/G "A" falls to 32%

C. AFW will be supplied to both S/G's when narrow range level in S/G "A" falls to 32%

D. AFW will be supplied to both S/G's when wide range level in S/G "A" falls to 32%

Question # 29 Rev 0

KA #: 059000 K3.02 Tier 2 Group 1: Main Feedwater System
Knowledge of the effect that a loss or malfunction of the MFW will have on the following: AFW system
Importance 3.6 / 3.7

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
EXPLAIN the operation of the AFW System following an Engineered Safeguards AFAS.

Question Pedigree
New question.

K/A Fit:
Question addresses AFW system response to a loss of main feedwater.

Choice A:
Distractor: Plausible if Applicant believes AFAS is initiated by narrow range S/G level. Incorrect because it is initiated by wide range level.

Choice B:
Correct answer. Since the stem states that water level in the "A" S/G is lower than the level in the "B" S/G and the level in both S/G's is lowering at the same rate, 32% wide range level in S/G "A" will be reached first and will initiate flow to S/G "A" only.

Choice C:
Distractor: Plausible if Applicant believes AFAS is actuated on narrow range level and also believes that the logic initiates flow to both S/G's. Incorrect because it is initiated on wide range level and only feeds the S/G with wide range level below 32%.

Choice D:
Distractor: Plausible if Applicant believes that the logic initiates flow to both S/G's. Incorrect because it only feeds the S/G with wide range level below 32%.

KA#:	059000 K3.02	Bank Ref #:	N/A
LP# / Objective:	0711-01 01.04	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	NEW
Reference:	STM 04	Handout:	NONE

1.17 During plant shutdown, the main feedwater pumps are used to feed the steam generators until the plant heat load is within the capacity of the motor-driven or engine-driven AFW pump. AFW is aligned to the main feed header for plant shutdown and the main feed pump is shut down.

1.18 The turbine-driven AFW pump is not normally used for routine plant operation. During heatup, the turbine is a heat loss that slows the heatup rate. The turbine exhaust is to atmosphere, so the inventory is lost. The turbine is designated the emergency pump and is periodically started for surveillance testing to verify operability.

1.19 If the steam generator level instrumentation senses a low level (32 percent wide range) in either steam generator, the AFW System is designed to start both AFW pumps and supply emergency feedwater to the intact steam generator that has low level. Feedwater is automatically directed through the auxiliary feedwater nozzles. After the situation stabilizes, the operator can realign the system for AFW flow through the feed regulating bypass valves or take manual control to avoid thermal shock to the steam generators.

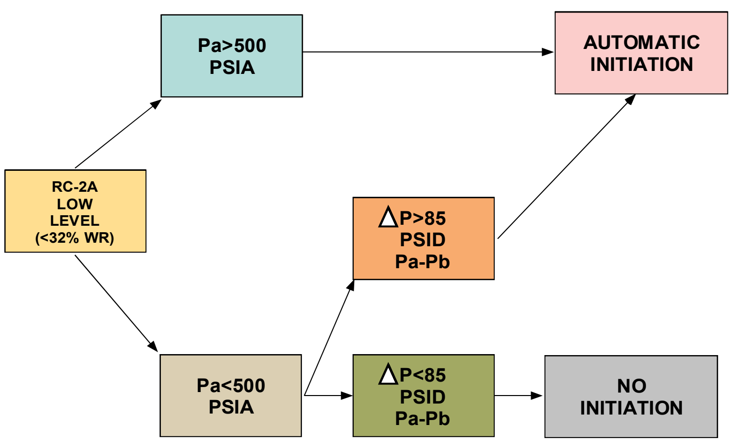


FIGURE 1-3: AUTOMATIC INITIATION AUXILIARY FEEDWATER

1.20 When both AFW pumps are operating, the turbine-driven pump will be providing a slightly greater proportion of the total flow because of its head versus flow curve being higher and flatter. The steam header pressure is normally maintained by the steam dump and bypass system at 900 psig, during startup and shutdown. The motor-driven pump is a constant speed pump and its discharge pressure is controlled by the feedwater header pressure. The shut-off head of the motor-driven pump is about 1200 psi.

QUESTION NUMBER: 030

When placing a second charging pump in service, what action is necessary to match charging and letdown flows and minimize the pressurizer level transient?

A. The pressure setpoint on Pressure Controlled, "PIC-210," is adjusted until charging and letdown flows are matched.

B.✔ The level bias potentiometer on Letdown Flow Controller, HIC-101-1/101-2, is adjusted until charging and letdown flows are matched.

C. The selected Letdown Control Valve, LCV-101-1 or LCV-101-2, is manually adjusted until charging and letdown flows are matched

D. The setpoint on the selected Pressurizer Level Controller, LRC-101X or LRC-101Y, is adjusted until charging and letdown flows are matched.

Question # 30 Rev 0

KA #: 004000 A4.08 Tier 2 Group 1:Chemical and Volume Control System
Ability to manually operate and/or monitor in the control room: Charging
Importance 3.8 / 3.4

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
EXPLAIN, the manual and automatic functions of control valves in the CVCS.

Question Pedigree
Bank Question used on 2001-2 NRC exam. Revised two distractors but left the stem as is. Not counted as modified question.

K/A Fit:
Question addresses operations for placing an additional charging pump in service.

Choice A:
Distractor:Plausible because PIC-210 is adjusted to maintain letdown pressure at 300 psig. Incorrect because the bias pot is used to match charging and letdown flows.

Choice B:
Correct answer:The bias pot is used to match charging and letdown flows.

Choice C:
Distractor: Plausible because letdown flow could be manually adjusted until charging and letdown flow are matched. Incorrect because OI-CH-1 directs using the bias pot.

Choice D:
Distractor: Plausible because the pressurizer level controller setpoint could be manually adjusted until charging and letdown flow are matched. Incorrect because OI-CH-1 directs using the bias pot.

KA#:	004000 A4.08	Bank Ref #:	07-11-02 010
LP# / Objective:	0711-02 01.02	Exam Level:	RO-5
Cognitive Level:	LOW	Source:	BANK
Reference:	OI-CH-1	Handout:	NONE

Attachment 3 - Raising Charging and Letdown Flows

PROCEDURE (continued) (✓) INITIALS

2

f. Open and backseat the selected charging pump Discharge Valve:

- CH-193 (for CH-1A) ___
- CH-192 (for CH-1B) ___
- CH-190 (for CH-1C) ___ ________

____
Ind Verif

g. Place control switch for selected pump in AFTER STOP:

- CH-1A ___
- CH-1B ___
- CH-1C ___

h. Repeat this step as necessary for each affected charging pump. ________

3. Start the selected Charging Pump:

- CH-1A, Charging Pump ___
- CH-1B, Charging Pump ___
- CH-1C, Charging Pump ___ ________

NOTES

1 PIC-210 Letdown Press Cntrlr should be continuously monitored while adjusting letdown flow.

2. Steps 4 and 5 may be performed concurrently without the procedure in hand. Sign-offs may be completed after these steps are performed.

4. Raise the bias on HIC-101-1/101-2, Letdn Throttle Valves Controller, and observe an increase in Letdown flow. ________

5. Adjust PIC-210 as necessary to maintain Letdown pressure approximately 300 psig. ________

6. Continue to adjust the bias on HIC-101-1/101-2 until Pressurizer level is stabilized at the programmed setpoint with two Charging Pumps in operation. ________

Completed by____Date/Time_______/_______

QUESTIONS REPORT
for ILO EXAM BANK

07-11-02 010

When placing a second Charging Pump in service, what action is necessary to match Charging and Letdown flows and minimize the Pressurizer level transient?

A. The Letdown Control Valves, LCV-101-1 or LCV-101-2, are controlled manually when more than one (1) Charging Pump is in operation.

B.✓ The Level Bias Potentiometer on Letdown Flow Controller, HIC-101-1/101-2, is manually adjusted to match flows.

C. The pressure setpoint on PIC-210 is changed until Charging and Letdown flows are matched.

D. No manual adjustments are required. Charging and Letdown are matched automatically with no change in Pressurizer level.

QUESTION NUMBER: 031

Given the following plant conditions:

- A plant cooldown was in progress
- Shutdown cooling is in service using LPSI Pump, SI-1A
- The CRS has entered AOP-19, "Loss of Shutdown Cooling," due to lowering RCS water level
- RVLMS, LI-197, LI-199 and LIS- 119 all indicate that RCS water level is below the centerline of the hot legs.

What action should be taken immediately by the RO per AOP-19?

A. Manually initiate PPLS

B. Start LPSI Pump, SI-1B

C. Start all charging pumps

D.✔ Stop LPSI Pump, SI-1A

Question # 31 Rev 0

KA #: 005000 2.4.01 Tier 2 Group 1: Residual Heat Removal System
Knowledge of EOP entry conditions and immediate action steps.
Importance 4.6 / 4.8

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Use the Loss of Shutdown Cooling Procedure to mitigate the consequences of a loss of cooling to the Reactor Coolant System.

Question Pedigree
New question.

K/A Fit:
Question addresses immediate action to be taken if LPSI pump is about to lose suction while on shutdown cooling.

Choice A:
Distractor:Plausible because PPLS is used for a loss of RCS inventory during normal operation. Incorrect because ECCS is not aligned for normal operation and AOP-19 directs stopping the running LPSI pump.

Choice B:
Distractor: Plausible if Applicant believes this will increase RCS inventory. Incorrect because the result may be two cavitating LPSI pumps and AOP-19 directs stopping the running LPSI pump.

Choice C:
Distractor: Plausible as this will increase RCS inventory. Incorrect because AOP-19 directs stopping the running LPSI pump.

Choice D:
Correct answer: Aop-19 directs stopping the running LPSI pump to prevent suction voiding.

KA#:	005000 2.4.01	Bank Ref #:	N/A
LP# / Objective:	0717-19 01.00	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	AOP-19	Handout:	NONE

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Ensure **ALL** refueling operations are stopped.	
2. **(STA)** IMPLEMENT Attachment C, Time to Boil Determination.	
3. IMPLEMENT the Emergency Plan.	
4. Verify RCS Water Level is above the centerline of the Hot Leg using at least two of the following level indications: • RVLMS (29%) • LI-197 (1006.5 feet) • LI-199 (1006.5 feet, Containment) • LIS-119 (1006.5 feet)	4.1 **IF** RCS Water Level is **NOT** above the centerline of the Hot Leg, **THEN** stop the operating LPSI or CS Pump.

QUESTION NUMBER: 032

Which of the following will result in a Containment Isolation Actuation Signal (CIAS)?

A. Containment Radiation Monitor, RM-051, fails high and alarms.

B. Pressure in both steam generators drops below 500 psia due to a steam line break downstream of the MSIVs.

C. Pressurizer pressure channel B/PIA-102Y fails low.

D.✓ Inadvertent actuation of ESF relay, 86A/CPHS.

Question # 32 Rev 0

KA #: 006000 K4.30 Tier 2 Group 1: Emergency Core Cooling System
Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Containment isolation
Importance 3.6 / 3.9

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
EXPLAIN how each prime and backup actuation signal is developed.

Question Pedigree
Bank question used on the 2005 NRC exam.

K/A Fit:
Question addresses failure of a relay that will initiate containment isolation.

Choice A:
Distractor: Plausible because this will cause a VIAS actuation. Incorrect because it will not initiate CIAS.

Choice B:
Distractor: Plausible because this will cause a SGLS actuation. Incorrect because it will not initiate CIAS.

Choice C:
Distractor: Plausible because this will provide a partial logic to a PPLS signal that would also initiate CIAS. Incorrect because only one pressure channel failed and 2/4 logic is required.

Choice D:
Correct answer: Actuation of relay 86A/CPHS will result in CPHS.

KA#:	006000 K4.30	Bank Ref #:	07-12-14 017
LP# / Objective:	0712-14 01.04	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	BANK
Reference:	STM 19	Handout:	NONE

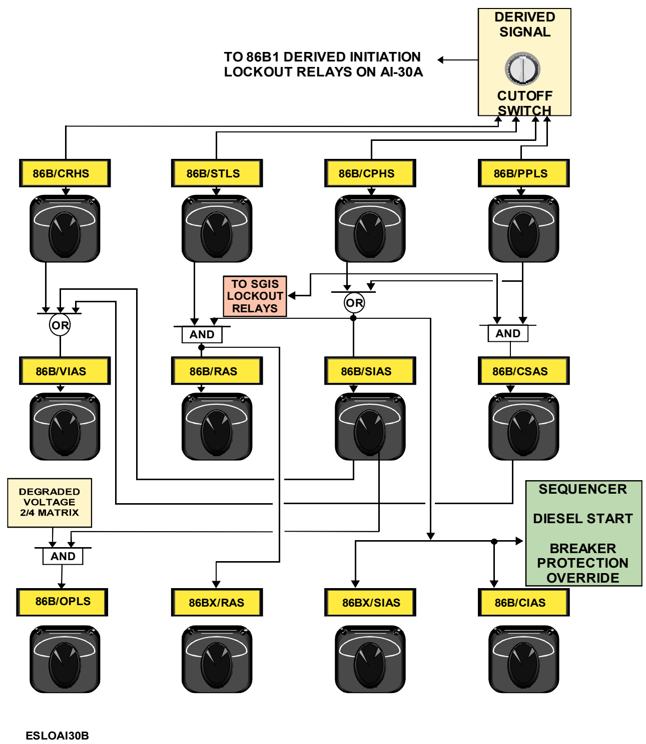


FIGURE 1-4: LOCKOUT RELAYS AI-30B

1.34 Most of the lockout relays are arranged in four groups on AI-30A and AI-30B. Each group comprises twelve lockout relays. The 24 lockout relays on AI-30B are functionally redundant to those on AI-30A and are arranged to form a mirror image of the lockout relays on AI-30A. The prime initiation relays and prime actuation relays form the upper group of 12 lockout relays on each panel and are designated 86A on AI-30A and 86B on AI-30B. The derived initiation relays and backup actuation relays form the lower group of 12 lockout relays on each panel and are designated 86B1 on AI-30A and 86A1 on AI-30B. Each derived signal lockout relay is tripped by its associated prime initiation relay. For example, 86A1/CPHS will trip when 86A/CPHS trips. Each backup actuation lockout relay is tripped by the appropriate logical combination of derived lockout relays located in the same group. For example, 86A1/CSAS will trip when 86A1/CPHS and 86A1/PPLS trip.

2.129 CIAS lockout relay actuation:

- De-energizes 22 containment isolation relays and 6 ventilation isolation relays.
- Trips lockout relays on AI-43A and AI-43B.
- Prime lockout relays provide signals to the ERF computer.
- Provides annunciation.

••Location

2.130 Containment isolation panels AI-43A and AI-43B are in the Control Room and contain lockout relays, the containment isolation relays, position indicating lights for all containment isolation valves, control switches for all isolation valves which do not have control switches elsewhere in the Control Room, and annunciator panels.

••Power Supplies

2.131 The self-reset alarm relays are powered from the 125V DC buses.

••Instrumentation and Control

2.132 Containment isolation panels AI-43A and AI-43B are in the Control Room and contain lockout relays, the containment isolation relays, position indicating lights for all containment isolation valves, control switches for all isolation valves which do not have control switches elsewhere in the Control Room, and annunciator panels. If a given line has two isolation valves, normally one valve receives an isolation signal from each containment isolation panel. If a flowpath has only one isolation valve, it receives an isolation signal from both containment isolation panels.

QUESTION NUMBER: 033

Given the following:

- ECCS actuation has occurred due to a PPLS
- EOP-03, LOSS OF COOLANT ACCIDENT, has been entered
- Step 33 has been reached
- Step 33 says, "Verify that the Containment Sump level rises as the SIRWT level lowers"
- SIRWT level is 95 inches and lowering
- Containment sump level is low and not rising

What action should be taken per EOP-03?

A. Minimize ECCS flow to the minimum required to remove decay heat.

B. Initiate containment spray to ensure adequate containment sump level.

C. ✔ Begin blended makeup to the SIRWT to increase SIRWT inventory.

D. Trip the LPSI pumps to reduce SIRWT depletion rate.

Question # 33 Rev 0

KA #: 006000 A1.15 Tier 2 Group 1: Emergency Core Cooling System
Ability to predict and/or monitor changes in parameters RWST Level and temperature
Importance 3.3 / 3.9

CFR Number 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
GIVEN a copy of the Technical Basis Documents (TBDs), EXPLAIN the bases behind the major operator actions contained in EOP-03, LOCA.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
Question addresses monitoring level in the SIRWT during an interfacing system LOCA.

Choice A:
Distractor: Plausible because this is an action directed for indications of containment sump strainer clogging .Incorrect because the ECCS pumps are still drawing suction from the SIRWT.

Choice B:
Distractor: Plausible because this is an action that may be recommended by SAMG evaluators during an interfacing system LOCA. Incorrect because it is not directed by the EOPs.

Choice C:
Correct answer: EOP-03 directs a blended makeup to the SIRWT.

Choice D:
Distractor: Plausible because action would reduce SIRWT depletion rate. Incorrect because the LPSI pumps may still be required to operate at this point in a LOCA.

KA#:	006000 A1.15	Bank Ref #:	07-18-13 076
LP# / Objective:	0718-13 01.04	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	EOP-03	Handout:	NONE

INSTRUCTIONS	CONTINGENCY ACTIONS
✘32. (continued) c. <u>Close</u> **ALL** of the following valves: • HCV-268, Boric Acid Pump Header to Charging Pumps Isolation Valve • HCV-265, CH-11A Gravity Feed Valve • HCV-258, CH-11B Gravity Feed Valve Time: __________	
✘33. <u>Verify</u> that the Containment Sump level rises as the SIRWT level lowers. (continue)	33.1 **IF** Containment Sump level is **NOT** rising, **THEN** <u>commence</u> blending to the SIRWT by performing the following: a. <u>Open</u> CH-152, "CHARGING PUMPS CH-1A,B&C SUCT HDR SI AND REFUELING WATER TANK SI-5 BLENDED BORIC ACID SUPPLY VALVE" (Corridor 26). b. <u>Start</u> at least one Boric Acid Pump, CH-4A/B. (continue)

QUESTION NUMBER: 034

Given the following plant conditions:

- The Pressurizer Quench Tank is being filled to clear a low level alarm
- During the fill, the “QUENCH TANK PRESS HI” annunciator went into alarm on CB-1,2,3/A4.
- Quench Tank pressure is 10.5 psig

Which one of the following actions should be taken to restore pressure in the Pressurizer Quench Tank per OI-RC-6, “PRESSURIZER QUENCH TANK NORMAL OPERATION?”

A✓ Open HCV-155, "Quench Tank Vent Valve."

B. Lower the setpoint on PCV-2624, "Nitrogen Pressure Regulator."

C. Open HCV-2637, "N_2 Hdr to PQT/RCDT Regulator Bypass valve."

D. Open HCV-153, "Quench Tank Drain Valve."

Question # 34 Rev 0

KA #: 007000 A2.05 Tier 2 Group 1: Pressurizer Relief Tank / Quench Tank System
Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Exceeding PRT high-pressure limits
Importance 3.2 / 3.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
EXPLAIN the purpose of the quench tank.

Question Pedigree
New question.

K/A Fit:
Question addresses mitigating a high pressure in the quench tank during filling operations.

Choice A:
Correct answer: OI-RC-6 directs opening the vent valve to maintain pressure below 10 psig.

Choice B:
Distractor: Plausible because nitrogen is provided to the quench tank through this regulator. Incorrect because nitrogen to the quench tank would be isolated during this operation.

Choice C:
Distractor: Plausible if Applicant believes opening this bypass valve will allow nitrogen to flow from the quench tank to the RCDT. Incorrect because it won't.

Choice D:
Distractor: Plausible because draining the quench tank to the RCDT will lower pressure. Incorrect because it will also lower level and the objective of this operation is to raise level in the quench tank.

KA#:	007000 A2.05	Bank Ref #:	N/A
LP# / Objective:	0711-20 01.09	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	OI-RC-6	Handout:	NONE

C
Continuous Use

Attachment 1 - Filling the Quench Tank

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision No. ________________ Date:________ ________

2. Demineralized Water System is operable to HCV-1560A and HCV-1560B per OI-DW-4. ________

3. Checklist OI-RC-6-CL-A has been completed. ________

PROCEDURE

> **NOTE**
>
> Quench Tank level and pressure may also be read on ERF Computer Points L132, P130 and P131.

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

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

> **CAUTION**
> During normal Reactor operation, Quench Tank pressure shall not exceed 10 psig on PIA-131.

2. Monitor PIA-131, Pressurizer Quench Tank, pressure (CB-1/2/3). ________

3. Open HCV-155, Quench Tank Vent Valve, as necessary, to maintain pressure less than 10 psig (CB-1/2/3). ________

4. WHEN filled to the desired level on LIA-132, Pressurizer Quench Tank level (normal 73%),
   THEN close the following valves:

   - HCV-1560A __
   - HCV-1560B __ ________

Completed by ____Date/Time_______/_____

QUESTION NUMBER: 035

A SIAS signal will automatically ________ the CCW inlet and outlet valves to CCW Heat Exchanger AC-1C, HCV-492A/B, and _________ the CCW Heat Exchanger Bypass Valve, HCV-497.

A. Open, Open

B.✓ Open, Close

C. Close, Open

D. Close, Close

Question # 35 Rev 0

KA #: 008000 A3.08 Tier 2 Group 1: Component Cooling Water System
Ability to monitor automatic operation of the CCWS, including: Automatic actions associated with the CCWS that occur as a result of a safety injection signal
Importance 3.6 / 3.7

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective
EXPLAIN the response of the CCW System to signals from the Engineered Safeguards Control System.

Question Pedigree
Modified question. Source question was used on the 1995 NRC exam.

K/A Fit:
Question addresses automatic actions for the CCW system following a SIAS.

Choice A:
Distractor: Plausible if Applicant believes all valves open to maximize system flow. Incorrect because HCV-497 closes.

Choice B:
Correct answer: CCW heat exchanger inlet and outlet valves all open and HCV-497 closes to maximize flow through the heat exchangers. Note: HCV-497 was modified. In the original plant design, HCV-497 did not change position.

Choice C:
Distractor: Plausible if Applicant does not fully understand system design. Incorrect because heat exchanger valves open and HCV-497 closes.

Choice D:
Distractor: Plausible if Applicant does not fully understand system design. Incorrect because heat exchanger valves open.

KA#:	008000 A3.08	Bank Ref #:	07-11-06 012
LP# / Objective:	0711-06 01.05	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	MODIFIED
Reference:	LP 0711,06	Handout:	NONE

••Location

2.80 The CCW inlet and outlet isolation valves are located near their respective heat exchangers. HCV-489A/B and HCV-490A/B are at the east end of Corridor 4 by heat exchangers AC-1A and AC-1B. HCV-491A/B and HCV-492A/B are located in Room 18 with heat exchangers AC-1C and AC-1D.

••Power Supplies

2.81 Not applicable

••Instrumentation and Controls

2.82 The following is a detailed description of the Instrumentation and Controls for the Component Cooling Water Heat Exchanger Isolation valves and Bypass valve.

•••Local

2.83 The heat exchanger isolation valves are equipped with a hand jack that is used to prevent automatic opening of the valves during heat exchanger maintenance. Section 3.0 has details on their operation.

•••Remote

2.84 The isolation valves HCV-489A/B through HCV-492A/B (eight valves) for the CCW heat exchangers are operated in pairs (inlet and outlet valves) from CB-1/2/3. Each pair of valves has a control switch with three positions (CLOSED/NORMAL/OPEN).

2.85 The CCW heat exchangers can be bypassed by remotely operated valve HCV-497 with a hand controller on panel AI-45. The valve can be positioned from zero to 100% open.

•••Interlocks

2.86 A safety injection actuation signal (SIAS) automatically opens the CCW heat exchanger inlet/outlet valves and closes the bypass valve. However, if CCW is lost during the accident, the signal may be overridden to a selected set of CCW isolation valves (four) and heat exchangers (two), so that the RW interface valves can be manually opened to supply backup cooling. The signal may also be overridden to the bypass valve.

2.87 The control switch HC-4350 overrides SIAS to the CCW isolation valves for AC-1A and AC-1C. The control switch HC-4351 overrides SIAS to the valves for AC-1B and AC-1D. Both override switches are on CB-1/2/3 and have two positions (NORMAL/ OVERRIDE).

2.88 The control switch HC-497 can override SIAS to the CCW Heat Exchanger Bypass Valve or can close the bypass valve. This switch is located on CB-1, 2, 3 and has three positions (OVERRIDE, NORMAL/CLOSE).

QUESTIONS REPORT
for ILO EXAM BANK

07-11-06 012

Which one of the following describes the expected response of the CCW Heat Exchanger Bypass Valve (HCV-497) to an SIAS?

A. Goes fully open.

B.✓ Goes fully closed.

C. Closes to minimum flow (15% open).

D. Does not change position.

QUESTION NUMBER: 036

Given the following plant conditions:

- The plant is operating at full power
- Channel PRC-103X is selected as the controlling channel and is in automatic
- RPS Channel “B” TMLP Trip unit is in bypass
- RPS Channel "C" High Pressurizer Pressure Trip Unit is in bypass

Assuming NO operator action, which of the following combinations of pressurizer pressure instrument failures would result in a reactor trip?

A✓ Pressurizer Control Channel, PT-103X, and Pressurizer Safety Channel, A/PT-102, both fail high.

B. Pressurizer Safety Channel, A/PT-102, and DSS Pressure Channel, B/PT-120, both fail high.

C. DSS Pressure Channels, C/PT-120 and D/PT-120, both fail low.

D. Pressurizer Control Channel, PT-103Y, and Pressurizer Safety Channel, D/PT-102, both fail low.

Question # 36 Rev 0

KA #: 010000 K1.01 Tier 2 Group 1: Pressurizer Pressure Control System
Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: RPS
Importance 3.9 / 4.1

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective
When given specific plant conditions, EXPLAIN operating principles to predict response of Reactor Coolant System (RCS) Instrumentation.

Question Pedigree
New question.

K/A Fit:
Question address pressurizer control channel failure that results in a RPS reactor trip.

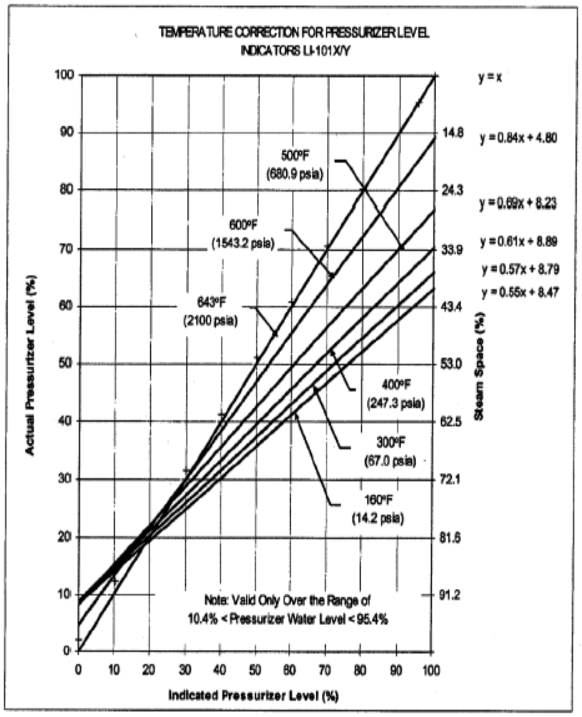
Choice A:
Correct answer: Since PRC-103X is selected as the controlling channel, if PT-103X fails high, pressurizer spray valves will open and lower pressure to the RPS TMLP setpoint. With B TMLP channel bypassed and A pressure channel failed high, TMLP channels C & D will still provide the 2/4 logic required for a reactor trip.

Choice B:
Distractor: Plausible because 2 RPS channels have failed high. Incorrect because only normal RPS channel has failed and only 1 DSS channel has failed.

Choice C:
Distractor: Plausible because if these two channels fail high a diverse scram trip will result. Incorrect because both channels failed low and there is no DSS low pressure trip.

Choice D:
Distractor: Plausible because the answer would be correct if PRC-103Y was selected as the controlling channel. Incorrect because PRC-103X is the controlling channel.

KA#:	010000 K1.01	Bank Ref #:	N/A
LP# / Objective:	0712-20 04.00	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	NEW
Reference:	STM 36	Handout:	NONE



Commitment ID # 790101

FIGURE 2-8: LI-101 X & Y TEMPERATURE CORRECTION (TDB.111.1.A)

2.101 The control channels function to maintain pressurizer level at the setpoint. The setpoint is developed by the RRS. RRS A provides the setpoint for channel X, and RRS B provides the setpoint for channel Y. The level control functions respond to deviation from setpoint. **NOTE**: The T_{ave} signal from the RRS is hard wired and, therefore, not selectable with the A/B selector switch on CB-4.

2.102 There are six control channels of pressurizer pressure: four are safety channels and two are control channels. The control channels provide input to the ERF computer and pressurizer heater controls (automatic control rod positioning has been disabled). The safety channels provide input for RPS trips (high-pressure and TM/LP) and for PPLS. The only instrumentation on the pressurizer that is safety related is the pressurizer pressure instrumentation.

2.103 The four pressurizer pressure safety channels, A/B/C/D/PT-102, each have a transmitter and amplifier in a current loop with three voltage dropping resistors. The pressurizer pressure sigma meters have an internal voltage dropping resistor. The voltage dropped across the resistors is used for the high pressurizer pressure trip, TM/LP trip and PPLS.

2.104 The safety channels, A/B/C/D/PT-120, provide input for the Diverse Scram System (DSS). The pretrip is set at 2400 psia, and the trip is set at 2450 psia. The instrument range is from 1900 to 2900 psia. A/PI-120 provides pressure indication on AI-66A, and B/PI-120 provides pressure indication on AI-66B.

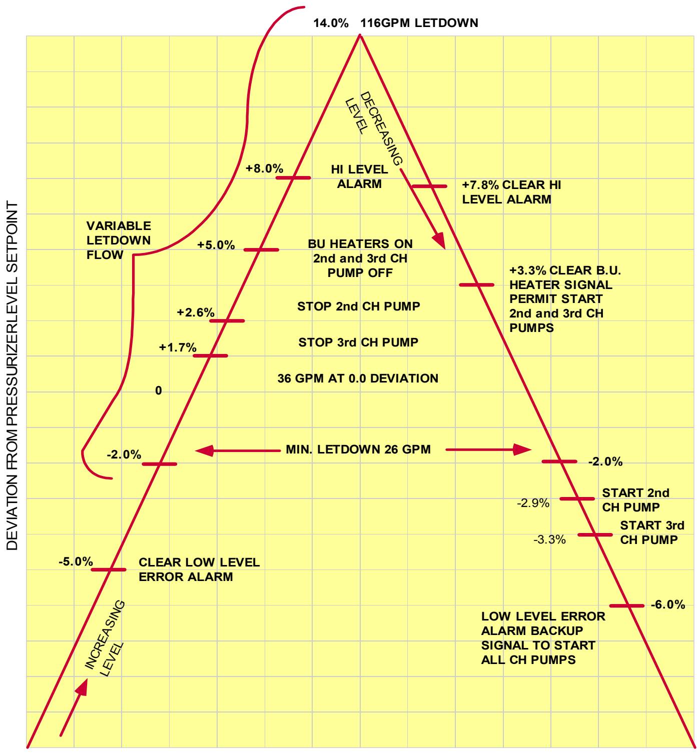


FIGURE 2-9: LEVEL CONTROL CHRISTMAS TREE

2.105 Safety channels A/B/C/D/PT-102 provide the following interlocks/controls. At 1700 psia decreasing, the pressurizer pressure safety channels allow a block of the PPLS signal. A PPLS is generated by safety channels at 1600 psia decreasing if not previously bypassed and actuates the LO-LO PRESSURE alarm. Safety channels also provide a TM/LP trip at a variable setpoint, a HIGH PRESSURE PRETRIP alarm at 2300 psia, and a high pressure trip and PORV actuation at 2350 psia.

2.106 Each safety channel pressure (A/B/C/D/PIA-102) is displayed on CB-1/2/3 from 1500 to 2500 psia. A signal from each channel is supplied to the computer. Meters A/B/C/D/PIA-102X display the TM/LP setpoints generated by the RPS.

2.134 Two single-pen pressure recorder controllers, PRC-103X and PRC-103Y, are located on CB-1/2/3. Channel selector switch HC-103 selects which channel controls the heaters and spray valves. The setpoint, nominally 2100 psia, is adjustable by depressing the respective setpoint pushbutton (up/down) on each pressure controller. The selected controller generates an analog signal that modulates the proportional heater power and spray valve opening. A HI-LO PRESSURE alarm actuates at 2145 psia (HI) or 2080 psia (LO).

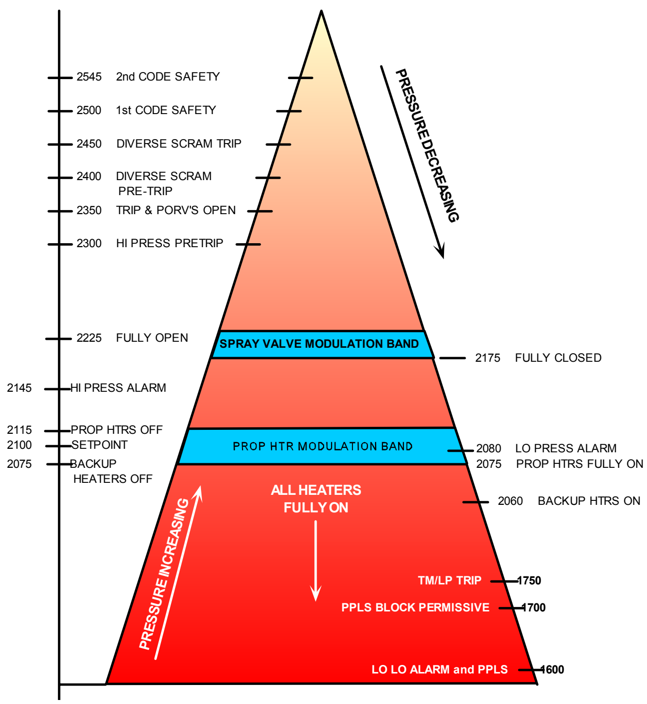


FIGURE 2-12: RCS PRESSURE CONTROL WITH 2100 PSIA SETPOINT

2.135 Low-range pressure channel P-118 provides Control Room indication on CB-1/2/3 from zero to 1600 psia. The low-range pressure channel actuates a close signal to the shutdown cooling suction valves if pressure exceeds 300 psia. P-118 must be energized to allow opening of HCV-347 and HCV-348. A loss of power override switch is provided to allow shutdown cooling operation during a loss of power from AI-40B. The redundant shutdown cooling valve interlock is provided by P-115, which requires power to provide the interlock, so no override switch is needed.

QUESTION NUMBER: 037

Given the following plant conditions:
- A Reactor Coolant System (RCS) cooldown is in progress for a Refueling Outage
- Shutdown Cooling is in operation per OI-SC-1
- RCS temperature is 120°F
- RCS pressure is 235 psia
- Reactor Coolant Pumps, RC-3A and RC-3B, are operating.
- Pressure is being controlled manually using Main pressurizer spray and heaters.

What actions should be taken before shutting down Reactor Coolant Pumps, RC-3A and RC-3B, in accordance with OI-RC-9?

A. RCS pressure should be lowered to less than 215 psia.

B. RCS temperature should be lowered to less than 110°F.

C. The RCS should be borated to the refueling boron concentration.

D.✓ RCS pressure control using Auxiliary spray should be established.

Question # 37 Rev 0

KA #: 010000 K6.03 Tier 2 Group 1: Pressurizer Pressure Control System
Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: PZR sprays and heaters

Importance 3.2 / 3.6
CFR Number 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
DISCUSS the prerequisites and precautions for shutdown of the RCS.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
Question addresses loss of RCS pressure control following a loss of pressurizer spray due to tripping RCP's.

Choice A:
Distractor:Plausible because there are RCS pressure limitations for operation of the RCP's. Incorrect because because pressure should be maintained above 225 psia with RCPs running.

Choice B:
Distractor: Plausible if Applicant believes that are temperature restrictions on shutting down RCP's. Incorrect because there is no requirement to lower RCS temperature before tripping RCPs.

Choice C:
Distractor: Plausible if the Applicant believes RCP's are needed for RCS mixing until the RCS is at the refueling boron concentration. Incorrect because refueling boron concentration is not yet required.

Choice D:
Fort Calhoun suffered a loss of shutdown cooling from these conditions when the reactor coolant pumps were secured. Securing the pumps caused main pressurizer spray flow to be lost and RCS pressure increased to the setpoint where shutdown cooling loop isolation valves automatically closed. The caution in the provided reference (OI-RC-9) about establishing auxiliary pressurizer spray was added after this event.

KA#:	010000 K6.03	Bank Ref #:	07-11-20 187
LP# / Objective:	0711-20 03.06A	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	OI-RC-9	Handout:	NONE

Attachment 2 - Shutdown Reactor Coolant Pumps (Coupled)

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision No.________ Date:________ ________

2. The Reactor is Shutdown (Mode 3, Mode 4 or Mode 5). ________

3. The Zero Power Mode Bypass Switches on AI-31A/B/C/D are in bypass to disable the RPS Low Flow Trip before the first RCP is stopped. ________

PROCEDURE

NOTE

When a RCP is shutdown its seal Bleedoff temperature will rise.

CAUTION

Failure to establish pressure control using Auxiliary PZR Spray prior to securing all Reactor Coolant Pumps may result in Reactor Coolant System pressure increasing and potentially result in a Loss of Shutdown Cooling if pressure exceeds the HCV-347/348, SHUTDOWN COOLING LOOP 2 ISOLATION VALVES interlock setpoint value of 300 psia.

1. Secure the selected RCP by placing its Control Switch to AFTER STOP:

   - RC-3A, RC Pump ___
   - RC-3B, RC Pump ___
   - RC-3C, RC Pump ___
   - RC-3D, RC Pump ___ ________

QUESTION NUMBER: 038

Given the following plant conditions:

- Bus Power Failure DC Distribution Panel 1 Light is off
- "DC BUS#1 LOW VOLT" annunciator is in alarm
- DC Bus #1 Voltage indicates 0 volts
- Power has been lost to the "A" DSS 86-Relay

How will operation of the Diverse Scram System (DSS) be affected?

A. The DSS will initiate a Reactor Trip.

B. The DSS will go to a "1 of 3" Trip Logic.

C.✓ The "A" DSS 86-Relay will not be capable of generating a Reactor Trip.

D. The DSS will not be capable of generating a Reactor Trip.

Question # 38 Rev 0

KA #: 012000 K2.01 Tier 2 Group 1: Reactor Protection System
Knowledge of bus power supplies to the following: RPS channels, components, and interconnections
Importance 3.3 / 3.7

CFR Number 55.41(b)(6)
Design, components,and functions of reactivity control mechanisms and instrumentation.

Fort Calhoun Objective
DESCRIBE the effects of a loss of power on the Diverse Scram System.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
Question addresses a loss of a DC power supply to the DSS portion of the RPS.

Choice A:
Distractor: Plausible if Applicant believes a loss of DC bus will initiate a 2/4 reactor trip. Incorrect because DSS is an energize to trip system

Choice B:
Distractor: Plausible if the Applicant believes that 1 of the 4 inputs are tripped. Incorrect because the power for the 2/4 logic comes from the AC instrument buses.

Choice C:
Correct answer: DSS is an energize to trip system. There are two independent trains, each powered by a different DC bus. With power to the "A" train lost, it would not be capable of generating a reactor trip, but the "B" train would be.

Choice D:
Distractor: Plausible if the Applicant believes a loss of DC bus #1 affects both the "A" and "B" DSS trains. Incorrect because the "B" DSS 86 Relay could still produce a trip.

KA#:	012000 K2.01	Bank Ref #:	07-12-25
LP# / Objective:	0712-25 05.06	Exam Level:	RO-6
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM-38	Handout:	NONE

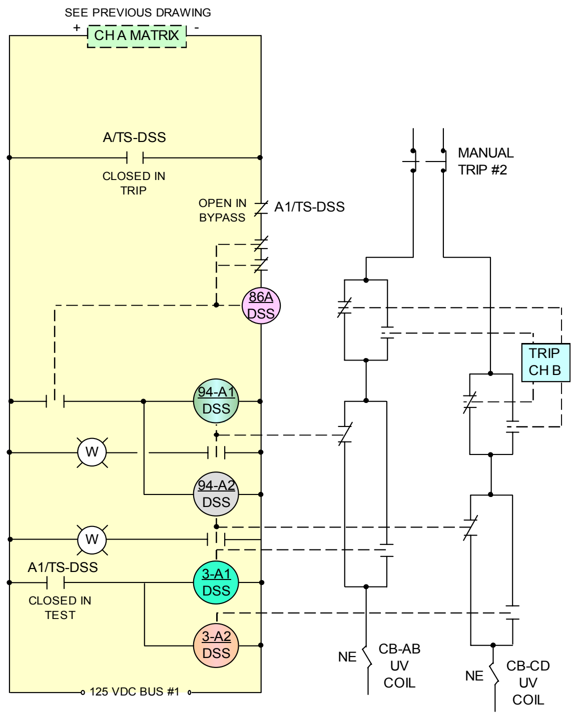


FIGURE 2-43: CHANNEL A TRIP CIRCUIT

2.298 The 86A/DSS relay is normally de-energized. If a trip occurs, 86A/DSS energizes to close a contact that energizes 94-A1/DSS and 94-A2/DSS, opening contacts to remove power to the undervoltage coil of both CB-AB and CB-CD. This removes power to the CEDM clutches, tripping the reactor.

2.299 Without 125 VDC power, a reactor trip from the DSS is not possible.

2.300 Either matrix A or B of the DSS is capable of tripping the reactor.

2.301 The following conditions would be indicative of a 125 VDC power failure:

- Loss of matrix lights
- Loss of the 86A/DSSS (or B) LOCKOUT RELAY light
- The DSS 86A/DSS (or B) TROUBLE alarm actuates.

QUESTION NUMBER: 039

With the plant operating at 100% power, pressurizer pressure instrument A/PIA-102Y failed low.

What effect does this have on the Engineered Safeguards Actuation System?

A. PPLS Matrix A is in 1 out of 3 logic, PPLS Matrix B is unaffected.

B. PPLS Matrix A is in 2 out of 3 logic, PPLS Matrix B is unaffected.

C.✓ Both PPLS Matrix A and PPLS Matrix B are in 1 out of 3 logic.

D. Both PPLS Matrix A and PPLS Matrix B are in 2 out of 3 logic.

Question # 39 Rev 0

KA #: 013000 K6.01 Tier 2 Group 1: Engineered Safety Features Actuation System
Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors
Importance 2.7 / 3.1

CFR Number: 55.41(b)(8)
Components, capacity, and functions of emergency systems.

Fort Calhoun Objective:
EXPLAIN how ESC signals respond to loss of power or sensor failures.

Question Pedigree
Bank question.

K/A Fit:
Question addresses how a loss of a pressure detector would affect the engineered safeguards features actuation system logic.

Choice A:
Distractor: Plausible if Applicant believes A channel detector only affects the "A" matrix. Incorrect because it affects both the A matrix and the B matrix.

Choice B:
Distractor: Plausible if Applicant believes A channel detector only affects the "A" matrix and the channel is bypassed instead of tripped. Incorrect because it affects both the A matrix and the B matrix and the channel is tripped.

Choice C:
Correct answer: Failure of the detector trips the channel placing both the A matrix and the B matrix in 1/3 logic.

Choice D:
Distractor: Plausible if the Applicant believes the channel is bypassed. Incorrect because the channel is tripped.

KA#:	013000 K6.01	Bank Ref #:	07-12-14 055
LP# / Objective:	0712-14 02.04	Exam Level:	RO-8
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM-19	Handout:	NONE

2.8 The following components are located on AI-30A/B:

- CPHS lockout relays
- CPHS test switches
- PPLS test switches
- SIAS lockout relays and backup SIAS lockout relays
- CPHS/SIAS test switch keys
- RAS test switches

2.9 The following auxiliary SIAS relays are located inside main control board CB-4:

- A/94-1/SIAS
- A/94-2/SIAS
- B/94-1/SIAS
- B/94-2/SIAS
- B/94-5/SIAS

2.10 43-SIAS/FW2 and 43-SIAS/FW4 are located on CB-10/11.

2.11 The load shed test switch CS-A(B)/LST is located in AI-109A(B).

••Power Supplies

2.12 Power supplies for engineered safeguards components such as an indicator relay or lockout relay are fed from their respective AC or DC engineered safeguards power supply. For example, pressurizer meter relay A/PIA-102Y is fed from 120 VAC instrument bus AI-40A, and the "B, C and D" relays are fed from their respective bus (AI-40B/C/D). Similarly, lockout relay A/94-3/SIAS is fed from 125 VDC bus AI-41A, and the "B" relay is fed from AI-41B.

••Instrumentation and Control

2.13 The following describes the instrumentation and controls for the safety injection actuation signal (SIAS).

2.14 Four pressurizer pressure meter relays, A/B/C/D/PIA-102Y provide signals for PPLS. These pressurizer pressure safety channel current loops are the same instrumentation loops that serve the RPS thermal margin/low pressure (TM/LP) trip and RPS high pressurizer pressure reactor trip and PORV control circuits. Each pressure meter relay in CB-1,2,3 provides pressurizer pressure indication, and actuates 2 auxiliary relays, one for the PPLS actuation circuitry and one for the PPLS block circuity. The 2 auxiliary relays provide a PRESSURIZER SAFETY INJECTION SIGNAL LOW-LOW PRESS alarm at 1600 psia, one contact to the PPLS matrix relays, and two contacts to PPLS block circuitry.

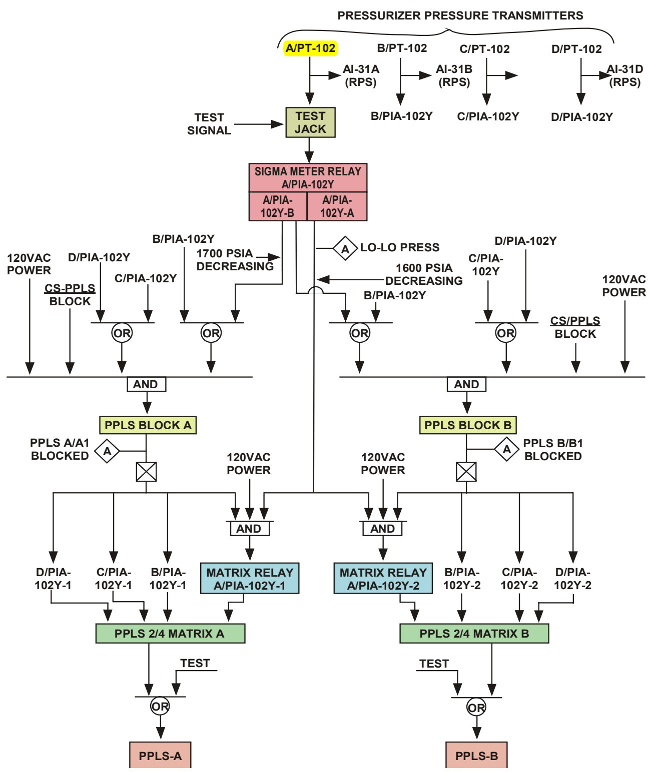


FIGURE 2-2: PPLS LOGIC

2.15 Each pressurizer pressure channel provides one contact to the PPLS Block-A circuit and one contact to the PPLS Block-B circuit (via the "B" auxiliary block relay). PPLS is manually blocked during shutdown to allow cooldown and depressurization of the RCS. Closure of contacts from channels A or B and channels C or D allows the operator to block PPLS when pressure decreases below 1700 psia. When pressure increases above 1700 psia, the PPLS block is removed automatically, or may be manually removed at any time by the operator.

2.16 The key for the PPLS block switch is normally captive in the key holder. When in the REMOVE position, the key holder provides a PPLS BLOCK CIRCUIT KEY REMOVED alarm on CB-1,2,3. The CS-PPLS block switch is a three-position key-operated switch with EMERGENCY RESET/NORMAL/BLOCK positions with spring return to NORMAL.

2.17 The operator blocks PPLS by momentarily turning the key switch to the BLOCK position. Operating the switch energizes the PPLS Block A and PPLS Block B relays if pressurizer pressure is less than 1700 psia. Turning the switch to the EMERGENCY RESET position unblocks PPLS.

QUESTION NUMBER: 040

The plant is operating at full power
Detector well cooling fan VA-12A is out of service due to maintenance on its breaker.
The "480 VOLT OVERLOAD TRIP" annunciator is in alarm
Detector well cooling fan VA-12B tripped and cannot be restarted.

Using the attached references and the following data, determine the time available until the plant must be in Hot Shutdown.

TI-732A	**130°**	**TI-735A**	**155°**
TI-733A	**132°**	**TI-736A**	**146°**
TI-733B	**139°**	**TI-736B**	**142°**
TI-734A	**155°**	**TI-737A**	**139°**

A. 9-11 minutes

B. 12-14 minutes

C.✓ 15-17 minutes

D. 18-20 minutes

Question # 40 Rev 0

KA #: 022000 K3.02 Tier 2 Group 1: Containment Cooling System
Knowledge of the effect that a loss or malfunction of the CCS will have on the following:
Containment instrumentation readings
Importance 3.0 / 3.3

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Given the Operating Instruction OI-VA-1, explain how to calculate the length of time allowed to operate without Nuclear Detector Well Cooling.

Question Pedigree
Bank question.

K/A Fit:
Question addresses containment temperature readings following a loss of detector well cooling fans that are a part of the containment cooling system.

Choice A:
Distractor: Plausible if Applicant uses the references incorrectly. Incorrect because the time is too short.

Choice B:
Distractor: Plausible if Applicant uses the references incorrectly. Incorrect because the time is too short.

Choice C:
Correct answer: the time available is approximately 16 minutes.

Choice D:
Distractor: Plausible if Applicant uses the references incorrectly. Incorrect because the time is too long.

Handouts: OI-VA-1 Attachment 3D and TDB III.24

KA#:	022000 K3.02	Bank Ref #:	07-14-05
LP# / Objective:	0714-05 03.00	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	OI-VA-1, TDB III 26	Handout:	OI-VA-1, TDB III 24

Attachment 3B - Concrete H/U Determination All NDWC Lost

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

Revision No. ___________ Date:___________ ________

PROCEDURE

NOTE

TIC-732A is the most representative of bulk concrete temperature and should be used if operable. TIC-734A and TIC-736A are conservatively high and should be used only if TIC-732A is inoperable. TIC-733B and TIC-735A read more conservatively high than TIC-734A and TIC-736A and should be used only if TIC-732A, TIC-734A, and TIC-736A are inoperable.

CAUTION

Concrete Temperature Heatup determination should be performed as soon as possible upon loss of NDWC.

1. Record the following temperatures: (AI-44)

   - TIC-732A ___
   - TIC-733B ___
   - TIC-734A ___
   - TIC-735A ___
   - TIC-736A ___ ________

2. Determine the concrete temperature using TIC-732A per TDB Figure III.24. ________

3. IF TIC-732A is inoperable,
   THEN determine the concrete temperature using TIC-734A and/or TIC-736A. ________

4. IF TIC-732A, TIC-734A, and TIC-736A are inoperable, THEN determine the concrete temperature using TIC-733B and/or TIC-735A. ________

Attachment 3B - Concrete H/U Determination All NDWC Lost

PROCEDURE (continued) (✓) INITIALS

5. Determine the time remaining before the Plant must be shutdown by performing the following:

   - 148°F - ___________°F = _________°F ÷ 1°F/min = _________ Minutes ________
     (Conc Temp)

6. Notify the Shift Manager of the time before Plant Shutdown. ________

Completed by ____ Date/Time _______/_______

Fort Calhoun Station
Unit 1

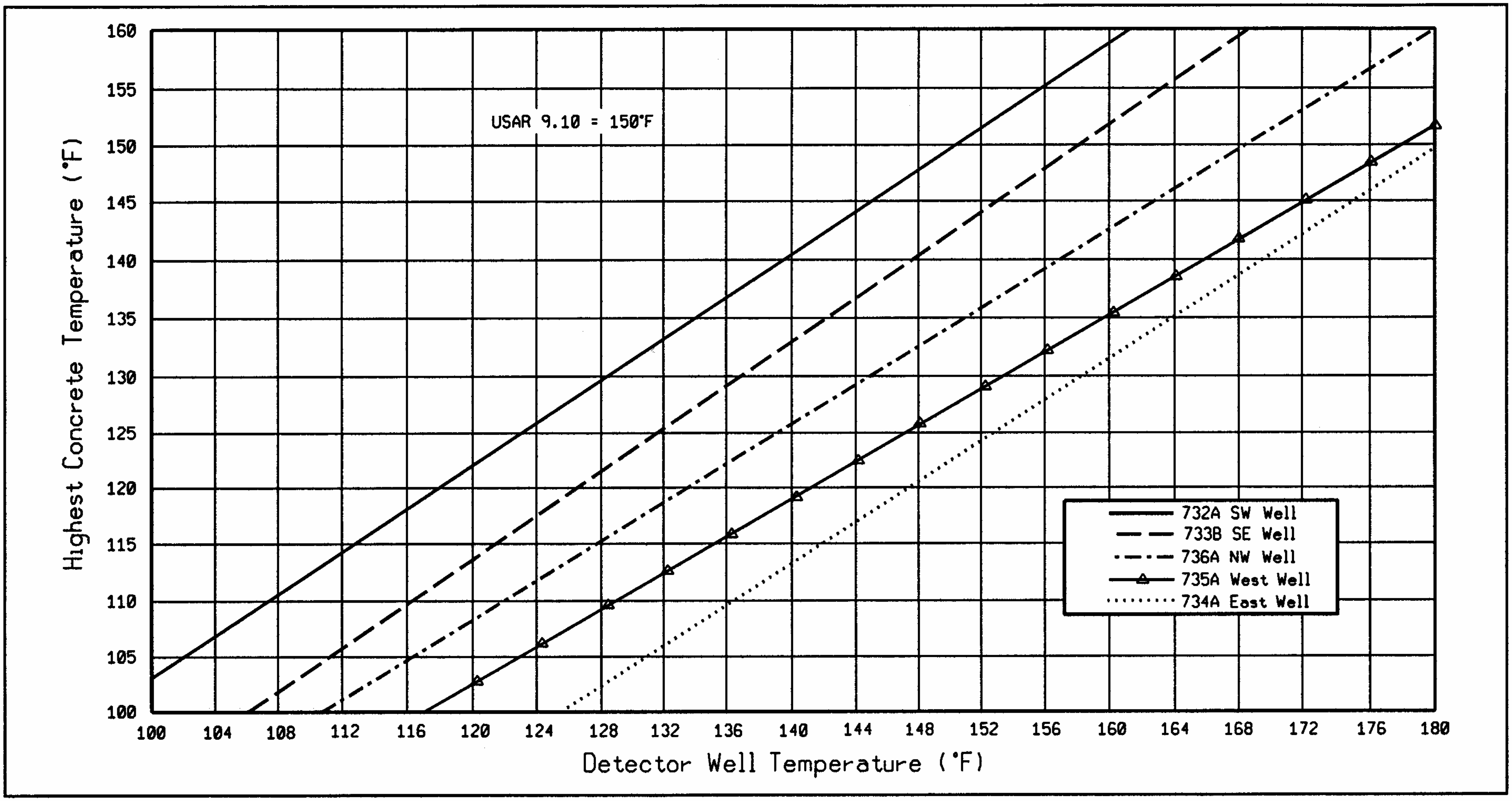
TDB-III.24

TECHNICAL DATA BOOK

NUCLEAR DETECTOR WELL TEMPERATURES

Change No.	EC 36060
Reason for Change	Converted from WordPerfect to Word.
Requestor	N/A
Preparer	J. Collier
Editorial Correction	N/A
Issue Date	04-13-05 3:00 pm

Detector Well Temperature vs Highest Concrete Temperature



NOTE: TIC-732A is the most representative of bulk concrete temperature and should be used if operable. TIC-734A and TIC-736A are conservatively high and should be used only if TIC-732A is inoperable. TIC-733B and TIC-735A read more conservatively high than TIC-734A and TIC-736A and should be used only if TIC-732A, TIC-734A, and TIC-736A are inoperable.

QUESTION NUMBER: 041

Technical Specification Limiting Conditions for Operations 2.4, "Containment Cooling" allows the reactor to be taken critical for power operation with one _________ inoperable.

A. Containment Air Cooling Unit

B. CCW Heat Exchanger

C. CCW Pump

D.✓ RW Pump

Question # 41 Rev 0

KA #: 022000 2.2.22 Tier 2 Group 1: Containment Cooling System
Knowledge of limiting conditions for operations and safety limits.
Importance 4.0 / 4.7

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
Given a copy of Technical Specifications, apply the applicable Limiting Conditions for Operation (LCO).

Question Pedigree
New question.

K/A Fit:
Question addresses LCO for containment cooling

Choice A:
Distractor: Plausible because there are 2 air cooling and filtering units and 2 air cooling units. Incorrect because all must be operable before taking the reactor critical.

Choice B:
Distractor: Plausible because there are 4 CCW heat exchangers. Incorrect because all must be operable before taking the reactor critical.

Choice C:
Distractor: Plausible because there are 3 CCW pumps. Incorrect because all must be operable before taking the reactor critical.

Choice D:
Correct answer: T.S. 2.4 does allow the reactor to be taken critical with 1 Raw Water pump inoperable.

KA#:	022000 2.2.22	Bank Ref #:	N/A
LP# / Objective:	0711-22 01.12	Exam Level:	RO-5
Cognitive Level:	LOW	Source:	NEW
Reference:	TS 2.4	Handout:	NONE

2.0 LIMITING CONDITIONS FOR OPERATION

2.4 Containment Cooling

Applicability

Applies to the operating status of the containment cooling systems.

Objective

To assure operability of equipment required to remove heat from the containment during normal operating and emergency situations.

Specifications

(1) Minimum Requirements

* a. The reactor shall not be made critical, except for low-temperature physics tests, unless all the following are met:

i. The following equipment normally associated with diesel-generator DG-1 (4.16-kV bus 1A3 and associated non-automatically transferring 480-Volt bus sections) is operable, except as noted:[(1)]

Raw water pump	AC-10A
Raw water pump	AC-10C
Component cooling water pump	AC-3A
Component cooling water pump	AC-3C
Containment spray pump	SI-3A
Containment air cooling and filtering unit	VA-3A
Containment air cooling unit	VA-7C

ii. The following equipment normally associated with diesel-generator DG-2 (4.16-kV 1A4 and associated non-automatically transferable 480 Volt bus sections) is operable, except as noted.[(1)]

Raw water pump	AC-10B
Raw water pump	AC-10D
Component cooling water pump	AC-3B
Containment spray pump	SI-3B
Containment air cooling and filtering unit	VA-3B
Containment air cooling unit	VA-7D

iii. Four component cooling heat exchangers shall be operable.

iv. All valves, piping and interlocks associated with the above components and required to function during accident conditions are operable.

(1) Reactor may be made critical with one inoperable raw water pump. LCO action statements shall apply.

2.0 LIMITING CONDITIONS FOR OPERATION
2.4 Containment Cooling (Continued)

b. During power operation one of the components listed in (1)a.i. or ii. may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. For cases involving Raw Water pump inoperability, if the river water temperature is below 60 degrees Fahrenheit, one Raw Water pump may be inoperable indefinitely without applying any LCO action statement. When the river water temperature is greater than 60 degrees Fahrenheit, an inoperable Raw Water pump shall be restored to operability within 7 days or the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable Raw Water pump is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

a. During power operation, the minimum requirements may be modified to allow a total of two of the components listed in (1)a.i. and ii. to be inoperable at any one time. (This does not include: 1) One Raw Water pump which may be inoperable as described above if the river water temperature is below 60 degrees Fahrenheit or, 2) SI-3A and SI-3B being simultaneously inoperable; or 3) VA-3A and VA-3B, or VA-7C and VA-7D, being simultaneously inoperable. Only two raw water pumps may be out of service during power operations. Either containment spray pump, SI-3A or SI-3B, must be operable during power operations. One train of the containment air cooling and filtering systems (VA-3A and VA-7C), or (VA-3B and VA-7D), must be operable during power operations). If the operability of one of the two components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. LCO 2.4(1)b. shall be applied if one of the inoperable components is restored within 24 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

b. During power operation one component cooling heat exchanger may be inoperable. If the operability of the heat exchanger is not restored within 14 days, the reactor shall be placed in a hot shutdown condition within 12 hours. If two component cooling heat exchangers are inoperable, the reactor shall be placed in hot shutdown condition within 12 hours. If the inoperable heat exchanger(s) is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

c. Any valves, interlocks and piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as for that component.

* d. Any valve, interlock or piping associated with the containment cooling system which is not included in the above paragraph and which is required

QUESTION NUMBER: 042

A large Steam Line Break in containment coincident with numerous tube ruptures in the faulted S/G has caused PPLS, CPHS and SGLS.

Following RAS, What would be the expected trends in SIRWT level (LIC-381 and LIC-382) and Containment Sump levels (LI-387 and LI-388) if the seals on one of the Containment Spray Pumps failed?

	SIRWT Level	Containment Sump Level
A.	Steady	Steady
B.✓	Steady	Lowering
C.	Rising	Lowering
D.	Lowering	Rising

Question # 42 Rev 0

KA #: 026000 A1.03 Tier 2 Group 1: Containment Spray System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment sump level
Importance 3.5 / 3.5

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
EXPLAIN what is meant by an "Interfacing system LOCA."

Question Pedigree
Bank question used on 2004 exam.

K/A Fit:
Question addresses monitoring containment sump level during containment spray operation.

Choice A:
Distractor: Plausible if Applicant does not understand sump level response due to CS pump seal failure. Incorrect because containment sump level will lower.

Choice B:
Correct answer: SIRWT level will be steady because RAS has occured. Containment sump level will be lowering because recirculating water is flowing out the CS pump seal.

Choice C:
Distractor: Plausible if Applicant does not understand that ECCS recirculation valves to the SIRWT close on RAS and does not understand the results of the CS pump seal leak.

Choice D:
Distractor: Plausible because these are the expected trends before RAS occurs. Incorrect because containment sump level will be lowering.

KA#:	026000 A1.03	Bank Ref #:	026000 017
LP# / Objective:	0715-23 01.08	Exam Level:	RO-5
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM-15	Handout:	NONE

1.2 Functions

The following is a summary of the functions associated with the Emergency Core Cooling System.

1.2.1 Safety-Related Functions

A. The ECCS is required to provide emergency core cooling following a loss of primary or secondary coolant. Portions of the ECCS equipment are used to provide shutdown cooling. Auxiliary functions of the Emergency Core Cooling System equipment include:

- Fill and drain of the refueling cavity
- Provide a backup cooling system for the Spent Fuel Pool Cooling System
- Provide a means of cooling containment spray water following a Recirculation Actuation Signal (RAS)
- Provide a means to fill and drain the safety injection tanks

B. Provide a supply of water for initial fill and flushing of the reactor coolant pump mechanical seals.

1.2.2 Non-Safety-Related Functions

None

1.3 Basic System Overview

The following is a brief overview of the Emergency Core Cooling System.

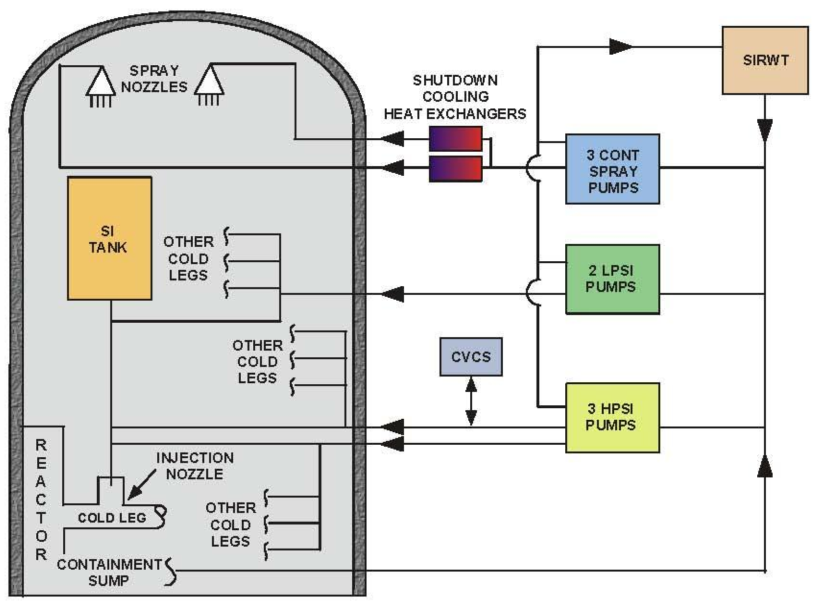
1.3.1 Flowpaths

A. During normal plant operation at power, the Emergency Core Cooling System is maintained in a standby mode with all of its components aligned for emergency operation. Upon receiving a safety injection actuation signal (SIAS), two of the three high-pressure safety injection pumps and the low-pressure safety injection pumps automatically start and the motor-operated safety injection loop isolation valves automatically open.

B. During the injection mode of operation, the safety injection pumps take a suction from the SIRWT and inject borated water into the Reactor Coolant System. The safety injection tanks are pressurized to inject borated water when the Reactor Coolant System pressure falls below their set pressure.

1.3.1 (continued)

Figure 1-2 - Safety Injection and Containment Spray Block Diagram



C. SI-3A and SI-3B containment spray pumps are started by the containment spray actuation signal (CSAS) coincident with a steam generator low pressure signal (SGLS). The CSAS is actuated by high containment pressure coincident with low pressurizer pressure. The combination of these two signals CSAS and SGLS function to limit the containment pressure rise following a MSLB.

D. Low water level in the SIRWT automatically initiates recirculation. The recirculation actuation signal (RAS) shuts down the low-pressure safety injection pumps because they are no longer needed, opens both recirculation line isolation valves to provide suction for HPSI and containment spray, and closes the suction header isolation valves and the minimum flow line isolation valves to isolate the SIRWT from the containment sump water. The HPSI pumps continue to operate with suction from the containment sump to provide cooling for the complete spectrum of break sizes. A portion of the cooled water from the Containment Spray System can be diverted to the suction of the HPSI pumps.

E. A provision is made for maintaining core cooling and preventing boric acid buildup in the core by simultaneous hot and cold leg injection of the HPSI flow in the event of a large break LOCA. The Reactor Coolant System can also be cooled down and depressurized by opening the power-operated relief valves to accelerate refilling of the RCS in the event of a small-break LOCA.

QUESTION NUMBER: 043

With the plant operating at full power, inadvertent isolation of extraction steam to FW Heater, FW-16B will result in decreased ______________.

A. FW Temperature to S/G-2B only

B.✓ FW Temperature to both S/G's

C. FW Flow to S/G-2B only

D. FW Flow to both S/G's

Question # 43 Rev 0

KA #: 039000 K1.08 Tier 2 Group 1:
Main and Reheat Steam System: Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: MFW
Importance 2.7 / 2.9

CFR Number: 55.41(b)(4)
Secondary coolant and auxiliary systems that affect the facility.

Fort Calhoun Objective:
EXPLAIN the maximum allowable consequences for the Excess Load and Loss of Feedwater Heating events.

Question Pedigree
New question

K/A Fit:
Question addresses the affect of a loss of MRSS extraction steam to a feedwater heater.

Choice A:
Distractor: Plausible if Applicant believes that B train of feedwater heaters supplies water to the B S/G. Incorrect because FW-16A and FW-16B are in parallel

Choice B:
Correct answer: FW-16A and FW-16B are in parallel

Choice C:
Distractor: Plausible if Applicant believes that B train of feedwater heaters supplies water to the B S/G and doesn't understand that FW flow is controlled downstream of where the heater drain pumps discharge joins the feedwater lines. Incorrect because FW-16A and FW-16B are in parallel

Choice D:
Distractor: Plausible if the Applicant doesn't understand that FW flow is controlled downstream of where the heater drain pumps discharge joins the feedwater lines. Incorrect because FW flow is controlled downstream of heater drain connection.

KA#:	039000 K1.08	Bank Ref #:	N/A
LP# / Objective:	0715-20 03.02	Exam Level:	RO-4
Cognitive Level:	HIGH	Source:	NEW
Reference:	STM-20	Handout:	NONE

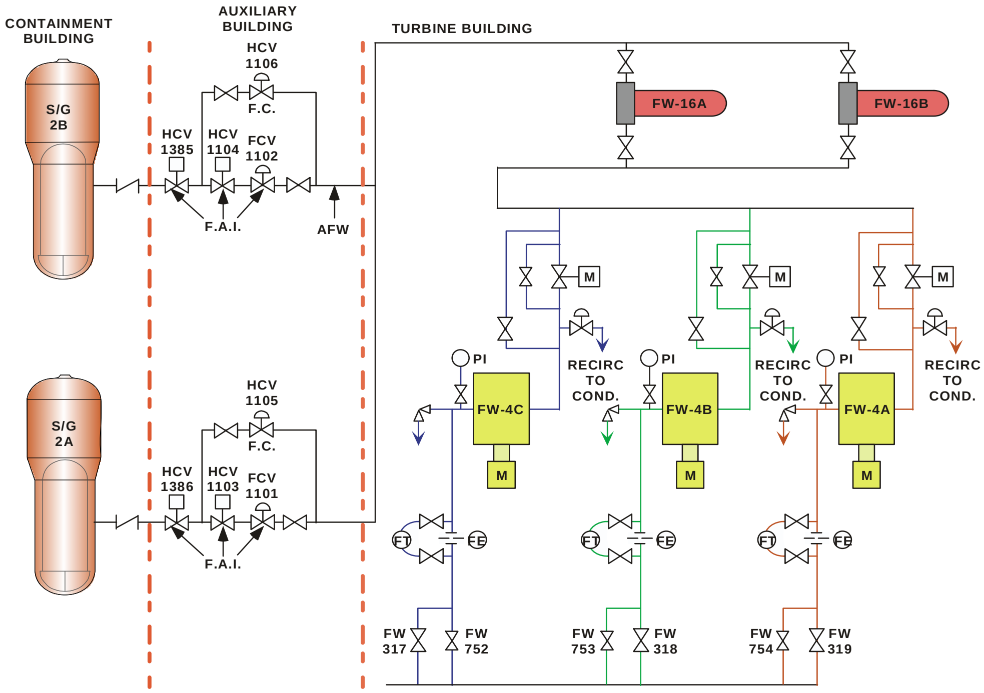


FIGURE 1-7: FEEDWATER SYSTEM

1.17 The outlet of each feed pump connects to a common header that splits to supply high-pressure feedwater heaters FW-16A and -16B in parallel. Normally each heater carries about 50 percent of the feedwater flow. Each heater is isolable and capable of carrying 80 percent of full power feedwater flow. The outlets of the high-pressure heaters combine and again split to supply the two feedwater headers, one for each steam generator.

1.18 The flow through each feedwater header passes through a flow nozzle and a feedwater regulating valve that controls the amount of feedwater to the steam generators above 15 percent power. A feedwater bypass valve around each feedwater regulating valve is used to control steam generator water level at low power levels. The feedwater then flows through two motor-operated isolation valves in each header and into the steam generators.

1.19 The heater drains are cascaded from the highest pressure to the lowest pressure heaters. One drain flow path (or train) starts with high-pressure feedwater heater FW-16A (B) draining to low-pressure heater FW-15A (B), then to heater FW-14A (B) to the heater drain tank. Four moisture separator drain tanks also discharge to the heater drain tank. The contents of the heater drain tank are pumped by the heater drain pumps back to the condensate header between low-pressure heaters FW-14A/B and FW-15A/B.

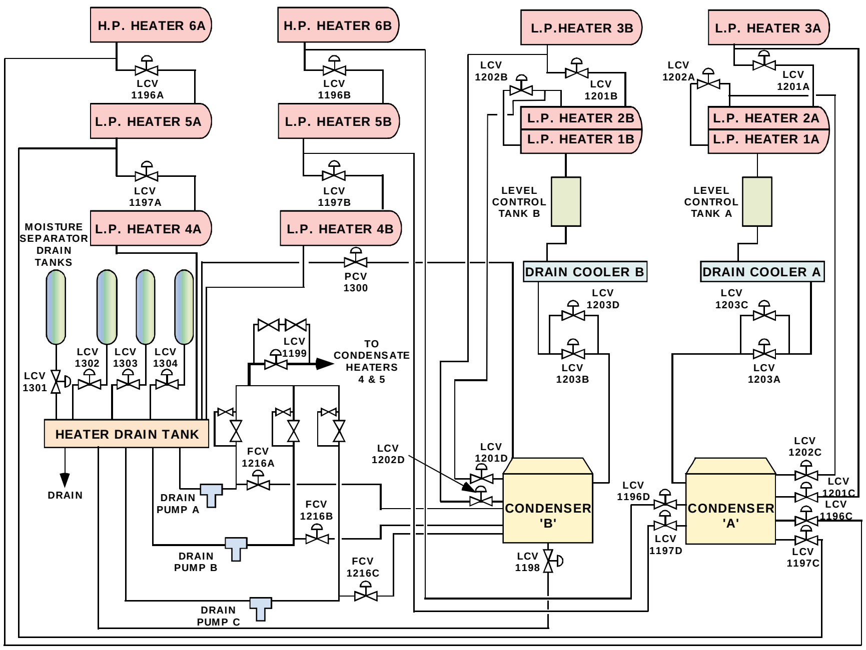


FIGURE 1-8: FEEDWATER HEATER DRAIN SYSTEM BLOCK DIAGRAM

1.20 The second drain flow path starts with low-pressure heater FW-13A (B) draining to heater FW-12A (B) then to heater FW-11A (B) and its level control tank. The level control tanks drain through the drain coolers to the main condenser. Each heater (except FW-14A/B and FW-11A/B) also has a drain path to the main condenser to dump excess water if the associated level controller fails to maintain the proper level.

1.21 The shell side of each heater is vented to the main condenser. Each heater is provided with connections that continuously vent through orifices sized to remove noncondensables while restricting steam losses to a minimum. The orifices on FW-14A/B, FW-15A/B, and FW-16A/B have a bypass valve that can be opened during plant startup to speed up the removal of noncondensible gases.

••Interfaces

1.22 The following systems interface with the Feedwater and Condensate System:

QUESTION NUMBER: 044

Which of the following conditions would result in the greatest reactivity addition due to a steam line break assuming all systems operate as designed.

A. Beginning of cycle, zero power

B. Beginning of cycle, full power

C.✓ End of cycle, zero power

D. End of cycle, full power

Question # 44 Rev 0

KA #: 039000 K5.08 Tier 2 Group 1: Main and Reheat Steam System
Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity
Importance 3.6 / 3.6

CFR Number: 55.41(b)(1)
Fundamentals of Reactor Theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Fort Calhoun Objective:
EXPLAIN the response of primary system parameters.

Question Pedigree
Bank question.

K/A Fit:
The question addresses reactivity addition due to a steam line break.

Choice A:
Distractor: Plausible because S/G inventory is greatest at zero power. Incorrect because the MTC is less negative at BOC.

Choice B:
Distractor: Plausible because most reactivity accidents analyzed in the USAR are more severe at BOC. Incorrect because the greatest reactivity addition is EOC, zero power.

Choice C:
Correct answer: The reactivity addition is greatest at EOC, zero power because the MTC is the most negative and the S/G inventory is the greatest.

Choice D:
Distractor: Plausible because the MTC is most negative at EOC. Incorrect because the S/G water inventory is lower at full power.

KA#:	039000 K5.08	Bank Ref #:	07-15-20 006
LP# / Objective:	0715-20 01.01	Exam Level:	RO-1
Cognitive Level:	LOW	Source:	BANK
Reference:	LP 07-15-20	Handout:	NONE

OUTLINE OF INSTRUCTION	COMMENTS

II. A. 4.

c. It can be seen that an excessive heat removal event has the potential to meet two of the three conditions required to produce a brittle fracture (At least in the very conservative calculations). Proper operator action will reduce the severity of the cooldown and minimize the total tensile stress during an excessive heat removal event.

5. Reactivity added by an excessive heat removal event — EO 1.5

a. The reactivity addition that occurs during an excessive heat removal event depends on the extent of the cooldown and on the moderator temperature coefficient. The extent of the cooldown depends on the type, severity and duration of the cooldown event.

b. In the case of a major steam line break, the extent of the cooldown will be much more severe if the break is not isolated by SGIS since that will allow one steam generator to blow down completely. In that case, the steam generator inventory will affect the extent of the cooldown. If the steam generator water inventory is increased, more heat will be removed from the primary side and the extent of the cooldown will be greater. Since the steam generator water inventory is greater at zero power than it is at full power, a steam line break initiated at zero power will result in a greater reactivity addition than one initiated at full power. Additional feedwater flow to the steam generator will have the same effect as increased water inventory; it will add additional reactivity.

OUTLINE OF INSTRUCTION	COMMENTS

c. An excessive heat removal event occurring at End of Cycle will result in a greater reactivity addition than one at Beginning of cycle because the moderator temperature coefficient is more negative at EOC.

B. MITIGATION

1. Potential Causes of an Excessive Heat Removal Event — EO 2.1

a. There are many different potential causes of an excessive heat removal event. We will discuss several of them.

A steam line break is the classic cooldown event from an analysis point of view. The location of the break can have a significant affect on the plant response and indications. The break locations could be:

(1) Inside containment could result in a rapid cooldown and an increase in containment pressure. The affected steam generator could not be isolated.

QUESTION NUMBER: 045

The TDSF (Time Delayed Start Failure) relay on diesel generator, DG-1, will initiate a "DIESEL START FAIL" alarm on AI-30A/A30 B-2 if:

A.✔ DG-1's speed is less than 100 rpm 10 seconds after an autostart signal

B. DG-1's speed is less than 750 rpm 10 seconds after an autostart signal

C. DG-1's speed is less than 100 rpm 15 seconds after an autostart signal

D. DG-1's speed is less than 750 rpm 15 seconds after an autostart signal

Question # 45 Rev 0

KA #: 064000 K4.05 Tier 2 Group 1: Emergency Diesel Generators
Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Incomplete-start relay
Importance 2.8 / 3.2

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
Explain abnormal operation of the EDG.

Question Pedigree
New question

K/A Fit:
The question addresses the TDSF relay that initiates the Diesel Start Failure alarm.

Choice A:
Correct answer: The TDSF relay initiates the diesel start failure alarm if the D/G speed is less than 100 rpm, 10 seconds after an autostart signal.

Choice B:
Distractor Plausible because the time is correct and the D/G's field flashes at 750 rpm. Incorrect because the TDSF speed setpoint is 100 rpm.

Choice C:
Distractor: Plausible because the speed is correct and the diesel lube oil pressure low alarm has a time delay of 15 seconds after an autostart. Incorrect because the TDSF relay has a 10 second time delay.

Choice D:
Distractor: Plausible because the D/G's fiel flashes at 750 rpm and the diesel lube oil pressure low alarm has a time delay of 15 seconds after an autostart. Incorrect because the values for the TDSF relay are 10 seconds and 100 rpm.

KA#:	064000 K4.05	Bank Ref #:	N/A
LP# / Objective:	0713-05 01.15	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	NEW
Reference:	ARP-AI-30A/A30	Handout:	NONE

Panel: **AI-30A**	Annunciator: **A30**	Window: **B-2**

DIESEL GENERATOR #1 FAILED TO START

SAFETY RELATED

DIESEL START FAIL

Tech Spec References: 2.7

Initiating Device TDSF Relay   Setpoint <100 rpm after 10 sec   Power DP2-D1

OPERATOR ACTIONS

1. Verify DG-1 not running.

   1.1 IF DG1 is not running, THEN check the following:

   - DG-1 Starting Air Pressures
   - DG-1 Control Switches for proper position
   - Any DG-1 Trip Signals

   1.2 IF DG-1 is required for emergency operations, THEN attempt to start DG-1 with the Manual Diesel Emergency Start pushbutton.

   1.3 IF DG-1 is declared inoperable, THEN test DG-2 according to Technical Specifications 2.7.

   1.4 Notify System Engineer of failure to start on demand.

PROBABLE CAUSES

- Low Starting Air Pressure
- DG-1 Switch(es) not aligned for Automatic operation
- DG-1 Trip Signal present if not an Emergency Start

REFERENCES

B120F14501 Sh 1 17396   B120F14501 Sh 2 17397   161F597 Sh 9 09809

QUESTION NUMBER: 046

According to Tech Spec 2.5, "Emergency Feedwater Storage Tank," FW-19, must contain a minimum of ________ gallons of water to ensure that reactor decay heat can be removed for 8 hours.

A. 50,000

B.✓ 55,000

C. 60,000

D. 65,000

Question # 46 Rev 0

KA #: 061000 K5.02 Tier 2 Group 1: Auxiliary / Emergency Feedwater System
Knowledge of the operational implications of the following concepts as the apply to the AFW: Decay heat sources and magnitude
Importance 3.2 / 3.6

CFR Number: 55.41(b)(4)
Secondary coolant and auxiliary systems that affect the facility.

Fort Calhoun Objective:
Given a copy of the Technical Specifications, INTERPRET the requirements for the AFW System.

Question Pedigree
New question

K/A Fit:
Question discusses the amount of AFW needed to remove decay heat for 8 hours.

Choice A:
Distractor: Plausible because 50,000 gallons is a reasonable value. Incorrect because tech spec 2.5 requires 55,000 gallons.

Choice B:
Correct answer: Tech spec 2.5 requires 55,000 gallons based on removing decay heat for 8 hours.

Choice C:
Distractor: Plausible because 60,000 gallons is a reasonable value. Incorrect because tech spec 2.5 requires 55,000 gallons.

Choice D:
Distractor: Plausible because 65,000 gallons is a reasonable value. Incorrect because tech spec 2.5 requires 55,000 gallons.

KA#:	061000 K5.02	Bank Ref #:	N/A
LP# / Objective:	0711-01 01.05	Exam Level:	RO-4
Cognitive Level:	LOW	Source:	NEW
Reference:	TS 2.5	Handout:	NONE

2.0 LIMITING CONDITIONS FOR OPERATION

2.5 Steam and Feedwater Systems

Applicability

When steam generators are relied upon for reactor coolant system heat removal.

NOTE: When heating the reactor coolant above 300°F the steam driven auxiliary feedwater (AFW) pump is only required to be OPERABLE prior to making the reactor critical.

Objective

To define certain conditions for the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

* (1) Two AFW trains shall be OPERABLE when T_{cold} is above 300°F.

A. With one steam supply to the turbine driven AFW pump inoperable, restore the steam supply to OPERABLE status within 7 days and within 8 days from discovery of failure to meet the LCO.

B. With one AFW train inoperable for reasons other than condition A, restore the AFW train to OPERABLE status within 24 hours.

C. If the required action and associated completion times of condition A or B are not met, then the unit shall be placed in MODE 2 in 6 hours, in MODE 3 in the next 6 hours, and less than 300°F without reliance on the steam generators for decay heat removal within the next 18 hours.

D. With both AFW trains inoperable, then initiate actions to restore one AFW train to OPERABLE status immediately. Technical Specification (TS) 2.0.1 and all TS actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.

(2) The motor driven train is required to be OPERABLE when T_{cold} is below 300°F and the steam generators are relied upon for heat removal. With the motor driven AFW train inoperable, then initiate actions to restore one AFW train to OPERABLE status immediately. Technical Specifications (TS) 2.0.1 and all TS actions requiring MODE changes are suspended until one AFW train is restored to OPERABLE status.

(3) A minimum of 55,000 gallons of water in the emergency feedwater storage tank (EFWST) and a backup water supply to the emergency feedwater storage tank shall be available. With the EFWST inoperable, verify operability of the backup water supply within four hours and once per 12 hours thereafter, and restore the EFWST

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.5 Steam and Feedwater Systems

to OPERABLE status within 24 hours. If these action requirements cannot be satisfied, then the unit shall be placed in at least MODE 3 within 6 hours, and less than 300°F without reliance on the steam generators for decay heat removal within the next 18 hours.

(4) The main steam stop valves are OPERABLE when T_{cold} is above 300°F and capable of closing in four seconds or less under no-flow conditions.

Basis

A reactor shutdown from power requires a removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generator is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety and atmospheric dump valves. Either auxiliary feed pump can supply sufficient feedwater for removal of decay heat from the plant. Technical Specification 2.1.1 establishes when the steam generators are required for heat removal. Each train includes the pump, piping, instruments, and controls to ensure the availability of an OPERABLE flow path capable of taking suction from the EFWST and delivering water to the steam generators. The eight day completion time for 2.5(1)A provides a limit in the maximum time allowed for any combination to be inoperable during any continuous failure to meet the LCO. With one of the required AFW trains inoperable, actions must be taken to restore OPERABLE status within 24 hours. With no AFW trains OPERABLE the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting cooldown with nonsafety grade equipment. In such a condition the unit should not be perturbed by any action, including a power change, that might result in a trip.

The minimum amount of water in the emergency feedwater storage tank is the amount needed for 8 hours of such operation. The tank can be resupplied with water from the raw water system.[1]

A closure time of 4 seconds for the main steam stop valves is considered adequate time and was selected as being consistent with expected response time for instrumentation as detailed in the steam line break analysis.[2][3]

References

(1) USAR, Section 9.4.6

(2) USAR, Section 10.3

(3) USAR, Section 14.12

QUESTION NUMBER: 047

Following a loss of DC bus #2, Auxiliary Feedwater Pump _________ cannot be started from the control room until _______________.

A. FW-6, the alternate DC power supply is selected.

B.✔ FW-10, the alternate DC power supply is selected.

C. FW-6, power is restored to DC bus #2

D. FW-10, power is restored to DC bus #2

Question # 47 Rev 0

KA #: 061000 A2.03 Tier 2 Group 1: Auxiliary / Emergency Feedwater System
Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc power
Importance 3.1 / 3.4

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective
List the primary (preferred) and alternate (if any) power supplies to each bus/component.

Question Pedigree
New question

K/A Fit:
Question addresses impact of a loss of a DC bus on the AFW system and mitigating actions.

Choice A:
Distractor: Plausible if Applicant believes that FW-6 breaker control power comes from DC bus 2. Incorrect because breaker control power of FW-6 comes from DC bus number 1.

Choice B:
Correct answer. DC bus 2 supplies DC power to FW-10. A switch is provided to transfer FW-10 DC power to DC bus 1.

Choice C:
Distractor: Plausible if Applicant believes that FW-6 breaker control power comes from DC bus 2 and doesn't realize breaker control power can be transfered to the other DC bus.. Incorrect because breaker control power for FW-6 comes from DC bus number 1.

Choice D:
Distractor: Plausible if Applicant does not realize the DC power to FW-10 can be transfered to DC-bus 1. Incorrect because it can be transfered.

KA#:	061000 A2.03	Bank Ref #:	N/A
LP# / Objective:	0713-04 01.03	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	NEW
Reference:	AFW STM	Handout:	NONE

- FW-10 TURBINE DRIVEN FEEDWATER PUMP FAILED TO START alarm actuates if valve YCV-1045 fails to fully open via time delay relay 74-3/1045.
- FW-10 TURBINE DRIVEN FEEDWATER PUMP IN TEST OR CONTROLS OFF NORMAL alarm actuates via HC-1045 if the hand control is not in AUTO or via auto start relay test switch TS-1045 if the switch in not in the NORMAL position.
- The FW-10 INLET SYSTEM LINE LOW PRESSURE alarm actuates via pressure switch PS-923 if the pressure in the steam line lowers to 400 psig.

2.36 The 40-psig pressure switch that de-energizes the auxiliary lube oil pump also illuminates a red light (PUMP FW-10 RUNNING) on panel AI-179.

2.37 A white light on panel AI-179 is illuminated if an overpressure condition occurs on the discharge header for AFW pump FW-10.

2.38 PI-5062 shaft driven oil pump LO-54 discharge pressure indicator (located on the southeast side of the machine near the mounting base). This gauge has no isolation valve and indicates pressure whenever FW-10 is running.

2.39 PI-5064 DC powered lube oil pump LO-39 discharge pressure gauge. This gauge is located on the front of the machine and is normally isolated.

•••Local Instrumentation

2.40 PI-5062 Shaft driven oil pump LO-54 discharge pressure indicator (Located on the wall side of the machine near the mounting base.) This gauge has no isolation valve and indicates pressure whenever FW-10 is running.

2.41 PI-5064 DC powered lube oil pump LO-39 discharge pressure gauge. This gauge is located on the front of the machine and is normally isolated.

••Failure Modes

2.42 The turbine driver for FW-10 is a reliable unit that operates without AC power with a wide range of steam pressures. During normal operation, DC power is used to start the oil pump which pressurizes the oil system to open the governor steam admission valve.

2.43 If the preferred source of DC power were to fail, the alternate bus can be selected from a switch in the inside back of AI-179 in the penetration area. In the unlikely event that both DC sources were to fail and FW-10 did not start when YCV-1045 and YCV-1045A/B fail open, a pry bar may be used to locally jack up the steam

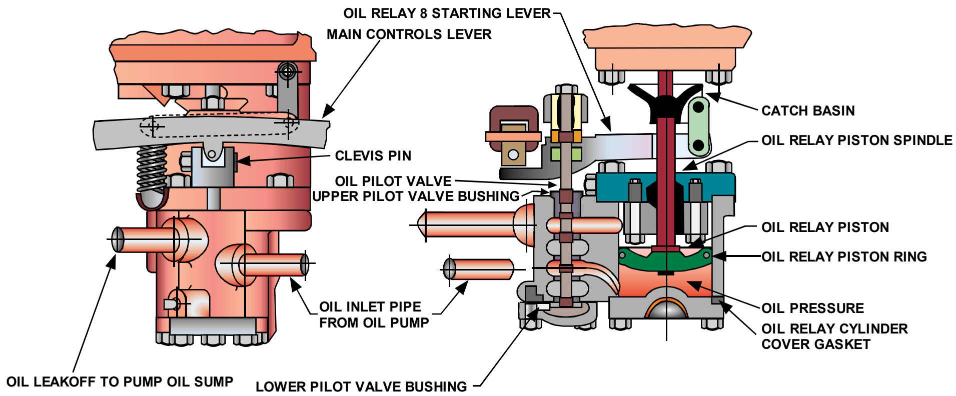


FIGURE 2-7: OIL RELAY ASSEMBLY (FW-10)

2.18 Closing air for YCV-1045 can be supplied from the Instrument Air System or from a permanently installed air accumulator (3-hour rating) to ensure closing air is available.

••Location

2.19 Turbine-driven AFW pump FW-10 is located in Room 19 on the 991-foot elevation of the Auxiliary Building, south of the motor-driven AFW pump, FW-6, and north of the plant air compressors.

2.20 Discharge header overpressure control panel AI-279 is mounted near AFW pump FW-10.

••Power Supplies

2.21 The DC motor for the auxiliary oil pump on FW-10 is powered from 125 VDC bus No. 2.

••Instrumentation and Controls

2.22 The following is a detailed description of the Instrumentation and Controls of the Turbine-Driven AFW Pump FW-10.

•••Local

2.23 FW-10 is manually controlled from the Control Room or locally from AFW shutdown panel AI-179. The controlling station is selected from REMOTE/LOCAL transfer switches on AI-179. The switches are normally positioned to REMOTE, which selects the Control Room as the controlling station. It is also possible to manually open the steam supply valves locally, which should start the turbine. If the balanced governor steam valve is closed, it can be opened with a pry bar to commence rolling the turbine.

QUESTION NUMBER: 048

With a normal electrical system lineup, how would the plant respond to the DC input breaker to inverter "C" failing open?

A. Power would be lost to instrument bus "C" until manually restored.

B.✔ The supply for instrument bus "C" would automatically switch to the bypass transformer for inverter "C".

C. The supply for instrument bus "C" would automatically switch to swing inverter, EE-8T

D. The cross tie breakers between instrument buses "A" and "C" would automatically close to supply instrument bus "C".

Question # 48 Rev 0

KA #: 062000 K4.10 Tier 2 Group 1:A.C. Electrical Distribution
Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Uninterruptable ac power sources
Importance 3.1 / 3.5

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
List the primary (preferred) and alternate (if any) power supplies to each bus/component.

Question Pedigree
Bank question used on the 2001-1 NRC exam. Replaced one of the distractors, changed stem to have DC input breaker fail open. Not counted as modified question.

K/A Fit:
Question addresses automatic transfer of power to an instrument bus to provide uninterruptable AC power.

Choice A:
Distractor: Plausible if Applicant does not understand auto transfer to bypass transformer. Incorrect because instrument bus power will auto transfer.

Choice B:
Correct answer: The supply for instrument bus "C" would automatically switch to the bypass transformer for inverter "C".

Choice C:
Distractor: Plausible because swing inverter EE-8T can be aligned to supply instrument bus "C." Incorrect because it requires manual alignment.

Choice D:
Distractor: Plausible because instrument bus "A" can be cross-tied to supply instrument bus "C." Incorrect because it requires manual alignment.

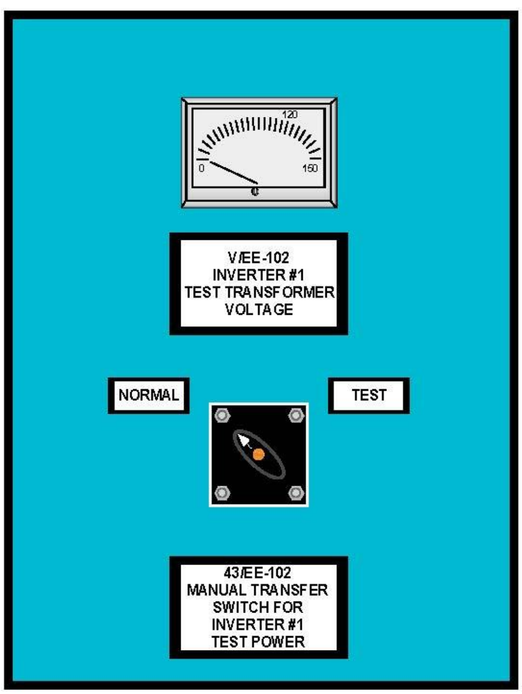
KA#:	062000 K4.10	Bank Ref #:	07-13-04 007
LP# / Objective:	0713-04 01.03	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	BANK
Reference:	ED STM	Handout:	NONE

2.16.2 Design/Specification

A. The 120V Instrument AC Distribution System comprises six buses, each supplied by a solid state inverter fed from a 125V DC bus with a backup source of power via a 480/120V voltage regulating transformer. There are two inverters that act as installed spares, one for busses A and C and one for busses B and D. An inverter functions to electronically convert DC to a reliable supply of regulated AC power. Each inverter is equipped with a static switch which monitors the output of the inverter and automatically switches the load to the backup source without a loss of power to the load if the inverter output is lost. The static switches can be manually controlled to bypass an inverter for maintenance.

B. Inverters 1 and 2 each have an additional 480/120V bypass transformer which provides a source of power to instrument bus No. 1 and No. 2 during maintenance and testing of the inverter. This transformer is selected for operation via a fast acting rotary snap switch installed between the inverter output and the instrument bus. This switch is a make before break switch to provide continuity of power to the instrument bus. This switch, along with a voltmeter to indicate the transformer output voltage, are provided in a junction box mounted near Inverters 1 and 2.

Figure 2-42 - EE-102 Inverter No. 1 Testing Transformer



C. The six safety-related instrument inverters A, B, C, D, EE-8T and EE-8U are 7.5 kVA 125V DC/120V AC single phase units with +2% voltage regulation. The non-safety related instrument inverters 1 and 2 are similar with 10 kVA capacity.

2.16.3 Location

A. The additional snap switch provided for inverters 1 and 2, along with a voltmeter to indicate the transformer output voltage, are provided in a junction box mounted near Inverters 1 and 2.

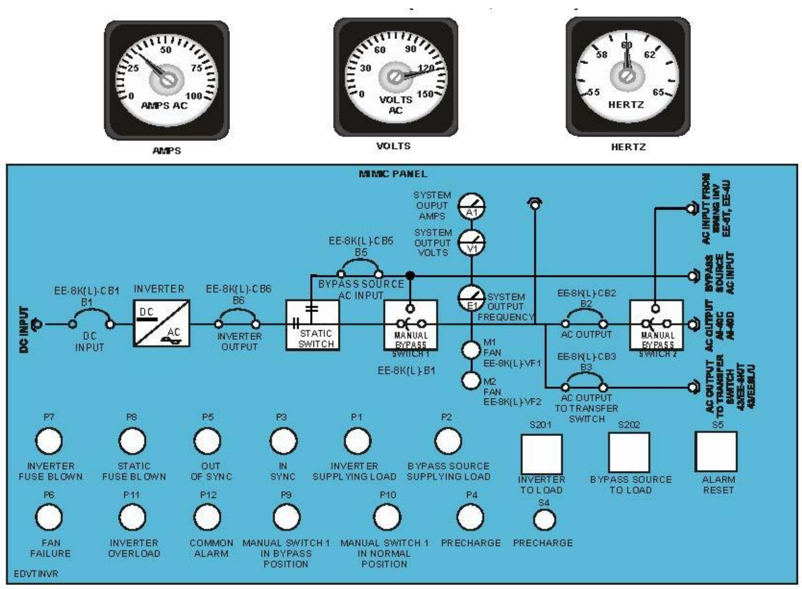
2.16.4 Power Supplies

A. Each 120VAC Instrument bus is supplied by a solid state inverter fed from a 125V DC bus with a backup source of power via a 480/120V voltage regulating transformer. There are two swing inverters that act as installed spares, one for busses A and C and one for busses B and D. The swing inverters also have a backup source of power via a 480/120V voltage regulating transformer.

2.16.5 Instrumentation and Control

A. The inverters are controlled from the switchgear rooms. Instruments on the front of the cabinets monitor the voltage and load. The output voltage and current of the inverters are also indicated in the control room. The safety-related inverters are indicated on panels AI-40A, AI-40B, AI-40C and AI-40D and the output of the non-safety related inverters is displayed at panels AI-42A and AI-42B. Ground detection indication for each inverter appear at each inverter distribution panel in the control room.

Figure 2-43 - Vital Inverters (EE-8K; EE-8L)



QUESTIONS REPORT
for ILO EXAM BANK

07-13-04 007

With a normal electrical system lineup, which one of the following statements would be true if Instrument Inverter "C" failed?

A. Power would be lost to instrument bus "C" until manually restored.

B.✔ The supply for instrument bus "C" would automatically switch to the bypass transformer for inverter "C".

C. The cross tie breakers between instrument buses "A" and "C" would automatically close to supply instrument bus "C".

D. The cross tie breakers between instrument buses "1" and "C" would automatically close to supply instrument bus "C".

QUESTION NUMBER: 049

Given the following plant conditions:

- A station blackout occurred 1 hour ago
- The Equipment Operator completed step 1 of EOP/AOP Attachment 6, “Minimizing DC loads,” 50 minutes ago
- The Equipment Operator opened G-03, “Generator ST-2 Vent Header Isolation Valve,” 30 minutes ago
- The turbine has just stopped rolling

What action should be taken from the control room at this time and why should it be taken?

A. Stop LO-4, the DC Oil Pump, to ensure the batteries can supply power for the next 3 hours.

B. Stop LO-12B, the DC Seal Oil Pump, to ensure the batteries can supply power for the next 3 hours.

C.✓ Stop LO-4, the DC Oil Pump, to ensure the batteries can supply power for the next 7 hours.

D. Stop LO-12B, the DC Seal Oil Pump, to ensure the batteries can supply power for the next 7 hours.

Question # 49 Rev 0

KA #: 063000 A1.01 Tier 2 Group 1: D.C. Electrical Distribution
Ability to predict and/or monitor changes in parameters associated with operating the dc electrical system controls including: Battery capacity as it is affected by discharge rate
Importance 2.5 / 3.3

CFR Number: 55.41(b)(8)
Components, capacity, and functions of emergency systems.

Fort Calhoun Objective:
GIVEN a copy of Attachment 6, EXPLAIN the steps necessary to minimize DC loads.

Question Pedigree
Modified question. The source question was used on the 2007 NRC exam.

K/A Fit:
The question addresses actions taken per procedures to reduce the battery discharge rate.

Choice A:
Distractor: Plausible because the action is correct and Fort Calhoun's blackout rule coping time is 4 hours. Incorrect because the objective of minimizing DC loads is the extend the battery life to 8 hours.

Choice B:
Distractor: Plausible because LO-12B is stopped as part of minimizing DC loads and Fort Calhoun's blackout rule coping time is 4 hours. Incorrect because LO-12B is tripped 2 hours after the blackout and the objective of minimizing DC loads is the extend the battery life to 8 hours.

Choice C:
Correct answer. LO-4 is stopped when the turbine stops rolling and the objective of minimizing DC loads is the extend the battery life to 8 hours.

Choice D:
Distractor: Plausible because LO-12B is stopped as part of minimizing DC loads. Incorrect because it is stopped 2 hours after the blackout.

KA#:	063000 A1.01	Bank Ref #:	07-18-17 021
LP# / Objective:	0718-17 02.03	Exam Level:	RO-8
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	EOP-AOP ATT 6	Handout:	NONE

Attachment 6

Minimizing DC Loads

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Performing the following step will allow up to 8 hours operation of the control and instrumentation devices required for Reactor shutdown without Battery Charger operation.

1. Reduce DC loads by performing the following steps **within 15 minutes** of the loss of Battery Chargers:

   a. Place **BOTH** of the following DC Bus 2 breakers in "OFF" (West Switchgear Room):

      - EE-8G-CB12, "400 CYCLE INVERTER EE-21"
      - EE-8G-CB8, "EMERGENCY LIGHTING PNL ELP-2 TRANSFER SWITCH"

   b. Place DC Bus 1 breaker EE-8F-CB11, "AUX BLDG EMGY LIGHTING PANEL ELP-1", in "OFF" (East Switchgear Room).

(continue)

Attachment 6

Minimizing DC Loads

INSTRUCTIONS | CONTINGENCY ACTIONS

1. (continued)

   c. Place 125 VDC Panel DC-PNL-1, Breaker 15, "ELP-5 EMERGENCY. LIGHTING PANEL" in "OFF" (West wall of Turbine Room, upper level).

2. Vent hydrogen from the Generator using G-03, "GENERATOR ST-2 VENT HEADER ISOLATION VALVE" (behind AI-134).

NOTE

To ensure adequate battery capacity, the DC Oil Pump should be stopped as soon as the turbine stops rolling, which will occur in approximately one hour.

3. **WHEN** the turbine has stopped rolling, **THEN** stop LO-4, DC Oil Pump.

Attachment 6

Minimizing DC Loads

INSTRUCTIONS | CONTINGENCY ACTIONS

4. **WHEN** two hours has elapsed since the loss of Battery Chargers, **THEN** reduce DC loads by performing the following steps:

   a. Ensure **BOTH** of the following breakers are closed (AI-42A):

      - I-BUS-I1-1, " INSTRUMENT BUS 1 MAIN BREAKER"
      - "CIRCUIT #1 AI-53 NORM FEED"

(continue)

Attachment 6

Minimizing DC Loads

INSTRUCTIONS | CONTINGENCY ACTIONS

4. (continued)

b. Place **ALL** of the following breakers in "OFF" (AI-42A):

- "CIRCUIT #2 AI-56 FEED"
- "CIRCUIT #3 AI-100 FEED"
- "CIRCUIT #4 IB-1A FEED"
- "CIRCUIT #5 CB-10,11 FEED"
- "CIRCUIT #7 CB-1,2,3 FEED"
- "CIRCUIT #8 AI-195 FEED"
- "CIRCUIT #9 AI-44 FEED"
- "CIRCUIT #10 AI-58 & AI-59 FEED"
- "CIRCUIT #11 CB-20 FEED"
- "CIRCUIT #12 AI-42 & AI-60 FEED"
- "CIRCUIT #13 CB-4 FEED"
- "CIRCUIT #18 AI-43A & AI-33C FEED"

c. Ensure **BOTH** of the following breakers are closed (AI-42B):

- I-BUS-I2-1, " INSTRUMENT BUS 2 MAIN BREAKER"
- "CIRCUIT #1 AI-53 EMERG FEED"

(continue)

Attachment 6

Minimizing DC Loads

INSTRUCTIONS

CONTINGENCY ACTIONS

4. (continued)

d. Place **ALL** of the following breakers in "OFF" (AI-42B):

- "CIRCUIT #2 AI-50 FEED"
- "CIRCUIT #3 AI-105, 107, 107-1, 181"
- "CIRCUIT #4 AI-55 FEED"
- "CIRCUIT #5 AI-101B FEED"
- "CIRCUIT #6 IB-2A FEED"
- "CIRCUIT #7 CB-10,11 FEED"
- "CIRCUIT #8 EE-32 ALARM PNL (EMGY)"
- "CIRCUIT #9 CB-4 FEED"
- "CIRCUIT #10 EE-32 AUX ALARM PNL (EMGY)"
- "CIRCUIT #11 AI-44 FEED"
- "CIRCUIT #12 AI-187 FEED"
- "CIRCUIT #13 CB-1,2,3 FEED"
- "CIRCUIT #14 AI-195 FEED"
- "CIRCUIT #15 AI-270 FEED"
- "CIRCUIT #16 AI-43B & AI-65B FEED"
- "CIRCUIT #17 CB-20 FEED"
- "CIRCUIT #18 AI-292 FEED"

Attachment 6

Minimizing DC Loads

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

Securing LO-12B, DC Seal Oil Pump takes precedence over complete venting of hydrogen from the Main Generator.

e. Stop LO-12B, Emergency Seal Oil Pump.

End of Attachment 6

Source question for question 49.

QUESTIONS REPORT
for ILO EXAM BANK

07-18-17 021

A station blackout has occurred due to a loss of 161 kv and 345 kv to the switchyard. The Equipment Operator reports that step one of "Minimizing DC loads has been completed. What additional action must be taken in control room.

A.✓ Stop LO-4, the DC Oil Pump, once the turbine stops turning.

B. Stop LO-12B, the DC Seal Oil Pump, when the generator stops turning.

C. Transfer AI-41A to its emergency DC power source.

D. Transfer AI-41B to its emergency DC Power source.

QUESTION NUMBER: 050

Given the following plant conditions:

- OI-DG-1, Attachment 1, "Idle Speed Start and Loading," is being performed
- Diesel Generator, D-1, is operating at 900 RPM
- The synchroscope is rotating slowly in the fast direction
- Incoming and outgoing voltages are matched.

What action should be taken after breaker 1AD1 is closed?

A. Increase voltage using voltage regulator, CS-90/D1

B. Decrease voltage using voltage regulator, CS-90/D1

C.✓ Increase load using governor control, CS-65/D1

D. Decrease load using governor control, CS-65/D1

Question # 50 Rev 0

KA #: 064000 A3.05 Tier 2 Group 1: Emergency Diesel Generators
Ability to monitor automatic operation of the ED/G system, including: Operation of the governor control of frequency and voltage control in parallel operation
Importance 2.8 / 2.9

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
Explain the interrelationships between the EDG and the Electrical Distribution System.

Question Pedigree
New question.

K/A Fit:
Question addresses operation of the governor control switch in parallel operation.

Choice A:
Distractor: Plausible if Applicant confuses voltage control with speed control. Incorrect because voltages are matched.

Choice B:
Distractor: Plausible if Applicant confuses voltage control with speed control and does not understand the relationship of the governor and load. Incorrect because voltages are matched.

Choice C:
Correct answer: Increasing the governor control switch setting will pick up load in parallel operation.

Choice D:
Distractor: Plausible if the Applicant does not understand the relationship of the governor and load. Incorrect because load must be increased.

KA#:	064000 A3.05	Bank Ref #:	N/A
LP# / Objective:	0713-05 01.02	Exam Level:	RO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	OI-DG-1	Handout:	NONE

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

3

h. Verify the following dampers are open:

- YCV-871G, Inlet ___
- YCV-871H, Inlet ___
- YCV-871E, Exhaust ___

i. Notify the Operator to perform the following:

- Visual inspection ___
- Verify oil visible in the Upper Sightglass ___

j. Record the Start Time in the Control Room Log and FC-1046, Diesel Generator Demand Record. ___ ________

4. IF Loading DG-1,
THEN perform the following:

a. Notify the Operator to verify Jacket Water Inlet Temperature is greater than 120°F. ___

b. Place CS-65/D1, Diesel Generator D1 Governor, to Raise until the Diesel Speed is 900 rpm. ___

c. Verify the Generator Field flashed. ___

d. Place D1/BUS 1A3 Sync Switch to ON. ___

e. Adjust CS-90/D1, Diesel Generator D1 Voltage Regulator, until the RUNNING VOLTS is approximately matched to the INCOMING VOLTS on the Synchroscope or the ERF DGD Display. ___

NOTE
Recommended synchroscope speed is less than 1 revolution per 10 seconds.

f. Adjust CS-65/D1 until the Synchroscope is rotating slowly in the **FAST** direction. ___

Attachment 1 - Idle Speed Start and Loading

PROCEDURE (continued) (✓) INITIALS

4

NOTE

Steps 4.g and 4.h may be performed without the procedure in hand. Sign-offs may be completed after these steps are performed.

CAUTIONS

1. Load must be immediately picked up following closure of 1AD1 to prevent motorizing the Diesel Generator.

2. Governor controls are extremely sensitive.

g. WHEN the Synchroscope is between 11 and 12 **O'CLOCK**, THEN close 1AD1 BREAKER. ___

h. Place CS-65/D1 to Raise to pick up 250-350 KW. ___

i. Place D1/BUS 1A3 Sync Switch to OFF. ___

j. IF the Diesel is loaded and Y3287A, ERF 1A3 Bus Voltage, is greater than 4375 VAC,
THEN immediately notify the System Engineer. ___

NOTES

1. Load should be maintained below the 2000 hr Rating vs Ambient Temp curve per TDB-III.26A Figure 1, DG-1 Output Power Rate.

2. Power factor may be determined by using TDB-III.26, Diesel Generator Capability Curve.

3. Current is normally limited to 400 amps at 2500 KW.

4. Diesel Generator manual loading and unloading rates should be maintained at less than 500 KW per minute.

5. Steps 4.k and 4.l may repeated as necessary while the diesel is loaded. Sign-offs may be completed after these steps are performed.

k. Place CS-65/D1 to RAISE picking up the desired DG-1 Load. ___

QUESTION NUMBER: 051

Given the following plant conditions:
- The Radiation Monitoring system is in normal alignment with all components operable
- RM-052 is aligned to the Auxiliary Building stack
- A waste gas decay tank has ruptured
- Radiation levels on RM-052 and RM-062 are reading $5x10^4$ CPM and rising
- RM-063 is reading low and is not changing

Why is RM-063 not responding?

A. RM-063 must be manually placed into operation.

B.✓ RM-062 countrate is not high enough to initiate sample flow to RM-063

C. RM-063 is not sensitive to noble gases contained in the waste gas decay tank

D. RM-063 sample flow is isolated when RM-052 is aligned to the stack

Question # 51 Rev 0

KA #: 073000 A4.02 Tier 2 Group 1: Process Radiation Monitoring System
Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel
Importance 3.7 / 3.7

CFR Number: 55.41(b)(11)
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Fort Calhoun Objective:
LIST radiation monitors with automatic actuations and STATE the automatic actuations that occur.

Question Pedigree
New question.

K/A Fit:
Question addresses monitoring RM-063 response during an accident.

Choice A:
Distractor: Plausible if Applicant believes RM-063 is operated like RM-064 and RM-065. Incorrect because RM-063 will begin to operate when the countrate on RM-062 is high enough.

Choice B:
Correct answer. Sample flow to RM-063 will begin when countrate is higher on RM-062.

Choice C:
Distractor: Plausible if Applicant does not understand that RM-063 is a noble gas monitor with the capability for manual grab sampling for particulate and iodine. Incorrect because RM-063 is sensitive to noble gases.

Choice D:
Distractor: Plausible if Applicant believes RM-063 is connected to RM-052. Incorrect because it is connected to RM-062.

KA#:	073000 A4.02	Bank Ref #:	N/A
LP# / Objective:	0712-03 04.01	Exam Level:	RO-11
Cognitive Level:	HIGH	Source:	NEW
Reference:	STM 33	Handout:	NONE

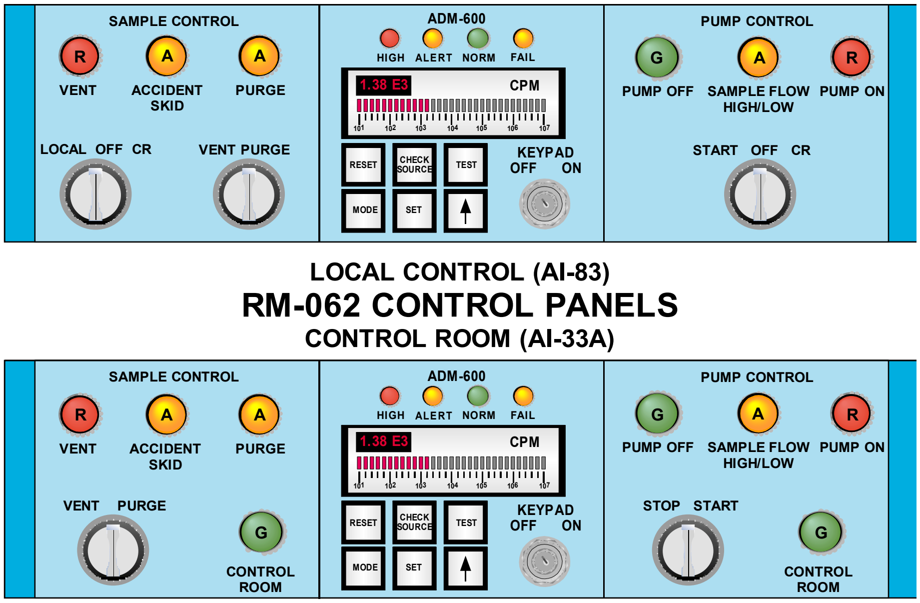


FIGURE 2-26: RM-062 CONTROL PANELS

••Failure Modes

2.136 RM-062 fails on loss of AC power. There are no air-powered actuation valves in the system. Compensatory actions are required whenever radiation monitor RM-062 is removed from service or fails. Details are in the ARP, ODCM and OI-RM-1.

•••Fail Position on Loss of Air or Power

2.137 RM-062 LED display becomes blank on loss of AC power. The CRHS signal is **not** actuated. The RM-062 AUX BLDG STACK TROUBLE alarm on AI-33C will actuate.

Accident Range Vent Stack Skid Monitor RM-063

2.138 The following is a detailed description of the accident range vent stack skid monitor, RM-063.

••Function

2.139 The accident range vent stack monitor skid (RM-063) is an off-line accident range noble gas sample skid designed to work in conjunction with RM-062 to sample the Auxiliary Building vent stack during accident conditions when RM-062 reaches full scale.

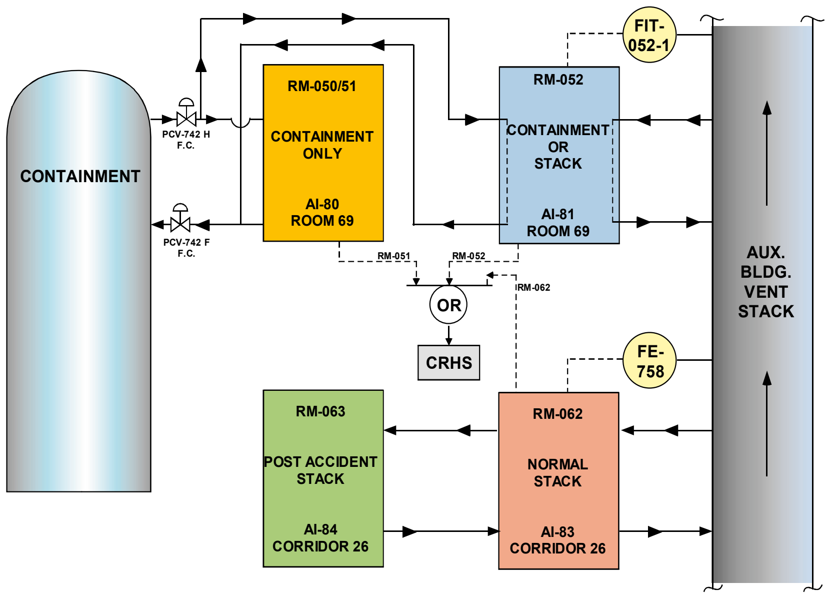


FIGURE 2-27: CONTAINMENT/AUX BLDG MONITOR ARRANGEMENT

••Design/Specification

2.140 The skid assembly (AI-84) is an open-frame construction and located adjacent to RM-062 in the east end of Auxiliary Building Corridor 26 and is comprised of the following equipment:

- RM-063 receives sample flow through a 3/4-inch sample line from a splitter block located on RM-062.
- Motor-operated inlet isolation valve MV1 opens automatically if RM-062 exceeds 5.0 E6 CPM or RM-063 exceeds 5.0 E-3 uCi/cc and closes when RM-063 decreases to less than 5.0 E-3 uCi/cc.
- Grab sample connections are located on the skid inlet and outlet piping.
- Two trains of fixed filter particulate/Iodine filters.

2.141 **NOTE**: The original design was to have active detector channels for each particulate/Iodine filter; however, due to calibration problems, it was decided to not use the channels and abandon the detectors (RE-044A/B) in place while still using the samplers for manual grab sampling.

- Solenoid valves SV1 and SV2 work together to close off sample flow from the particulate and Iodine filters and provide filtered room air for purging the flow through ion chamber.

QUESTION NUMBER: 052

During liquid releases to the river, what radiation monitors are used to ensure that the releases do not result in a dose that exceeds 1.5 mrem to the total body or 5 mrem to any organ during any given calendar quarter.

A.✓ RM-054A/B and RM-055

B. RM-055 and RM-057

C. RM-057 and RM-064

D. RM-054A/B and RM-064

Question # 52 Rev 0

KA #: 073000 K5.03 Tier 2 Group 1: Process Radiation Monitoring System
Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Relationship between radiation intensity and exposure limits
Importance 2.9 / 3.4

CFR Number: 55.41(b)(12)
Radiological safety principles and procedures.

Fort Calhoun Objective
Without references, state the purpose, applicability and content of Offsite Dose Calculation Manual.

Question Pedigree
New question.

K/A Fit:
Question addresses radiation monitors used to ensure radionuclide concentrations maintain public doses within limits during liquid releases.

Choice A:
Correct answer: RM-054A/B monitor blowdown releases and RM-055 monitors monitor tank releases.

Choice B:
Distractor: Plausible because both monitor releases. Incorrect because RM-057 (condenser offgas) monitors gaseous releases.

Choice C:
Distractor: Plausible because both monitor releases. Incorrect because RM-057 and RM-064 (steam line) monitor gaseous releases.

Choice D:
Distractor. Plausible because both monitor releases. Incorrect because RM-064 monitors gaseous releases.

KA#:	073000 K5.03	Bank Ref #:	N/A
LP# / Objective:	0762-02 5.01.00	Exam Level:	RO-12
Cognitive Level:	LOW	Source:	NEW
Reference:	ODCM 3.1.2	Handout:	NONE

Table 2.1.1 - Radioactive Liquid Effluent Monitoring Instrumentation

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release.		
a. Liquid Radwaste Effluent Line (RM-055)	1	1, 5
b. Steam Generator Blowdown Effluent Line (RM-054 A and B)	1 [1]	2, 5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	1	3
b. Steam Generator Blowdown Effluent Line	1	3
3. Radioactivity Recorders		
a. Liquid Radwaste Effluent Line	1	4
b. Steam Generator Blowdown Effluent Line	1	4

1 If one of the two radiation monitors is inoperable, the activity of both blowdown lines shall be monitored by the operable monitor within 2 hours of the declaration of inoperability by the Shift Manager, or the action steps of ACTION 2, Table 2.1.1 should be performed on the Steam Generator that is not being monitored.

3.1.2 Dose from Radioactive Liquid Effluents

A. Limiting Condition for Operation

1) The dose or dose commitment to an individual in unrestricted areas from radioactive materials in liquid effluents shall be limited to the following:

a) During any calendar quarter: Less than or equal to 1.5 mrem to the total body and 5 mrem to any organ; and

b) During any calendar year: Less than or equal to 3 mrem to the total body and 10 mrem to any organ.

APPLICABILITY: At all times

ACTION:

a) If the dose contribution, due to the cumulative release of radioactive materials in liquid effluents, exceeds the annual or quarterly dose objectives, submit a Special Report to the NRC, per Section 5.2.3, within 30 days.

B. Surveillance Requirements

1) Cumulative dose contributions from liquid effluents for the current calender quarter and the current calendar year shall be determined in accordance with the methodology and parameters in Part II of the Off-Site Dose Calculation Manual at least once per quarter.

QUESTION NUMBER: 053

Which of the following buses directly provides power to Raw Water Pump, AC-10C

A. 4160V Bus 1A1

B.✓ 4160V Bus 1A3

C. 480V Bus 1B3C

D. 480V Bus 1B3C-4C

Question # 53 Rev 0

KA #: 076000 K2.01 Tier 2 Group 1:Service Water System
Knowledge of bus power supplies to the following: Service water
Importance 2.7 / 2.7

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation,signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
When given specific plant conditions APPLY operating principles to diagnose Raw Water System response.

Question Pedigree
Bank question used on 2005 NRC exam

K/A Fit:
Question addresses power supply to a Raw water pump.

Choice A:
Distractor: Plausible if Applicant does not know power supply to AC-10C. Incorrect because power supply is bus 1A3.

Choice B:
Correct answer: Bus 1A3 is the power supply for AC-10C.

Choice C:
Distractor: Plausible if Applicant does not know power supply to AC-10C. Incorrect because power supply is bus 1A3.

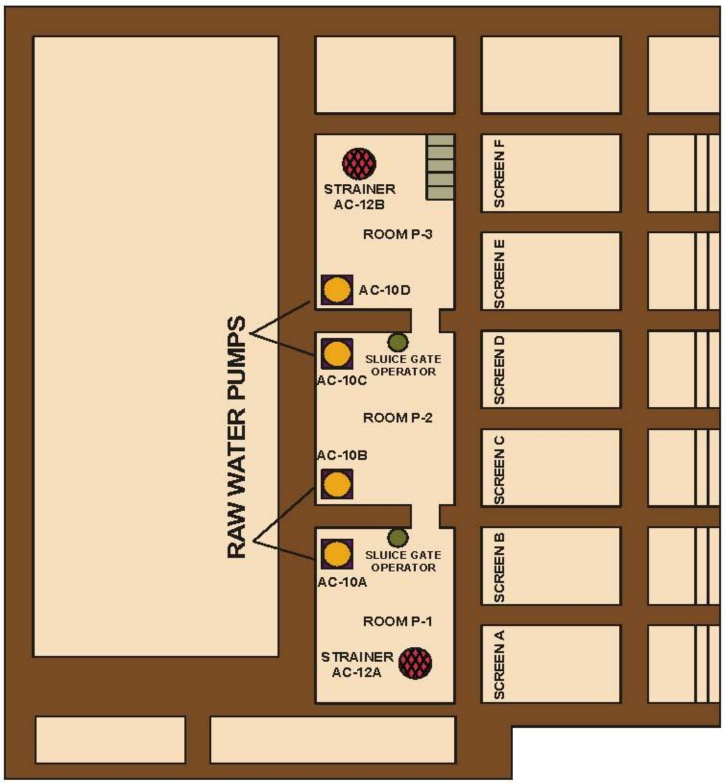
Choice D:
Distractor: Plausible if Applicant does not know power supply to AC-10C. Incorrect because power supply is bus 1A3.

KA#:	076000 K2.01	Bank Ref #:	07-11-19 037
LP# / Objective:	0711-19 01.00	Exam Level:	RO-7
Cognitive Level:	LOW	Source:	BANK
Reference:	STM-35	Handout:	NONE

2.1.3 Location

A. The raw water pumps are located inside the Intake Structure, 35 feet back from the river. Two of the pumps are located inside their individual cells while the remaining two pumps share a cell.

Figure 2-6 - Raw Water Pump Floor Plan Elevation 993'6"



B. The raw water pump breaker control switches are on CB-1,2,3.

2.1.4 Power Supplies

A. Each raw water pump is driven by a 3 phase, 200 hp, 1200 rpm, 4160 VAC motor that is powered from one-of-two vital electrical buses. Bus 1A3 supplies pumps AC-10A/AC-10C, and bus 1A4 supplies pumps AC-10B/AC-10D. During normal operation, off-site power from the 161 kV System supplies vital buses 1A3 and 1A4, and backup power is provided by its respective emergency diesel generator.

QUESTION NUMBER: 054

The plant was in hot shutdown when the following Annunciators came into alarm in the control room:

"COOLING WATER PRESSURE LOW"
"INSTRUMENT AIR PRESS LO"
"PLANT AIR PRESS LO"

Air pressure indication on both PI-1750 and PI-1700 was lowering.

The operator noted that the running bearing water pump and the running air compressors have tripped. The operators were able to start a backup air compressor, but it ran for only a few minutes. They were unable to start either bearing water pump.

What action is required to restore instrument air pressure?

A. Close the cross-tie valve between service air and instrument air.

B. Bypass the Instrument air dryers.

C. Align raw water backup cooling to an air compressor and then restart it.

D.✓ Align potable water backup cooling to an air compressor and then restart it.

Question # 54 Rev 0

KA #: 078000 K1.04 Tier 2 Group 1: Instrument Air System
Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor
Importance 2.6 / 2.9

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Explain the principles of Abnormal operation of the Compressed Air System in terms of flow paths, major paramaters, (temperature, pressure, flow, etc.), alarms and control devices.

Question Pedigree
Bank question used on 2007 NRC exam.

K/A Fit:
Question addresses backup cooling water to the air compressors.

Choice A:
Distractor: Plausible because this is the correct action for a service air leak. Incorrect because a loss of cooling water has occurred.

Choice B:
Distractor: Plausible because this is the correct action for a clogged dryer. Incorrect because a loss of cooling water has occurred.

Choice C:
Distractor: Plausible because Raw water backup cooling is used for many plant components. Incorrect because it is not used for the air compressors.

Choice D:
Correct answer: The air compressors have tripped due to a loss of normal cooling (bearing water). Backup cooling (potable water) should be aligned to the air compressors

KA#:	078000 K1.04	Bank Ref #:	07-11-07 022
LP# / Objective:	0711-07 01.05	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	BANK
Reference:	OI-CA-5	Handout:	NONE

Attachment 1 - Initiation of Backup Cooling to Air Compressor(s)

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision No.________________ Date:________ ________

2. Air Compressor(s) are operating per OI-CA-1. ________

3. Potable Water System is in service per OI-PW-1. ________

4. Verify Blair water supply pressure is greater than 45 psig on PI-1601-2, Blair Water Pressure gauge (Water Plant Office). ________

5. If this will be the only in service Air Compressor, protected equipment labels should be placed at the component and its associated breaker. ________

PROCEDURE

1. IF CA-1A is to be lined up to backup cooling, THEN complete the following:

   a. Ensure compressor is not running. ________

   b. Close the following valves:

      - AC-586, Air Compressor Aftercooler CA-2A Inlet Valve ___
      - AC-588, Air Compressor CA-1A Outlet Valve ___
      - AC-584, Air Compressor CA-1A Intercooler Inlet Valve ___
      - AC-589, Air Compressor CA-1A Aftercooler CA-2A Outlet Valve ___ ________

NOTE
AC-1043, Potable Water Outlet Valve AND AC-1045, Aftercooler Potable Water Outlet Valve may need to be throttled to prevent Air Compressor trip due to low cooling water pressure.

   c. Open the following valves:

      - AC-1042, Air Compressor CA-1A Intercooler Potable Water Inlet Valve ___
      - AC-1044, Air Compressor CA-1A Aftercooler CA-2A Potable Water Inlet Valve ___
      - AC-1028, Comp Cooling Wtr Valve FCV-1990A Bypass Valve ___ ________

Attachment 1 - Initiation of Backup Cooling to Air Compressor(s)

PROCEDURE (continued) (✓) INITIALS

1

NOTE

The air compressors will automatically trip on low jacket cooling water pressure at 30 psig or a cooling water temperature of 130°F.

d. Throttle open the following valves to maintain a pressure of 40 to 50 psig on PI-1988A:

- AC-1045, Air Compressor CA-1A Aftercooler CA-2A Potable Water Outlet Valve ___
- AC-1043, Air Compressor CA-1A Potable Water Outlet Valve ___ ________

e. Start CA-1A per OI-CA-1. ________

f. IF air compressor trips,
THEN raise cooling water pressure by throttling closed on the Aftercooler and/or air compressor cooling water outlet valve(s).

1) Adjust valves to desired pressure:

- AC-1045 ___
- AC-1043 ___

2) Go to Step 1.e. ___ ________

g. Close AC-1028, Comp Cooling Wtr Valve FCV-1990A Bypass Valve. ________

h. Verify potable water cooling flow by checking the following:

- FI-1955A, Air Compressor CA-1A Cooling Water Flow Indicator ___
- TI-1702A, Air Compressor Aftercooler CA-2A Temperature Indicator ___ ________

i. Allow temperatures to stabilize and if necessary, adjust flow in small increments using Aftercooler/Potable Water Outlet valves to maintain temperature and pressure.

- AC-1045 ___
- AC-1043 ___ ________

QUESTION NUMBER: 055

The plant is operating at full power. A containment pressure reduction is in progress.

Which of the following conditions is considered to be a Loss of Containment Integrity?

A. HCV-746A and HCV-746B, "Containment Pressure Relief Valves," are both open.

B. The outer Personnel Air Lock door has failed it's seal leakage test

C.✔ An automatic CIAS valve failed a surveillance test due to excessive stroke time.

D. The Limitorque operator for a locked closed containment isolation MOV is inoperable.

Question # 55 Rev 0

KA #: 103000 K3.02 Tier 2 Group 1: Containment System
Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under normal operations
Importance 3.8 / 4.2

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
DESCRIBE how containment integrity is monitored.

Question Pedigree
Bank question used on 1995 NRC exam. Minor rewording, not counted as modified question.

K/A Fit:
Question addresses loss of containment integrity during normal operation.

Choice A:
Distractor: Plausible because these are two containment isolation valves in series. Incorrect because they may both be opened during a containment pressure reduction with an approved release permit.

Choice B:
Distractor: Plausible because the PAL doors function to maintain containment integrity. Incorrect because the inner PAL door is still intact.

Choice C:
Correct answer per AOP-12

Choice D:
Distractor: Plausible because an inoperable containment isolation MOV could be a loss of containment integrity. Incorrect because it is locked closed.

KA#:	103000 K3.02	Bank Ref #:	07-11-08 023
LP# / Objective:	0711-08 02.01	Exam Level:	RO-5
Cognitive Level:	LOW	Source:	BANK
Reference:	AOP-12	Handout:	NONE

1.0 PURPOSE

This procedure provides guidance in the event of the loss of required Containment Integrity.

2.0 ENTRY CONDITIONS

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

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

B. The Equipment Hatch is not properly sealed.

C. Neither Personnel Air Lock Door is properly sealed.

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

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

F. Noticeable air leakage from Containment.

QUESTIONS REPORT
for ILO EXAM BANK

07-11-08 023

With the plant at power and no evolutions in progress, which of the following conditions is a Loss of Containment Integrity?

A. HCV-746A (Pressure Relief) is opened.

B. One of the Personnel Air Lock doors is open.

C.✓ An automatic CIAS valve fails a surveillance test due to not cycling full open.

D. A containment isolation MOV is inoperable but, is locked closed.

QUESTION NUMBER: 056

Excessive leakage from the RCS through the Control Element Drive Mechanism shaft seals will result in rising level in _______________.

A.✓ The Reactor Coolant Drain Tank

B. The Pressurizer Quench Tank

C. The Spent Regenerant Tank

D. The Volume Control Tank

Question # 56 Rev 0

KA #: 001000 K3.02 Tier 2 Group 2: Control Rod Drive System
Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: RCS
Importance 3.4 / 3.5

CFR Number: 55.41(b)(3)
Mechanical components and design features of the reactor primary system.

Fort Calhoun Objective:
Describe the interface/interaction between the CRDS and the following systems/components:

Question Pedigree
Bank Question.

K/A Fit:
Question addresses Control Rod Drive system leakage to a RCS component.

Choice A:
Correct answer.

Choice B:
Distractor: Plausible if Applicant does not understand where CRDS seal leakage goes. Incorrect because it goes to the RCDT.

Choice C:
Distractor: Plausible if Applicant does not understand where CRDS seal leakage goes. Incorrect because it goes to the RCDT.

Choice D:
Distractor: Plausible if Applicant does not understand where CRDS seal leakage goes. Incorrect because it goes to the RCDT.

KA#:	001000 K3.02	Bank Ref #:	07-12-26 01.02
LP# / Objective:	0712-26 01.02	Exam Level:	RO-3
Cognitive Level:	LOW	Source:	BANK
Reference:	STM 11	Handout:	NONE

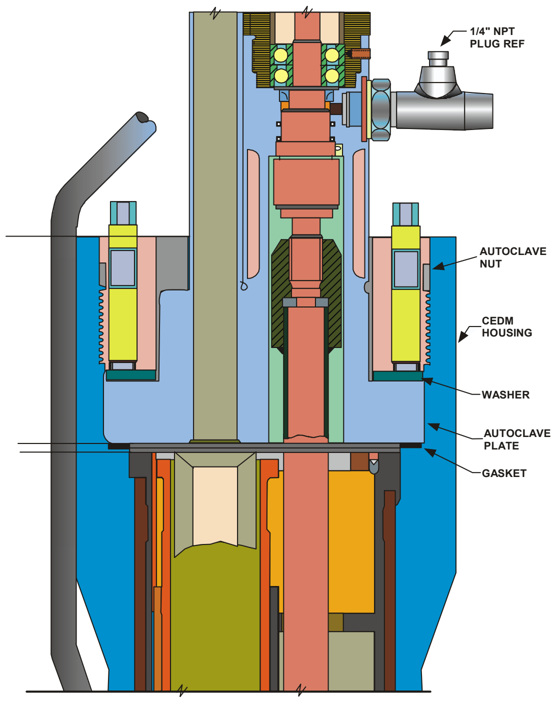


FIGURE 2-19: CEDM SEALING

2.21 The tool access tube is sealed by a bolted closure with a flexitalic gasket. The drive shaft from the clutch penetrates the autoclave and is sealed with a rotating mechanical seal. The mechanical seal uses a chrome oxide and graphitar mating surface. Approximately 1.5 gpm component cooling water (CCW) flow is supplied to a jacket around the seal to ensure the temperature does not exceed 250°F. Leakage through the seal is piped to the reactor coolant drain tank, and excessive RCS leakage is sensed by thermocouples installed in the leakoff lines which alarm at 200°F on the plant computer.

2.22 There are two clutches installed in each shutdown and regulating group CEDM, a trip clutch and an anti-reverse rotation clutch. The trip clutch transmits torque from the rod drive motor to the drive shaft. The trip clutch has two toothed sections. When the 33V DC power is applied, the lower toothed section rises and engages the teeth of the driving section. A reactor trip interrupts clutch power and allows gravity to pull the lower section away and disengage the clutch teeth. The pinion, drive shaft, and lower portion of the clutch rotate as the control rod falls into the core.

QUESTION NUMBER: 057

Following a control room evacuation, AOP-07, "Control Room Evacuation," directs the Operators to maintain RCS Pressure 2050-2150 psia by controlling "Pressurizer Back-Up Heaters," Bank 4 Groups 10, 11 and 12.

What 480 volt MCC powers these heaters?

A. MCC-3A4

B. MCC-3B3

C. MCC-3C4C-2

D.✓ MCC-4C1

Question # 57 Rev 0

KA #: 011000 K2.02 Tier 2 Group 2: Pressurizer Level Control System
Knowledge of bus power supplies to the following: PZR heaters
Importance 3.1 / 3.2

CFR Number: 55.41(b)(3)
Mechanical components and design features of the reactor primary system.

Fort Calhoun Objective
LIST the power supplies for the pressurizer heaters.

Question Pedigree
New question

K/A Fit:
Question addresses power supply to pressurizer heater bank 4, groups 10, 11 and 12.

Choice A:
Distractor: Plausible if Applicant does not know power supply to these heaters. Incorrect because the power supply is MCC-4C1

Choice B:
Distractor: Plausible if Applicant does not know power supply to these heaters. Incorrect because the power supply is MCC-4C1.

Choice C:
Distractor: Plausible if Applicant does not know power supply to these heaters. Incorrect because the power supply is MCC-4C1.

Choice D:
Correct answer. The power supply is MCC-4C1

KA#:	011000 K2.02	Bank Ref #:	N/A
LP# / Objective:	0711-20 01.06C	Exam Level:	RO-3
Cognitive Level:	LOW	Source:	NEW
Reference:	AOP-07	Handout:	NONE

Section I - Plant to Hot Shutdown

INSTRUCTIONS

CONTINGENCY ACTIONS

5. Monitor Reactor power on NR-004 and NI-004, "WIDE RANGE NEUTRON FLUX CHAN "D" SIGNAL PROCESSOR" (AI-212).

****

CAUTION

Charging to the RCS may cause overpressurization due to the isolation of Letdown and RCS Heatup.

****

6. Maintain PZR level (45%-60%) by operating CH-1B, Charging Pump (AI-185).

7. Maintain a record of Charging Pump run time to support estimates of RCS boron concentration.

8. Maintain RCS pressure 2050-2150 psia by controlling "PRESSURIZER BACK-UP HEATERS", Bank 4 Groups 10, 11 and 12 (MCC-4C1).

QUESTION NUMBER: 058

Given the following plant conditions:
- A power increase is in progress per OP-4, LOAD CHANGE AND NORMAL POWER OPERATION
- The reactor is at 80% power
- Group 4 CEAs are at 52 inches
- All other CEAs are fully withdrawn.

Using the attached PDIL curve, determine what procedure should be entered as a result of these conditions?

A. AOP-02, "CEA and Control System Malfunctions"

B.✓ AOP-03, "Emergency Boration"

C. AOP-05, "Emergency Shutdown"

D. EOP-00, "Standard Post-Trip Actions"

Question # 58 Rev 0

KA #: 014000 2.4.04 Tier 2 Group 2: Rod Position Indication System
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.
Importance 4.5 / 4.7

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
DISCUSS the power dependent insertion limit including: Describing required action when the Emergency Boration Limit is exceeded.

Question Pedigree
Bank question used on 2009 NRC exam.

K/A Fit:
Question addresses rod position that requires entry into AOP-03.

Choice A:
Distractor Plausible because AOP-02 can be entered for CEA mispositioning. Incorrect because PDIL is violated and emergency boration is required.

Choice B:
Correct answer: PDIL is violated and AOP-03, emergency boration is required.

Choice C:
Distractor Plausible because AOP-05 can be entered for CEA mispositioning. Incorrect because PDIL is violated and emergency boration is required.

Choice D:
Distractor Plausible because EOP-00 can be entered for CEA mispositioning. Incorrect because PDIL is violated and emergency boration is required.

KA#:	014000 2.4.04	Bank Ref #:	07-05-09
LP# / Objective:	0705-09 01.13E	Exam Level:	RO-5
Cognitive Level:	HIGH	Source:	BANK
Reference:	AOP-03	Handout:	PDIL CURVE

1.0 PURPOSE

This procedure provides guidance in the event Emergency Boration of the RCS is required.

2.0 ENTRY CONDITIONS

The following conditions require the initiation of Emergency Boration using this procedure:

A. Uncontrolled Plant Cooldown.

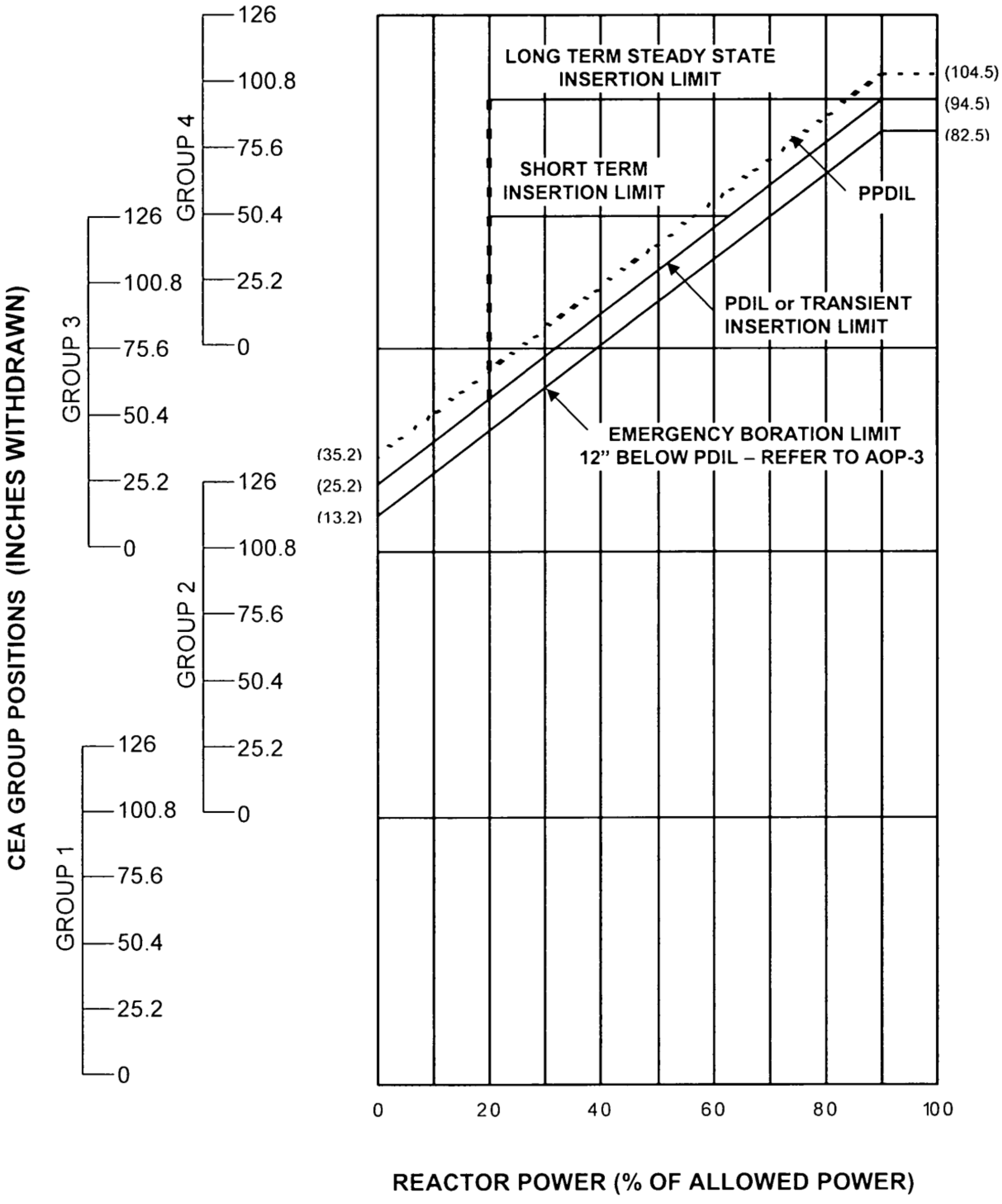
B. Two or more CEAs inserted below the Emergency Boration Limit of the COLR Power Dependent Insertion Limit Curve.

C. Reactivity anomaly in a positive direction with all CEA's inserted.

D. Unexpected rise in count rate or unexpected doubling of count rate on two or more channels during refueling.

E. Failure of more than one Regulating or Shutdown CEA to insert following a Reactor trip.*

* Guidance for Emergency Boration with these conditions is provided in more appropriate procedures.



COLR	POWER DEPENDENT INSERTION LIMIT	FIGURE 2

QUESTION NUMBER: 059

The wiring from one Core Exit Thermocouple (CET) has just failed open.

How would "Representative CET Temperature" as indicated on the ERF computer be affected?

A. The Representative CET Temperature would increase to 2300°F.

B. The Representative CET Temperature would decrease to containment temperature.

C. The Representative CET Temperature would decrease to 32°F.

D.✓ The Representative CET Temperature would not change.

Question # 59 Rev 0

KA #: 017000 K6.01 Tier 2 Group 2: In-Core Temperature Monitor System
Knowledge of the effect of a loss or malfunction of the following ITM system components: Sensors and detectors
Importance 2.7 / 3.0

CFR Number: 55.41(b)(2)
General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Fort Calhoun Objective:
EXPLAIN the principles of operation (both normal and abnormal) of the QSPDS System in terms of signal flow paths, major parameters (temperature, pressure, flow, etc.) alarms and control devices.

Question Pedigree
New question.

K/A Fit:
Question addresses how a loss of a CET would affect displayed representative CET temperature.

Choice A:
Distractor: Plausible if Applicant believes CET will fail to maximum scale but does not realize how representative CET temperature is calculated.

Choice B:
Distractor: Plausible if Applicant believes CET will fail to reference junction temperature but does not realize how representative CET temperature is calculated.

Choice C:
Distractor: Plausible if Applicant believes CET will fail to minimum scale but does not realize how representative CET temperature is calculated.

Choice D:
Correct answer: Representative CET temperature is calculated by rejecting the outlier CET temperatures and averaging the rest.

KA#:	017000 K6.01	Bank Ref #:	N/A
LP# / Objective:	0712-23 01.06	Exam Level:	RO-2
Cognitive Level:	HIGH	Source:	NEW
Reference:	OI-QSP-1	Handout:	NONE

Attachment 7 - Core Exit Thermocouple (CET) Temperature Calculation

PREREQUISITES (✓) INITIALS

1. Procedure Revision Verification

   Revision Number________________ Date:________ ________

2. At least one Qualified Safety Parameter Display System channel is operational. ________

PROCEDURE

1. The QSPDS displays individual Core Exit Temperatures with a Core Map, the highest and next highest Core Exit Temperature in each quadrant, and a representative Core Exit Temperature.

2. The Representative Core Exit Temperature is calculated based on statistical analysis with practical checks from other inputs.

   a. Using the Representative CET Temperature the Pressure AND Temperature Saturation Margins are calculated and displayed. However, the CET calculated variables will be flagged with a question mark during the time that the QSPDS is identifying Out-of-Range and Suspect CET inputs.

   b. WHEN the QSPDS completes one (1) complete cycle of the CET algorithm without flagging or removing a flag from any CET input, THEN the inputs are considered stable AND accurate and the question mark is removed from the calculated variables.

   c. The CET calculated variables will also be flagged with all question marks if the number of valid CET inputs is less than nine (9).

3. Inputs considered invalid are either failed or deviated from the mean of the CET inputs by a specific amount.

   a. Failed inputs are displayed as all question marks.

   b. Deviated inputs are displayed as a question mark in front of the displayed value indicating it is suspect.

QUESTION NUMBER: 060

Given the following plant conditions:

- The plant was operating at full power
- A large LOCA occurred in containment
- All safeguards equipment operated as designed
- The “CNTMT VENT FAN VA-3A FILTER TEMP HI” annunciator alarmed.
- TIC-866 indicates that the charcoal bed temperature is 460°F

What action should be taken regarding HCV-864, "Charcoal Filter Spray Valve," in response to these conditions?

A. Verify HCV-864 opened automatically.

B. Manually open HCV-864.

C. Manually initiate Containment Spray and open HCV-864.

D.✓ HCV-864 is not operated per FCS procedures.

Question # 60 Rev 0

KA #: 027000 A2.01 Tier 2 Group 2: Containment Iodine Removal System
Ability to (a) predict the impacts of the following malfunctions or operations on the CIRS; and (b) based on those predictions, use Procedures to correct, control, or mitigate the consequences of those malfunctions or operations: High temperature in the filter system
Importance 3.0 / 3.3

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Explain automatic controls associated with the Containment Air Cooling and Filtering System.

Question Pedigree
New question.

K/A Fit:
Question addresses procedural response for a high charcoal filter temperature.

Choice A:
Distractor: Plausible if Applicant believes the valve opens automatically. Incorrect because it doesn't.

Choice B:
Distractor: Plausible if Applicant believes there is procedural direction to open the valve and doesn't realize containment spray is also required for filter spray. Incorrect because the is no procedural direction to open this valve.

Choice C:
Distractor: Plausible if Applicant believes there is procedural direction and realizes that containment spray flow is required filter spray. Incorrect because there is no procedural direction.

Choice D:
Correct answer. HCV-864 was designed to mitigate a charcoal filter fire in the original plant design. However, opening HCV-864 was removed from FCS procedures following an engineering analysis.

KA#:	027000 A2.01	Bank Ref #:	N/A
LP# / Objective:	0714-02 01.04	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	ARP-AI-30A/A33-1	Handout:	NONE

Panel: **AI-30A**	Annunciator: **A33-1**	Window: **H-4**

CONTAINMENT FILTER HIGH TEMPERATURE

SAFETY RELATED

CNTMT VENT FAN
VA-3A
FILTER TEMP HI

Tech Spec References: 2.4

Initiating Device TIC-866 Setpoint >450ºF Power AI-41A

OPERATOR ACTIONS

1. Verify TIC-866, Containment Filter Temperature High (AI-30A).

2. Verify TE-715, Containment Temperature High.

CAUTION

Water vapor may cause a significant temperature rise (90-140ºF) when air flow through the filter is stopped.

2.1 Ensure VA-3A, Cntmt Vent Fan is in operation.

2.2 Ensure CCW is supplied to VA-15A.

2.3 Verify filter temperature is reduced.

3. Ensure compliance with requirements of Technical Specification 2.4.

PROBABLE CAUSES

- Water vapor in VA-15A housing with no flow thru VA-3A

REFERENCES

161F597 Sh 10 09810
136B2331 Sh 139 07251
IC-CP-01-0866
11405-M-1 Sh 1 10431

OUTLINE OF INSTRUCTION | COMMENTS

II. C. 6.

e. Each cooling and filtering unit contains 288 charcoal filter cells.

Adsorption - assimilation of gas, vapor, or dissolved matter by the surface of a solid or liquid.

f. Rated flow through each cell is 383 cfm.

g. Elemental iodine adsorption efficiency is 99.9%.

h. Overheating of the Charcoal is not probable, due to the following:

(1) Charcoal filter ignition temperature is 644EF.

(2) Maximum Containment temperature after a DBA is 417EF (EA-93-022), Containment temperature after a Super Heated Condition or a 300 PSIA spike is reduced to less than 288EF in 65 seconds.

(3) Maximum Charcoal temperature rise due to Iodine decay is less than 6EF (5.88) approximately 500 minutes after an accident which is well after DBA or MSLB peak temperatures.

i. FCS no longer uses Containment Spray to douse the Charcoal filters, due to the following:

(1) It has been found that temperature based fire detection methods in air cleaning units are unreliable.

OUTLINE OF INSTRUCTION	COMMENTS
II. C. 6. i.	
(2) Water sprays were found to be ineffective, and spraying the hot carbon beds results in an initial temperature rise due to the heat of adsorption of water (90-140EF).	
(3) U.S. NRC Regulatory Guides 1.52, Rev. 2, and 1.140, do not require water sprays for adsorber beds.	
j. The EOPs/AOPs no longer reference the use of HCV-864 and 865 for dousing the Charcoal filters, but the following is a description of the equipment locations and controls since they are still present in the system.	
(1) Inside containment are two air-operated isolation valves one from each spray header leading to the filter.	
(2) Valve HCV-864 operated from panel AI-30A provides dousing water to filter VA-6A.	EO 1.2
(3) Valve HCV-865 operated from panel AI-30B provides dousing water to filter VA-6B.	EO 1.2
(4) Handswitch positions CLOSE/ NORMAL/OPEN.	
(5) Green shut/red open indicating lights above handswitch.	EO 1.1
k. Temperature indication	Visual Aid 1.4 SHB page 6 EO 1.1

OUTLINE OF INSTRUCTION	COMMENTS

II. C. 6. kl.

(1) Thermistor strip temperature detectors pass through 24 uniformly separated beds within each cooling and filtering unit.

(2) TE-866 (TE-867) provides temperature indication on panel AI-30A (AI-30B) for vent fan VA-3A (VA-3B) filter VA-6A (VA-6B).

l. Alarms — NOTE: Refer to current revision of ARP. EO 1.3

(1) Containment ventilation fan VA-3A filter temperature HI on panel AI-30A annunciator A33-1.

(a) Setpoint 450EF

(b) Trip device TIC-866 — NOTE: Ask what actions are required.

(2) Containment ventilation fan VA-3B filter temperature HI on panel AI-30A annunciator A34-1.

(a) Setpoint 450EF

(b) Trip device TIC-867

7. Cooling coils (VA-1A and VA-1B) (VA-8A and VA-8B)

a. Located within containment cooling and filtering and containment cooling units VA-3A, VA-3B, VA-7C and VA-7D. — Visual Aid 1.3, SHB page 5

b. Cool containment air.

QUESTION NUMBER: 061

Given the following plant conditions:

- A LOCA occurred
- Containment Pressure is 8.2 psig and lowering
- You have been directed to place the containment hydrogen analyzers in service

What action must be taken to open the H2 Analyzer Containment Isolation Valves?

A. Place the valve control switches in "OPEN"

B.✓ Place the valve control switches in "O'RIDE"

C. Reset CIAS, then place the valve control switches in "OPEN"

D. Reset CIAS, then place the valve control switches in "O'RIDE"

Question # 61 Rev 0

KA #: 028000 A1.02 Tier 2 Group 2: Hydrogen Recombiner and Purge Control System
Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Containment pressure
Importance 3.4 / 3.7

CFR Number: 55.41(b)(7)
Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes,and automatic and manual features.

Fort Calhoun Objective:
Using the operating instructions DESCRIBE how to operate the hydrogen analyzer as required by plant conditions.

Question Pedigree
New question

K/A Fit:
Question addresses opening the hydrogen analyzer containment isolation valves with containment pressure above setpoint that allows reset of containment isolation.

Choice A:
Distractor: Plausible if Applicant does not recognize that the valves will not open in the open position with CIAS. Incorrect because then will not open with control switches in the open position.

Choice B:
Correct answer. Override position allows opening these valves with a CIAS.

Choice C:
Distractor: Plausible if Applicant believes CIAS can be reset. Incorrect because containment pressure is too high to reset CIAS.

Choice D:
Distractor: Plausible if Applicant does not realize that CIAS does not have to be reset to open valves in override position. Incorrect because CIAS can not be reset at this containment pressure.

KA#:	028000 A1.02	Bank Ref #:	N/A
LP# / Objective:	0714-03 02.00	Exam Level:	RO-7
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP/AOP ATT 16	Handout:	NONE

Attachment 16

Containment Hydrogen Analyzer Startup

INSTRUCTIONS

CONTINGENCY ACTIONS

1. Start the Hydrogen Analyzers, by performing the following steps:

   a. Open **ONE** of the Containment Hydrogen Sampling Valves, HCV-820C/D/E/F/G/H.

   b. Open **ONE** of the Containment Hydrogen Sampling Valves, HCV-883C/D/E/F/G/H.

   c. Place **ALL** of the following switches in "O'RIDE":

      - "H2 ANALYZER VA-81A ISOLATION VALVES OUTBD HCV-820A/821A"
      - "H2 ANALYZER VA-81B ISOLATION VALVES INBD HCV-883A/884A"
      - "H2 ANALYZER VA-81A ISOLATION VALVES INBD HCV-820B/821B"
      - H2 ANALYZER VA-81B ISOLATION VALVES OUTBD HCV-883B/884B"

(continue)

••Instrumentation and Controls

2.13 The following is a detailed description of the instrumentation and controls of the hydrogen analyzers, VA-81A/B.

•••Local

2.14 The local panels and remote panels each have a function selector switch and local-remote selector switch. However, the analyzers are normally operated from the remote panels in the control room. System operation from the local panels is normally performed by I&C technicians during calibration and surveillance testing. Each local panel has a breaker and motor-starter controller with a reset push button in the rear of the panels in Room 59.

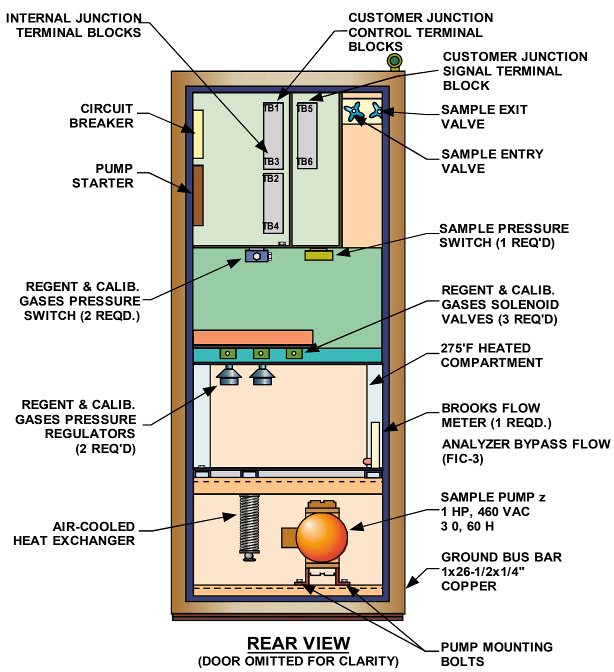


FIGURE 2-3: LOCAL CONTROL PANEL - REAR VIEW

•••Remote

2.15 The inboard containment isolation valves and outboard isolation valves are operated in pairs via a three-position control switch on the respective containment isolation panel, AI-43A/B. The switch positions are CLOSE, OPEN and OVERRIDE. Control switches HC-820A/821A and HC-883A/884A are on AI-43A, and control switches HC-820B/821B and HC-883B/884B are on AI-43B.

2.16 The control switches for the sample isolation valves are grouped in pairs, but only one sample valve may be opened at any one time by an individual switch. The control switch positions are OPEN, BOTH VALVES CLOSED and OPEN. For example, HCV-820C is opened by placing the control switch in the first (left-hand side) OPEN position, and HCV-820D is opened by placing the control switch in the second (right-hand side) OPEN position on AI-65A. The mid-position sends a signal to close both valves. Each valve has two status lights (green-closed and red-open) on AI-65A/B for position indication.

•••Interlocks

2.17 Actuation of a containment isolation actuation signal (CIAS) will close the inboard and outboard sample isolation valves. However, the signal may be overridden with its respective control switch on panel AI-43A/B, as allowed by Technical Specifications 2.12 and 2.15.

•••Alarms and Indications

2.18 The following alarms are located on control room panel AI-65A (similar alarms are located on AI-65B):

- CONTAINMENT H2 LEVEL HI alarm actuates if hydrogen concentration exceeds 4% on VA-81A.
- CONTAINMENT H2 SAMPLING SYSTEM COMMON FAILURE alarm actuates for any of the following: analyzer low temperature, sample flow low, reagent or calibration gas low pressure, hydrogen analyzer cell malfunction, hydrogen analyzer power supply failure.
- CONTAINMENT H2 SAMPLING SYSTEM REMOTE / LOCAL OFF NORMAL alarm actuates whenever the LOCAL position is selected on the push button control switch at remote panel AI-65A or local panel VA-81A.
- CONTAINMENT H2 SAMPLE VALVES LOSS OF POWER alarm actuates if power is lost to any of the six sample valves: HCV-820C/D/E/F/G/H.
- CONTAINMENT H2 SAMPLE VALVES OFF NORMAL alarm actuates if the control switch on AI-65A is in the OPEN position for any of the six sample isolation valves.

2.19 The following alarm is on control room panel AI-43A (a similar alarm is on panel AI-43B) - H2 PURGE/ANALYZER ISOLATION VALVES OVERRIDE alarm actuates if the control switch for any hydrogen analyzer (or hydrogen purge) isolation valve associated with AI-43A is in the OVERRIDE position.

QUESTION NUMBER: 062

Due to a Loss of Instrument Air, both MSIVs (Main Steam Isolation Valves) failed closed.

Assuming NO operator action, which of the following Control Room indications would indicate that MS-291 and MS-292 (Air Operated Safety Valves) are open?

A. The red position indication lights on CB-10

B. The valve position icons on the DCS display

C. Steam Generator pressure indicating approximately 900 psia

D.✓ Steam Generator pressure indicating approximately 1000 psia

Question # 62 Rev 0

KA #: 035000 K4.06 Tier 2 Group 2: Steam Generator System
Knowledge of S/GS design feature(s) and/or interlock(s) which provide for the following: S/G pressure
Importance 3.1 / 3.4

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
EXPLAIN how changes in plant conditions will effect the Main Steam System.

Question Pedigree
Bank Question used on the 2004 NRC exam. Two distractors changed. Not counted as modified question.

K/A Fit:
Question addresses automatic operation of S/G safety valves and setpoints.

Choice A:
Distractor: Plausible because red light will light if MS-291 and MS-292 are opened from the control room. Incorrect because red light will not light if the open due to high pressure.

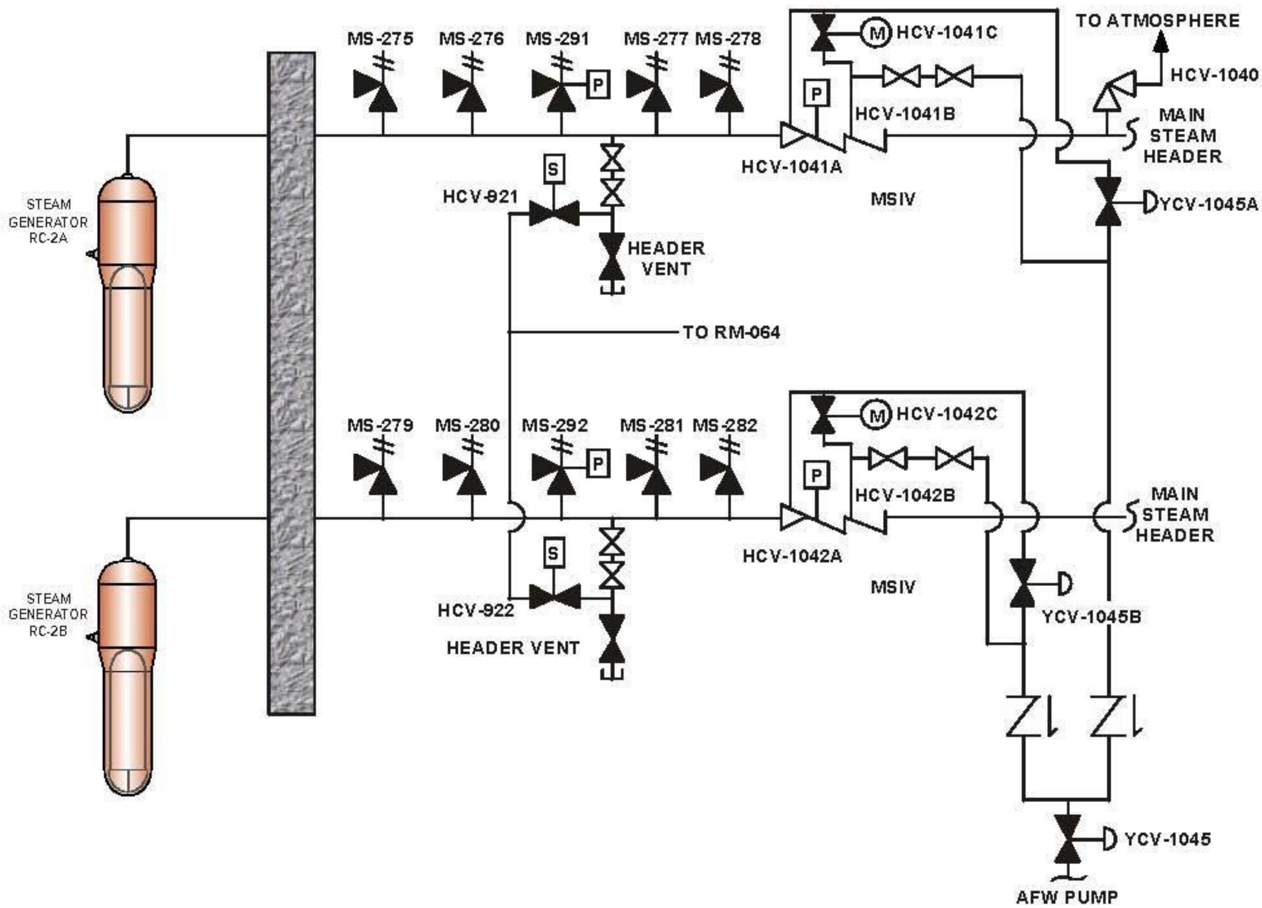
Choice B:
Distractor: Plausible because DCS display added many new indications. Incorrect because the positions of MS-291 and MS-292 are not indicated on DCS.

Choice C:
Distractor: Plausible because the steam dump and bypass system normally controls S/G pressure at 900 psia. Incorrect because both MSIVs are closed.

Choice D:
Correct answer. The opening setpoint for MS-291 and MS-292 is 1000 psia.

KA#:	035000 K4.06	Bank Ref #:	07-11-17 032
LP# / Objective:	0711-17 01.06	Exam Level:	RO-5
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM 25	Handout:	NONE

Figure 2-12 - Main Steam System Diagram



2.3.2 (continued)

C. Each valve starts to open when the steam pressure reaches the initial setpoint and will be fully open when the steam pressure reaches 103% of the initial setpoint value (3% accumulation). The safety valves have a blowdown of 4% which means the valve is fully closed when the steam pressure drops below 96% of the initial relief setpoint. When all five main steam safety valves on one main steam header are simultaneously wide open, they are capable of relieving 3,302,500 lbm/hr. The actual setpoints are shown below:

Table 1 - Main Steam Safety Valve Setpoints

Valves	Setpoints
MS-291/292	1000 psia
MS-275/279	1015 psia
MS-276/280	1025 psia
MS-277/281	1040 psia
MS-278/282	1050 psia

2.3.2 (continued)

D. Safety valves MS-275 through MS-282 (total of eight) are each 6-inch valves, capable of discharging 794,062 pounds mass of steam per hour (lbm/hr). The remaining two 2.5-inch valves (MS-291 and MS-292) each have a discharge capacity of 126,299 lbm/hr. Valves MS-291/292 have a pneumatic operator to allow opening the valves from the Control Room. This allows the steam generators to be used for plant cooldown or RCS temperature control if the MSIVs are closed or the main condenser is otherwise unavailable.

E. The handswitches for MS-291/292 are located on CB-10/11 and have three positions: OPEN, NORMAL, or CLOSE. On loss of air or control power, the pneumatic operator will fail shut. The power for solenoid operation comes from the 125V DC buses. The solenoid is energized to admit control air to the pneumatic operating cylinder to open the valve.

2.3.3 Location

A. All ten main steam safety valves are located in Room 81 of the Auxiliary Building on the 1039-foot elevation. Each steam header has five safety valves.

2.3.4 Power Supplies

None

2.3.5 Instrumentation and Control

None

A. Local

1. Main steam safety valves MS-291 and MS-292 are provided with non-CQE air accumulators to allow valve operation after a reactor trip for cooldown purposes. The time limit the pneumatic operator will remain operable is not specified. Manual Lift Levers are provided for manual operation to continue a controlled cooldown.

B. Remote

1. The steam safety valves MS-291/292 each have a three-position (CLOSE, NORM and OPEN) control switch and valve position status lights (green-closed and red-open) on the vertical section of CB-10/11 in the Control Room.

QUESTIONS REPORT
for ILO EXAM BANK

07-11-17 032

Due to a Loss of Instrument Air, both MSIVs (Main Steam Isolation Valves) failed closed. Assuming no operator action, which of the following Control Room indications would indicate that MS-291 and MS-292 (Pneumatically Operated Safety Valves) are open?

A. The red position indication lights on CB-10

B. Safety tail pipe temperature indicating 300°F

C. Red LEDs on the Safety Valve Acoustic Monitors

D.✓ Steam Generator pressure indicating approximately 1000 psia

QUESTION NUMBER: 063

20 minutes after an uncomplicated reactor and turbine trip from full power, RCS T-cold stabilizes at 532°F.

Which of the following are the expected positions of the steam dump valves (TCV-909A/B/C/D) and the turbine bypass valve (PCV-910)?

A.✔ The steam dump valves will be closed and the turbine bypass valve will modulate to maintain RCS T-cold at approximately 532°F

B. The steam dump valves and the turbine bypass valves will all modulate to maintain the RCS T-cold at approximately 532°F

C. The turbine bypass valve will be closed and one of the steam dump valves will modulate to maintain RCS T-cold at approximately 532°F

D. The turbine bypass valve will be closed and all of the steam dump valves will modulate to maintain RCS T-cold at approximately 532°F

Question # 63 Rev 0

KA #: 041000 K1.05 Tier 2 Group 2: Steam Dump System and Turbine Bypass Control
Knowledge of the Physical connections and/or cause-effect relationships between the SDS and the following systems: RCS
Importance 3.5 / 3.6

CFR Number: 55.41(b)(4)
Secondary coolant and auxiliary systems that affect the facility.

Fort Calhoun Objective:
EXPLAIN the operation of the Reactor Regulating System.

Question Pedigree
Modified bank question.

K/A Fit:
Question addresses how steam dump and bypass valves operate to control RCS temperature.

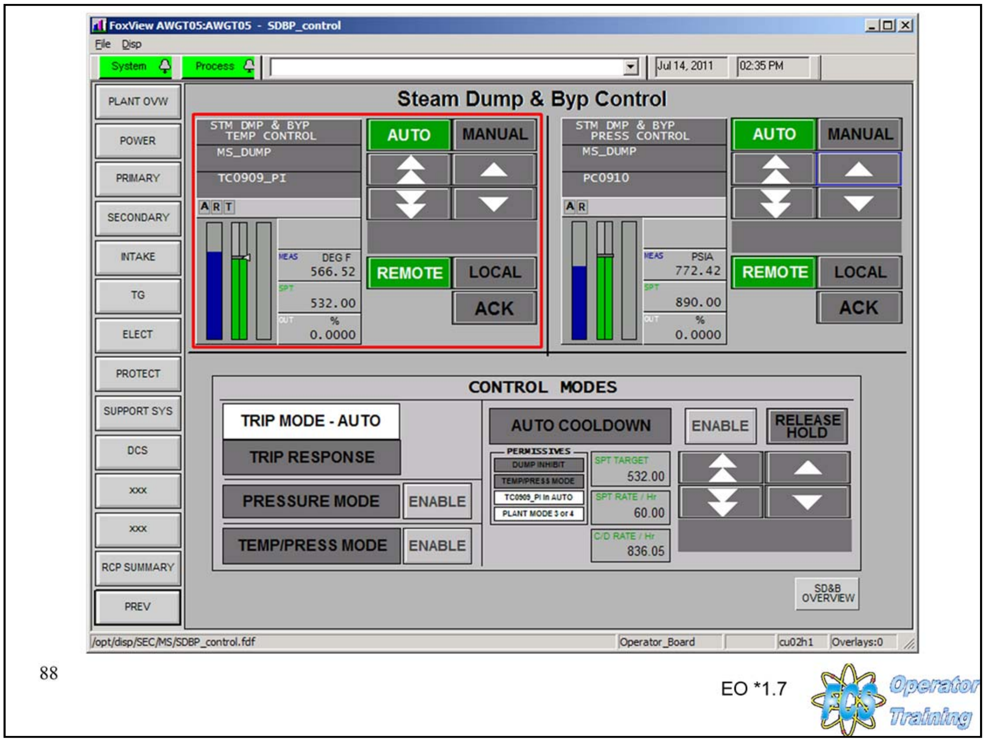
Choice A:
Correct answer: 20 minutes after a reactor trip, the turbine bypass valve, PCV-910, will be the only valve open.

Choice B:
Distractor: Plausible if the Applicant does not realize how the steam dump and bypass valve positions are staged by DCS. Incorrect because only PCV-910 will be open.

Choice C:
Distractor: Plausible if the Applicant does not realize how the steam dump and bypass valve positions are staged by DCS. Incorrect because only PCV-910 will be open.

Choice D:
Distractor: Plausible if the Applicant does not realize how the steam dump and bypass valve positions are staged by DCS. Incorrect because only PCV-910 will be open.

KA#:	041000 K1.05	Bank Ref #:	07-12-31 015
LP# / Objective:	0712-31 02.00	Exam Level:	RO-4
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	SYSTEM LESSON PLAN 3	Handout:	NONE



EO *1.7

Major Component Description

Steam Dump and Turbine Bypass Valves (TCV-909-1/2/3/4 and PCV-910)

The valves are operated in automatic or manual mode as selected at DCS workstations in the Control Room.

The control system utilizes RCS T_{ave} and main steam pressure to control the steam dump and bypass valves via staging to maintain steam header pressure/RCS temperature at the required setpoint.

Normal valve staging is as follows:

(1) PCV-910 opens first

(2) When needed TCV-909-1 opens

(3) Followed by TCV-909-2

(4) Then TCV-909-3

(5) Finally TCV-909-4.

When less steaming is required, the valves will be staged closed in reverse order.

Source question for question 63.

QUESTIONS REPORT
for ILO EXAM BANK

07-12-31 015

Following a reactor and turbine trip from full power, main steam pressure stabilizes at 900 psia. With regard to the Steam Dump and Turbine Bypass valves, which of the following statements is correct?

A.✓ The dumps will be closed and the Bypass valve will modulate to hold 900 psia.

B. The dumps will be partially open and the Bypass valve will modulate to hold 900 psia.

C. The dumps and the Bypass valve will modulate to hold 900 psia.

D. The dumps will be closed and the Bypass valve will remain wide open until pressure is < 750 psia.

QUESTION NUMBER: 064

An Operator was involved in the performance of OI-WDL-5, Attachment 2, "Spent Resin Transfer and Liner Dewatering in RadWaste Bldg."

During this operation he spent 45 minutes in an area with a general area dose rate of 12 mrem/hr before he inadvertently spent 1 minute above the level of the process shield where the dose rate was 75 R/hr.

What is the Operator's total estimated dose for this operation?

A. 9 mrem

B. 10 mrem

C. 1250 mrem

D.✓ 1259 mrem

Question # 64 Rev 0

KA #: 068000 K5.03 Tier 2 Group 2: Liquid Radwaste System
Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: Units of radiation, dose, and dose rate
Importance 2.6 / 2.6

CFR Number: 55.41(b)(12)
Radiological safety principles and procedures.

Fort Calhoun Objective:
SOLVE stay-time and dose problems.

Question Pedigree
New question.

K/A Fit:
Question addresses dose due to liquid radwaste operation and requires knowledge of dose and dose rate units.

Choice A:
Distractor: Plausible because this is the dose assuming only the first 45 minutes. Incorrect because it is not the correct dose.

Choice B:
Distractor: Plausible if the Applicant uses 75 mr/hr instead of R/hr. Incorrect becasue it is not the correct dose.

Choice C:
Distractor: Plausible if the Applicant only considers dose from the 75 R/hr exposure. Incorrect because it is not the total dose.

Choice D:
Correct answer: The total dose is (12 mr/hr X 0.75 hrs) +(75000 mr/hr x 01667 hrs)

KA#:	068000 K5.03	Bank Ref #:	N/A
LP# / Objective:	1924-03A 06.02	Exam Level:	RO-12
Cognitive Level:	HIGH	Source:	NEW
Reference:	OI-WDL-5, ATT 2	Handout:	NONE

Attachment 2 - Spent Resin Transfer and Liner Dewatering in RadWaste Bldg

PROCEDURE (✓) INITIALS

1. IF desired,
THEN perform Spent Resin Transfer and Liner Dewatering as follows:

NOTE

The following steps are categorized for easy reference. Complete the steps in any order and frequency to keep resin flowing. Maintain correct levels in the SRST and the disposal container. Maintain frequent dose rate surveys in accessible areas during resin transfer.

CAUTIONS

1. Minimize time spent above the level of the process shield. During resin transfer radiation levels could exceed 100 R/HR.
2. Closely monitor all tank and container levels. Do not overfill.

a. Close WD-533, Spent Resin Storage Tank WD-33 Gas Analyzer Valve (Room 28). ________

CAUTION

Do not let Vent Header pressure exceed 2.25 psig.

b. Open WD-539, Spent Resin Storage Tank WD-33 Nitrogen Sparging Valve to break up hard packed resin in the SRST while ensuring flow through FI-644. ________

c. Monitor Vent Header pressure and Operate Waste Gas Compressors as necessary to maintain pressure less than 2 psig. ________

d. Continue to nitrogen sparge the SRST for a minimum of 10 minutes. ________

e. When N2 sparging is complete, close WD-539. ________

f. Throttle Open HCV-693, Spent Resin Tk Recycle Valve (AI-101B, Drumming Panel) as necessary to provide sufficient backpressure, but not to exceed 15 psig on PT-635 during the following flush and resin transfer steps. ________

QUESTION NUMBER: 065

PCV-1753, Service Air System Automatic Isolation Valve, automatically closes when _____ air pressure is ________ psig or less.

A. service, 70

B. service, 80

C. instrument, 70

D.✓ instrument, 80

Question # 65 Rev 0

KA #: 079000 A4.01 Tier 2 Group 2: Station Air System
Ability to manually operate and/or monitor in the control room: Cross-tie valves with IAS
Importance 2.7 / 2.7

CFR Number: 55.41(b)(4)
Secondary coolant and auxiliary systems that affect the facility.

Fort Calhoun Objective:
Given the caution statements and/or notes listed in this AOP, explain the reason for each.

Question Pedigree
Bank Question.

K/A Fit:
Question addresses monitoring automatic operation of Service air isolation valve.

Choice A:
Distractor: Plausible if Applicant believes the valve closes on service air pressure and does not know the setpoint. Incorrect because the valve closes when instrument air pressure drops to 80 psig.

Choice B:
Distractor: Plausible if Applicant believes the valve closes on service air pressure and knows the setpoint. Incorrect because the valve closes when instrument air pressure drops to 80 psig.

Choice C:
Distractor: Plausible if Applicant knows the valve closes on instrument air pressure but does not know the setpoint. Incorrect because the valve closes when instrument air pressure drops to 80 psig.

Choice D:
Correct answer: The valve closes when instrument air pressure drops to 80 psig.

KA#:	079000 A4.01	Bank Ref #:	07-11-07 016
LP# / Objective:	0717-10 01.05	Exam Level:	RO-4
Cognitive Level:	LOW	Source:	BANK
Reference:	AOP-17	Handout:	NONE

INSTRUCTIONS		CONTINGENCY ACTIONS	
3.	**IF** the "AIR DRYER TROUBLE" alarm (CB-10,11; A11) annunciates, **THEN** direct an operator to proceed to AI-79 to determine the cause of the alarm (Room 19).		
4.	Direct all available operators to search for the source of the air leakage.		
5.	**IF** Instrument Air pressure is less than 80 psig, **THEN** verify PCV-1753, "SERVICE AIR SYSTEM AUTOMATIC ISOLATION VALVE", is closed (Room 19).	5.1	**IF** PCV-1753 is **NOT** closed, **THEN** close CA-121, "SERVICE AIR SUPPLY SYSTEM MANUAL ISOLATION VALVE" (Room 19).
6.	**IF** Instrument Air pressure returns to a normal 98-108 psig after isolating Service Air, **THEN** GO TO Section 5.0, Exit Conditions.		
7.	**IF** Instrument Air pressure is less than 78 psig, **THEN** verify that PCV-1752, "AIR DRYERS CA-31 & CA-12 BYPASS VALVE", is open (Room 19).	7.1	**IF** Instrument Air pressure is less than 78 psig **AND** PCV-1752 is closed, **THEN** open CA-197, "BYPASS CONTROL VALVE PCV-1752 BYPASS VALVE" (Room 19).

QUESTION NUMBER: 066

Which one of the following meets the requirements for personnel within the "Control Room Boundary" whenever fuel is in the reactor?

A. One RO

B. One CRS

C.✓ One RO and one Shift Manager.

D. Two RO's and one unlicensed STA.

Question # 66 Rev 0

KA #: 000000 2.1.01 Tier 3 Group 4:
Generic Knowledges and Abilities
Knowledge of conduct of operations requirements.
Importance 3.8 / 4.2

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
DISCUSS Personnel Presence and Conduct in the Control Room.

<u>Question Pedigree</u>
Modified question. Source question was used on the 2001-1 NRC exam.

<u>K/A Fit:</u>
Question addresses required control room staffing.

<u>Choice A:</u>
Distractor: Plausible if Applicant confuses with requirements for the "At the controls area." Incorrect because at least one RO and one SRO (CRS or Shift Manager) is required.

<u>Choice B:</u>
Distractor: Plausible if Applicant confuses with requirements for the "At the controls area." Incorrect because at least one RO and one SRO (CRS or Shift Manager) is required.

<u>Choice C:</u>
Correct answer: One RO and one SRO are required.

<u>Choice D:</u>
Distractor: Plausible if Applicant does not understand that a SRO is also required.

KA#:	000000 2.1.01	Bank Ref #:	ADM-OPS 018
LP# / Objective:	0767-05 04.00B	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	MODIFIED
Reference:	SO-O-1	Handout:	NONE

5.9.2.D ERF Computer - See OI-ERFCS-1

5.9.2.E Operations Memorandums - See SO-O-13.

5.9.2.F Technical Data Book

5.9.2.G Temporary Labels, Curves, Notes, and Instructions - See SO-O-41.

5.10 Operations Shift Manning

5.10.1 Responsibility

NOTE: The RERP Table B-1 requires one STA and Control Room Communicator in all modes of operation for ERO duties.

5.10.1.A It is the responsibility of the Shift Manager to ensure that the required number of Licensed and Equipment Operators are available.

5.10.1.B Additional Operations Department personnel may be required on shift because of unusual plant conditions or operational needs. The Shift Manager shall obtain the additional personnel as necessary. Activities requiring additional personnel will not be undertaken until the required personnel are available.

5.10.2 Staffing Requirements (Minimum)

5.10.2.A Shift manning will be as specified in Technical Specification 5.2.2.a, b, c, and e. When overtime is required for shift coverage, then the SM will use the operator overtime list as a guideline. When overtime is not required for on-shift needs, additional personnel shall be scheduled, as appropriate, on the weekly schedule.

5.10.2.B The Operations Call List (located in the Shift Manager's Office), the Duty Assignment Call List or the Operator overtime list will be used by the Shift Manager to facilitate the recall of personnel as required.

5.10.2.C In the event it is anticipated that these requirements cannot be satisfied, the Manager-Operations, Manager-Shift Operations, or the Manager-Fort Calhoun Station or designated alternate shall be notified immediately.

5.10.2.D Technical Specification 5.2 designates Licensed Operator placement as follows:

5.10.2.D.1) Two Licensed individuals (Shift Manager, Control Room Supervisor, or Reactor Operator) must be within the Control Room boundary, as defined in Attachment 7.1 at all times whenever fuel is in the reactor except as noted in Step 5.10.2.E.

5.10.2.D.2) The placement of the two Licensed Operators required above must also meet the following requirements;

5.10.2.D.2)a) Either the SM or the CRS must be within the Control Room boundary, as defined in Attachment 7.1 at all times whenever fuel is in the reactor except as noted in Step 5.10.2.E.

5.10.2.D.2)b) One RO or SRO should be within the at the controls area as defined in Attachment 7.1 at all times whenever fuel is in the reactor except as noted in Step 5.10.2.E. Further clarification is provided in the OPD (Procedure Maintenance and Ownership).

5.10.2.D.2)c) Occasionally, two SRO's are present in the Control Room. It should be clearly understood that there is only one official CRS on watch at any one time and that the functional duties of the two SRO's are not to be swapped back and forth on the same shift. The official CRS is the one who accepted the turnover from his/her counterpart at the beginning of the shift and who signed into the shift turnover log as the oncoming CRS. This CRS shall remain as the official CRS for the duration of the shift unless properly relieved by the other CRS on-shift.

5.10.2.E The (2) Licensed members of the plant staff shall be in the Control Room at all times except during off-normal operation conditions having potential safety related impact which could jeopardize the health and safety of the public and which requires prompt operator actions outside the Control Room complex. Safety related impact is subject to Operator discretion at the time of the off-normal operating condition.

5.10.2.F Shift crew assignments during periods of core alterations shall include a Licensed Senior Reactor Operator to directly supervise the core alterations. This Licensed Senior Reactor Operator shall not have other concurrent duties.

5.10.2.G The Manager-Shift Operations will determine the appropriate amount of management oversight during critical evolutions, and during Refueling Outages. This oversight will be documented in an Operations RFO Organizational Plan. This plan will normally designate a shutdown SRO, in case one is needed, and will provide resources for oversight in addition to the SM and CRS both in the Control Room and in the field as appropriate.

QUESTIONS REPORT
for ILO EXAM BANK

ADM-OPS 018

What are the minimum requirements for personnel within the "at the controls" area whenever fuel is in the reactor?

A.✔ One licensed individual (RO or SRO).

B. One SRO.

C. One RO and one SRO.

D. Two licensed individuals (RO or SRO).

QUESTION NUMBER: 067

The Shift Manager has implemented the two-person rule for vital areas per AOP-37, "Security Events."

Assuming you are one of two extra operators in the control room, which one of the following actions would require you to take an additional operator with you?

A.✓ Acknowledging an annunciator alarm on AI-100, Alarm Panel A50

B. Acknowledging an annunciator alarm on AI-134, Alarm Panel A30-2

C. Verifying the position of Auxiliary Feedwater Pump FW-54 engine, FW-56, throttle lever.

D. Verifying the position of Quadvoter solenoid valves, PCV-3303A./B and PCV-3303C/D.

Question # 67 Rev 0

KA #: 000000 2.1.13 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of facility requirements for controlling vital / controlled access.
Importance 2.5 / 3.2

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
DESCRIBE how the plant operators respond to Security Events.

Question Pedigree
New question

K/A Fit:
Question addresses requirements for access to vital areas.

Choice A:
Correct answer: AI-100 is located in a vital area.

Choice B:
Distractor: Plausible if Applicant believes AI-134 is located in a vital area. Incorrect because AI-134 is located within a non-vital area of the turbine building.

Choice C:
Distractor: Plausible because FW-54 is important to safety per the PRA and is located in its own room. Incorrect because the FW-54 room is not a vital area.

Choice D:
Distractor: Plausible if Applicant believes the Quadvoter is located in a vital area. Incorrect because the Quadvotor is located within a non-vital area of the turbine building.

Note: AOP-37 is contained on the "SUNSI" CD.

KA#:	000000 2.1.13	Bank Ref #:	N/A
LP# / Objective:	0717-37 01.04	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	AOP-37	Handout:	NONE

QUESTION NUMBER: 068

The reactor has been operating at 100% power for three months. In accordance with OPD-4-19, "Reactivity Management," the "Shift Reactivity Briefing" is required to include discussion of:

A. Core xenon reactivity

B.✓ RCS boron concentration

C. Moderator temperature coefficient

D. Available shutdown margin

Question # 68 Rev 0

KA #: 000000 2.1.37 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of procedures, guidelines, or limitations associated with reactivity management.
Importance 4.3 / 4.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
DESCRIBE the Performance Standards listed in the OPD Manual.

Question Pedigree
New question.

K/A Fit:
The question addresses the required contents of a reactivity briefing.

Choice A:
Distractor:Plausible because this is a significant reactivity parameter. Incorrect because it is not required by OPD-4-19.

Choice B:
Correct answer. OPD-4-19 requires that RCS boron concentration be discussed during a reactivity briefing.

Choice C:
Distractor:Plausible because this is a significant reactivity parameter. Incorrect because it is not required by OPD-4-19.

Choice D:
Distractor:Plausible because this is a significant reactivity parameter. Incorrect because it is not required by OPD-4-19.

KA#:	000000 2.1.37	Bank Ref #:	N/A
LP# / Objective:	0767-05 02.00	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	OPD-4-19	Handout:	NONE

Table 3 - Shift Reactivity Briefing Checklist

Review Current Reactivity Status	
□	RCS boron concentration and neutral blend/batch
□	Last shift's and expected reactivity additions
□	Reactivity thumb-rules
Discuss Potential External Impacts on Reactivity	
□	Weather/River conditions affecting Condenser operations
□	Raw Water and CCW operations
□	Electrical Grid conditions
□	Are there any contingency actions required?
Discuss Planned Activities and Reactivity Impacts	
□	Reactor power changes for activities (e.g., Control Valve testing)
□	Feedwater/Condensate/Heater Drain system operations
□	Main Steam/Steam Dump and Bypass/SG Blowdown operations
□	Turbine EHC operations
□	CVCS operations
□	Other items (e.g., testing on critical components/systems)
□	Are there any contingency actions required?
Discuss Operator Challenges with Reactivity Impacts	
□	Review Operator Challenge List from Plan of the Day for reactivity impacts
□	Reactivity control or feedwater controls system deficiencies
□	Annunciator/Alarm deficiencies
□	Are there any contingency actions required?
Input/Questions? Follow-up Actions?	
□	Get input from all personnel in attendance
□	Do any of the above items require RE, Reactivity SRO, or other support?

The purpose of the Shift Reactivity Briefing Checklist is to provide a consistent method for crew reactivity briefings in accordance with SO-G-117, Reactivity Management. These briefings should occur at the beginning of each shift and at the discretion of the SM/CRS should plant conditions warrant.

The briefing is led by the CRS, with input provided by the LOs, STA, and SM. The CRS, or an LO as designated by the CRS, should review the checklist in a section by section manner. Each section provides a list of items intended to stimulate discussion during the briefing.

The contingency actions discussion should include items such as:
- Critical parameters to monitor when applicable,
- Post-trip actions that would be different due to plant/system/component status, and
- Reactivity-related AOP/EOP actions that would be different due to plant/system/component status.

The beginning-of-shift reactivity is not a substitute for a reactivity briefing for a specific activity (e.g., MTC testing).

QUESTION NUMBER: 069

According to Standing Order G-30, "Procedure Changes and Generation," who is responsible for ensuring that the 50.59 and/or 72.48 screening is completed for a proposed procedure change?

A. The Shift Manager or CRS.

B.✓ The Qualified Reviewer.

C. The PRC Technical Secretary.

D. The Document Control Supervisor.

Question # 69 Rev 0

KA #: 000000 2.2.06 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of the process for making changes in procedures.
Importance 3.0 / 3.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective
STATE some of the activities, covered by Standing Orders, which require written procedures per Regulatory Guide 1.33.

Question Pedigree
Bank question used on the 2007 NRC exam.

K/A Fit:
Question addresses responsibilities for procedures changes.

Choice A:
Distractor: Plausible if Applicant does not understand this is the responsibility of the Qualified Reviewer. Incorrect because the Qualified Reviewer is responsible.

Choice B:
Correct answer: The Qualified Reviewer is responsible for this action.

Choice C:
Distractor: Plausible if Applicant does not understand this is the responsibility of the Qualified Reviewer. Incorrect because the Qualified Reviewer is responsible.

Choice D:
Distractor: Plausible if Applicant does not understand this is the responsibility of the Qualified Reviewer. Incorrect because the Qualified Reviewer is responsible.

KA#:	000000 2.2.06	Bank Ref #:	ADM-CONTROL 031
LP# / Objective:	0762-01 02.00	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	BANK
Reference:	SO G-30	Handout:	NONE

3.3.9 Ensuring proposed procedure changes which are determined to involve a change in the Technical Specifications are submitted to the NRC for approval prior to implementation.

3.4 The Qualified Reviewer is responsible for:

3.4.1 Ensuring proper review of proposed procedure changes and documenting recommendations for approval/disapproval in accordance with guidance of Step 4.17.

3.4.2 Ensuring the 50.59 and/or 72.48 Screen and, if applicable, 50.59 and/or 72.48 Evaluation is completed as required by NOD-QP-3. The review may be performed by the QR or may be performed prior to the QR review process. (AR 19408)

3.4.3 Ensuring proposed procedure changes are reviewed in a timely manner.

3.4.4 Ensuring completeness and thoroughness of any cross-disciplinary/ cross-functional or other reviews.

3.5 Document Control is responsible for:

3.5.1 Distributing Operating Manual procedures in a timely manner.

3.5.2 Maintaining and issuing the current index of Operating Manual procedures.

3.5.3 Maintaining records required by this procedure.

3.5.4 Distributing a copy of Setpoint Change Review Form (PED-SEI-9.1) and 50.59 Screen/Evaluation (if applicable) to the Manager-Design Engineering Nuclear following approval of procedure changes involving any changes to instrument setpoint, tolerance, or acceptance criteria as delineated in PED-SEI-9.

3.5.5 Notifying on-shift Shift Manager if an OI Checklist has been revised and may require performance of the OI Checklist immediately after issuance (as specified by the QR). (CR 199701648)

3.6 The Plant Review Committee (PRC) is responsible for:

3.6.1 Reviewing procedure or procedure changes which require a 50.59 or 72.48 Evaluation.

3.6.2 Reviewing those procedure changes determined as a Very High Risk Activity.

3.6.3 Reviewing candidates to serve as Qualified Reviewers and recommending approval.

QUESTION NUMBER: 070

Safety Limit 1.1.1(a) states:

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

What is the basis for this Safety Limit?

A. To ensure that DNBR during normal operation is not greater than the DNBR limit.

B.✓ To ensure that DNBR during normal operation is not lower than the DNBR limit.

C. To ensure that DNBR during a design basis LOCA is not greater than the DNBR limit.

D. To ensure that DNBR during a design basis LOCA is not lower than the DNBR limit.

Question # 70 Rev 0

KA #: 000000 2.2.25 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of bases in Technical Specifications for limiting conditions for operations and safety limits.
Importance 3.2 / 4.2

CFR Number: 55.41(b)(5)
Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons.

Fort Calhoun Objective:
STATE the two (2) safety limits and the basis for each.

Question Pedigree
New question.

K/A Fit:
Question addresses knowledge of the basis for a safety limit.

Choice A:
Distractor: Plausible if Applicant does not realize that DNBR must stay above the limit.

Choice B:
Correct answer: DNBR must stay above the limit for normal operation.

Choice C:
Distractor: Plausible if Applicant does not realize that DNBR must stay above the limit and believes the limit is for a LOCA instead of normal operation.

Choice D:
Distractor: Plausible if Applicant believes the DNBR limit applies to a LOCA.

KA#:	000000 2.2.25	Bank Ref #:	N/A
LP# / Objective:	0762-08 03.00	Exam Level:	RO-5
Cognitive Level:	LOW	Source:	NEW
Reference:	TS 1.1.1(A) BASIS	Handout:	NONE

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs)

1.1.1 Reactor Core SLs

Applicability

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

Objective

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

Specifications

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

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

1.1.2 Reactor Coolant System Pressure SL

Applicability

Applies to the limit on reactor coolant system pressure.

Objective

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

Specification

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

1.0 **SAFETY LIMITS**

1.1 Safety Limits (SLs) (continued)

Basis

To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the reactor coolant, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB).

At DNB there is a sharp reduction of the heat transfer coefficient, which would result in high clad temperature and the possibility of clad failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of reactor thermal power and reactor coolant flow, temperature and pressure can be related to DNB through a correlation. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients corresponds to a 95% probability at a 95% confidence level that DNB will not occur, which is considered an appropriate margin to DNB for all operating conditions.[(1)]

The curves of Figure 1-1 represent the loci of points for reactor thermal power (either neutron flux instruments or ΔT instruments), reactor coolant system pressure, and cold leg temperature for which the minimum DNBR is not less than the minimum DNBR limit. The area of safe operation is below these lines.

SL 1.1.1(b) ensures that fuel centerline temperature remains below the fuel melt temperature 5081°F during normal operating conditions or design anticipated operational occurrences (AOOs) with adjustments for burnup and burnable poison. An adjustment of 58°F per 10,000 MWD/MTU has been established in XN-NF-82-06(P)(A), Revision 1, Supplements 2, 4 and 5 (Ref. 8) and adjustments for burnable poisons are established based on XN-NF-79-56(P)(A), Revision 1, Supplement 1 (Ref. 9).

The reactor core safety limits are based on radial peaks limited by the CEA insertion limits in Section 2.10 and axial shapes within the axial power distribution trip limits in the COLR. The Thermal Margin/Low Pressure trip requirements shall be within the limits provided in the COLR. The Thermal Margin/Low Pressure trip is based on an unrodded integrated total radial peak (F_R^T) that is provided in the COLR.

QUESTION NUMBER: 071

How will placing the key switch on a Control Room process radiation monitor ratemeter in the "KEYPAD" position affect operation of the radiation monitor?

A. The ratemeter will display 0 cpm and equipment actuations will be blocked.

B.✓ The ratemeter will display actual countrate but equipment actuations will be blocked.

C. The ratemeter and equipment actuations will still work but output to the ERF computer will be blocked.

D. Operation of the radiation monitor will not be affected.

Question # 71 Rev 0

KA #: 000000 2.3.05 Tier 3 Group 4: Generic Knowledges and Abilities
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
Importance 2.9 / 2.9

CFR Number: 55.41(b)(11)
Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Fort Calhoun Objective:
EXPLAIN the operations, actuations and applications of the individual radiation monitors.

Question Pedigree
Bank question used on the 2009 NRC exam. The wording of the choices has been changed but not counted as a modified question.

K/A Fit:
Question addresses the operation of installed radiation monitors control pads.

Choice A:
Distractor: Plausible if Applicant believes the display will go to zero. Incorrect because it will display the actual countrate.

Choice B:
Correct answer: The ratemeter will display actual countrate but equipment actuations will be blocked.

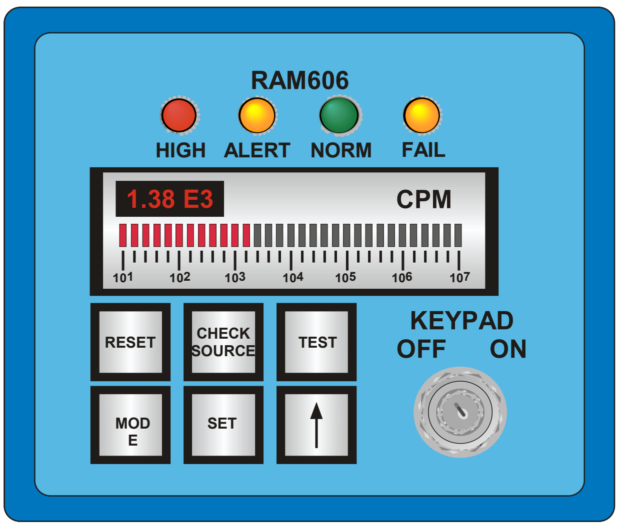
Choice C:
Distractor: Plausible if Applicant believes only the output to the ERF computer will be blocked. Incorrect because equipment actuations will be blocked.

Choice D:
Distractor: Plausible if Applicant believes placing ratemeter in keypad will have no effect. Incorrect because equipment actuations will be blocked.

KA#:	000000 2.3.05	Bank Ref #:	07-12-03 032
LP# / Objective:	0712-03 04.00	Exam Level:	RO-11
Cognitive Level:	LOW	Source:	BANK
Reference:	STM-33	Handout:	NONE

2.49 When the Control Room ratemeter keyswitch is in the OFF position, the local ratemeter will be unaffected and continue to provide a local read-out of activity and alarms only; however, the process signal will not be passed on to the ERF computer process radiation monitor recorder, high radiation and trouble alarms on AI-33C, and external relays for system actuation.

2.50 With the Control Room ratemeter keyswitch in the KEYPAD position, the alarms and system actuation relays will not operate. However, activity will still be displayed and sent to the process radiation monitor recorder and the ERF computer.



NOTE: ALSO USED AS LOCAL RATEMETERS ON SKID UNITS RM-050/51, RM-052, RM-062 AND RM-063

FIGURE 2-14: PROCESS RADIATION MONITOR CONTROL ROOM RATEMETER (RAM606)

2.51 The keypad contains the following switches:

2.52 RESET- switch is used to reset acknowledge alarm conditions (lights and actuation relays) when the count rate has returned to below the alert or alarm setpoints. The monitor must be reset before external alarms on AI-33C and external system actuation relays can be reset.

2.53 CHECK SOURCE- switch begins a pre-timed program for check source activation (30 to 120 seconds depending on type of monitor). The CHECK SOURCE program can be activated from the Control Room ratemeter with its keyswitch in the ON position. However, when activating CHECK SOURCE from the local ratemeter, its keyswitch must be in the KEYPAD position. While in

QUESTIONS REPORT
for ILO EXAM BANK

07-12-03 032

How will placing the keyswitch on a Control Room process radiation monitor ratemeter in the "KEYPAD" position affect operation of the radiation monitor?

A.✓ Annunciation and equipment actuations will be blocked.

B. Annunciation will still work but equipment actuations will be blocked.

C. Annunciation and equipment actuations will still work but output to the radiation monitor recorder will be blocked.

D. Annunciation and equipment actuations will still work but output to the ERF computer will be blocked.

QUESTION NUMBER: 072

Why does EOP-04, "Steam Generator Tube Rupture" direct the operators to ensure that FW-268, "Condensate Dump Valve LCV-1193 Outlet Isolation Valve" and FW-266, "Condensate Dump Valve LCV-1193 Bypass Valve" are closed?

A. To ensure that the water level in the condenser hotwell remains high enough to meet the NPSH requirements of the condensate pumps.

B. To ensure that the water level in the condenser hotwell remains high enough to adequately dilute radioactive water from the ruptured steam generator.

C. To isolate flow to the condensate storage tank to prevent it from overflowing.

D.✓ To isolate flow to the condensate storage tank to prevent the spread of contamination.

Question # 72 Rev 0

KA #: 000000 2.3.14 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
Importance 3.4 / 3.8

CFR Number: 55.41(b)(12)
Radiological safety principles and procedures.

Fort Calhoun Objective:
DEMONSTRATE the knowledge required to use EOP-04, Steam Generator Tube Rupture (SGTR), to mitigate the consequences of a SGTR.

Question Pedigree
New question.

K/A Fit:
The question addresses radiation contamination hazards during a S/G tube rupture.

Choice A:
Distractor: Plausible if the Applicant believes these valves are closed to retain water in the hotwell. Incorrect because they are closed to prevent contamination of the CST.

Choice B:
Distractor: Plausible if the Applicant believes these valves are closed to retain water in the hotwell. Incorrect because they are closed to prevent contamination of the CST.

Choice C:
Distractor: Plausible if Applicant believes the valves are closed to prevent overfilling the CST. Incorrect because they are closed to prevent contamination of the CST.

Choice D:
Correct answer: They are closed to prevent contamination of the CST.

KA#:	000000 2.3.14	Bank Ref #:	N/A
LP# / Objective:	0718-14 01.00	Exam Level:	RO-12
Cognitive Level:	HIGH	Source:	NEW
Reference:	TBD-EOP-04	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

d. Monitor S/G levels.

EPG Step: 12
Deviation: None

In order to minimize the release of radioactivity, the S/G with the tube leak must be identified so it can be isolated when plant conditions for S/G isolation are established. The S/G with the tube leak may be determined by some or all of the following:

- Sampling the S/Gs indicates higher activity in one of the S/Gs
- Monitoring the steam headers indicate higher activity in one of the steam headers
- Monitoring the S/G blowdown indicates higher activity in one of the blowdown lines
- For a large SGTR, a rise in the level of the affected S/G may be noticed or a constant level may be observed while a mismatch between feed rate and steaming rate exists.

✱ 13. Minimize the spread of contamination by performing the following steps:

a. Ensure HCV-2509, "SAMPLE DRAIN TO DRAIN HEADER", is open (AI-107, Room 60).

b. Ensure HCV-2508, "SAMPLE DRAIN TO CONDENSER C.W. TUNNEL", is closed (AI-107, Room 60).

c. Ensure FW-268, "CONDENSATE DUMP VALVE LCV-1193 OUTLET ISOLATION VALVE", is closed (Turbine Building Mezzanine).

d. Ensure FW-266, "CONDENSATE DUMP VALVE LCV-1193 BYPASS VALVE", is closed (Turbine Building Mezzanine).

INSTRUCTIONS | CONTINGENCY ACTIONS

EPG Step: None

Deviations:
1) The EOP provides direction for realigning blowdown sample discharge from the Raw Water system to the Waste Disposal system.
2) The EOP isolates Condensate dump to the storage tank.

Blowdown sample flow is normally discharged to the Raw Water system which discharges to the river. During a SGTR event the sample flow is redirected to the Spent Regenerant tank to maintain any radioactive sample within the plant's waste disposal system. This justifies deviation 1.

Condensate dump to the storage tank is isolated to prevent the spread of contamination. This justifies deviation 2.

✗ 14. **WHEN** RCS T_H is less than or equal to 510°F,
THEN isolate the most affected S/G by performing the following steps:

a. **IF** RC-2A is most affected,
THEN isolate RC-2A by performing the following steps:

a.1 **IF** RC-2B is most affected,
THEN isolate RC-2B by performing the following steps:

QUESTION NUMBER: 073

Given the following plant conditions:

- A LOCA occurred 5 hours ago
- RCS pressure is 150 psia and lowering
- SIRWT Level is 32 inches and lowering
- Pressurizer Level is 50% and rising slowly
- RCS T-cold is 325°F

What condition must change before EOP/AOP Attachment 7, “SDC with RAS” can be implemented?

A. 30 additional minutes must elapse

B. RCS pressure must be 140 psia or less

C.✓ SIRWT level must be 16 inches or less

D. RCS T-cold must be 300°F or less

Question # 73 Rev 0

KA #: 000000 2.4.02 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
Importance 4.5 / 4.6

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
GIVEN a copy of Attachment 7, EXPLAIN the steps necessary to initiate Shutdown Cooling Operations with RAS.

Question Pedigree
New question.

K/A Fit:
Question addresses setpoint entry conditions for an EOP attachment that is common to different EOPs.

Choice A:
Distractor: Plausible because this is required for EOP/AOP Attachment 9. Incorrect because it is not required for Attachment 7.

Choice B:
Distractor: Plausible because this is required for EOP/AOP Attachment 9. Incorrect because it is not required for Attachment 7.

Choice C:
Correct answer. SIRWT level must be less than the STLS setpoint, 16 inches, before RAS can occur.

Choice D:
Distractor: Plausible because RCS pressure must be 300 psia or less. Incorrect Attachment 7 can be performed with T-cold at 325°F.

KA#:	000000 2.4.02	Bank Ref #:	N/A
LP# / Objective:	0718-13 02.04	Exam Level:	RO-10
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-03	Handout:	NONE

INSTRUCTIONS

✘34. **IF** SIRWT level falls to 16 inches, **THEN** verify that STLS initiates RAS by performing the following:

a. Ensure **BOTH** of the following valves are open:

- HCV-383-3, SI Pump Suction Containment Isolation Valve
- HCV-383-4, SI Pump Suction Containment Isolation Valve

b. Ensure **ALL** of the following valves are closed:

- LCV-383-1, SI Pump Suction SIRWT Isolation Valve
- LCV-383-2, SI Pump Suction SIRWT Isolation Valve
- HCV-385, SIRWT Recirc Valve
- HCV-386, SIRWT Recirc Valve
- HCV-480, AC-4A CCW Inlet Valve
- HCV-481, AC-4B CCW Inlet Valve
- HCV-484, AC-4A CCW Outlet Valve
- HCV-485, AC-4B CCW Outlet Valve

c. Ensure **BOTH** LPSI pumps stop.

Time: __________

CONTINGENCY ACTIONS

34.1 **IF** RAS is **NOT** actuated by STLS, **THEN** perform the following to manually establish RAS flow paths:

a. Open both SI Pump Suction Containment Isolation Valves, HCV-383-3/4.

b. Close both SI Pump Suction SIRWT Isolation Valves, LCV-383-1/2.

c. Ensure both LPSI Pumps stop, SI-1A/B.

d. Close **BOTH** of the following SIRWT Recirc Valves:

- HCV-385
- HCV-386

Time: __________

c.1 **IF** LPSI pumps are not stopped, **THEN** IMPLEMENT Floating Step B, LPSI Stop and Throttle.

INSTRUCTIONS	CONTINGENCY ACTIONS
✗44. **WHEN ALL** of the following SDC entry conditions are established: • PZR level is greater than or equal to 45% and constant or rising • RCS subcooling is greater than or equal to 20°F • RCS pressure is less than or equal to 300 psia. • RCS T_C less than 350°F **THEN** initiate SDC operation PER **ONE** of the following Attachments: • Attachment 4, SDC Without RAS • Attachment 7, SDC with RAS • Attachment 8, Cooled SI Flow with RAS Time: ___________	44.1 **IF** SDC can **NOT** be established, **THEN** GO TO Step 45.

****

CAUTION

If RAS has occurred, area radiation levels may require access to AI-100 via the Emergency Access Plug.

****

45. Monitor both SI Pump Room Sumps for indication of excessive leakage (ERF Analog Data Points L565, L566).

QUESTION NUMBER: 074

While performing the actions of an AOP, a step in the AOP directs you to "GO TO" an EOP.

How should the procedures be used?

A.✔ Exit the AOP and enter the EOP.

B. Perform the EOP actions in parallel with the AOP actions.

C. Enter the EOP and use the reference material in the AOP.

D. Perform the actions in the EOP and then come back to the AOP.

Question # 74 Rev 0

KA #: 000000 2.4.08 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
Importance 3.8 / 4.5

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
DEMONSTRATE general knowledge about the structure, terminology and usage of the Emergency Operation Procedures (EOP's).

Question Pedigree
New question.

K/A Fit:
Question generically addresses how AOP's are used in conjunction with EOP's.

Choice A:
Correct answer:OPD-4-09 states that "GO TO" means exit this procedure and use the procedure identified after the "GO TO."

Choice B:
Distractor: Plausible if Applicant confuses with "Implement." Incorrect because the AOP must be exited.

Choice C:
Distractor: Plausible if Applicant confuses with "Refer to." Incorrect because the AOP must be exited.

Choice D:
Distractor: Plausible if Applicant does not realize that the AOP must be exited. Incorrect because the AOP must be exited.

KA#:	000000 2.4.08	Bank Ref #:	N/A
LP# / Objective:	0718-10 01.00	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	NEW
Reference:	OPD 4-09	Handout:	NONE

4.19 Declarative Steps

4.19.1 Some steps are merely statements, e.g., "a leak in the Aux Building is indicated". These are declarative steps and only meant as an operator aid.

4.20 Announcements

4.20.1 Announcements appear in quotes following a sentence beginning with the verb **"ANNOUNCE"**.

4.21 Evaluation Order

4.21.1 Steps are to be performed in order of appearance unless otherwise noted. If steps are permitted to be performed out of order, then a note will specify that. Additionally, steps within EOP's are designated as continuously applicable or non-sequential. These steps can be performed in any order as conditions are met, and are not considered deviations.

4.21.2 Time dependent steps will have the time limit in the step. When a set of steps is time dependent, there will be a note to state that ahead of the first step in the set.

4.21.3 Equally acceptable steps are a set of steps any of which may accomplish the goal of the overall step. The order of appearance is most-to-least preferred. Operators should attempt to perform step a. If unable to perform step a, attempt b, etc.

4.22 References

4.22.1 When an EOP/AOP refers to another procedure of any type, it will direct the operator to the appropriate section. When references are made to tables and figures this is not applicable.

4.23 Referencing Terms

4.23.1 **"REFER TO"** indicates that a source of information is available to aid in performance of the current step, at the discretion of the operator.

4.23.2 **"IMPLEMENT"** indicates another set of instructions is to be performed in parallel with the concurrent procedure.

4.23.3 **"PER"** can replace both **"IMPLEMENT"** and **"REFER TO"** where they would be inappropriate, e.g., "Verify RCP NPSH requirements are met **PER** Attachment 2."

4.24 Branching Terms

4.24.1 **"GO TO"** means exit this procedure and use the procedure identified after the **"GO TO"**.

QUESTION NUMBER: 075

Given the following:

- The AI-149 FIRE DETECTION ALARM OR TROUBLE alarm was received
- The Control Room Fire Detection Computer, PC-66, indicates that multiple detectors in the Cable Spread Room are in alarm
- The EONT was dispatched and reports that Halon is being discharged into the Cable Spread Room

In accordance with AOP-06, FIRE EMERGENCY, what action should be taken by the operating crew?

A.✓ Trip the reactor, open clutch power supply breakers, close the PORV block valves and evacuate the Control Room.

B. Trip the reactor, close the PORV block valves, start both Fire pumps, FP-1A and FP-1B, and evacuate the Control Room.

C. Trip the reactor, open clutch power supply breakers, close the PORV block valves and place the Control Room HVAC in Recirculation Mode.

D. Close the PORV block valves, start both Fire pumps, FP-1A and FP-1B, and place the Control Room HVAC in Recirculation Mode.

Question # 75 Rev 0

KA #: 000000 2.4.25 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of fire protection procedures.
Importance 3.3 / 3.7

CFR Number: 55.41(b)(10)
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Fort Calhoun Objective:
Describe how the plant may respond to a fire in the following locations: Upper Electrical Penetration Room.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
The question addresses actions to be taken per FCS Fire Protection Procedures.

Choice A:
Correct answer: AOP-06 requires that the Control Room be evacuated for a Cable Spread Room fire. The reactor is tripped, clutch power supply breakers are opened and PORV block valves are closed prior to evacuating the Control Room.

Choice B:
Distractor: Plausible because it contains some of the correct actions. Incorrect because fire pumps should not be started.

Choice C:
Distractor: Plausible because it includes some correct actions. Incorrect because the Control Room is not evacuated.

Choice D:
Distractor: Plausible because it includes some correct actions. Incorrect because the Control Room is not evacuated.

KA#:	000000 2.4.25	Bank Ref #:	07-17-06 013
LP# / Objective:	0717-06 01.02C	Exam Level:	RO-10
Cognitive Level:	LOW	Source:	BANK
Reference:	AOP-06	Handout:	NONE

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS | CONTINGENCY ACTIONS

1. Direct the Shift Manager to perform Attachment M, Shift Manager Actions in Support of AOP-06 Section II.

2. Perform the following steps prior to evacuating the Control Room:

   a. Manually trip the Reactor.

   b. Open **BOTH** Clutch Power Supply Circuit Breakers (AI-57):

      - "CB-AB"
      - "CB-CD"

   c. Close **BOTH** of the following PORV Block Valves:

      - HCV-150
      - HCV-151

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

2. (continued)

d. Place **BOTH** PORV handswitches in "CLOSE":

- PCV-102-1
- PCV-102-2

e. (**CRS**) Obtain the AOP-6 Keys and a transceiver.

f. Direct all operators in the Control Room to obtain AOP-06 Keys and a transceiver.

g. Ensure extra transceivers are taken to AI-185, "ALTERNATE SHUTDOWN PANEL" (West Upper Electrical Penetration Room).

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS	CONTINGENCY ACTIONS
1. Direct an operator to locate and report the status of the fire.	
2. **IF** an acrid odor is detected, **THEN** IMPLEMENT Attachment O, Acrid Odor.	
3. **IF** a Control Room evacuation is required, **THEN** GO TO AOP-06 Section II, Control Room Evacuation.	
4. **IF** the Cable Spread Room is affected, **THEN** GO TO AOP-06 Section II, Control Room Evacuation.	
5. **IF** a Plant fire exists **AND** fire brigade activation is warranted, **THEN** sound the Fire Alarm for 10-15 seconds.	

QUESTION NUMBER: 076

A plant transient has led to a high pressure reactor trip. Following the trip, a pressurizer safety valve opened and did not reclose. The RCS pressure then lowered and PPLS actuation started ECCS pumps. The operators tripped all RCPs.

It is 10 minutes after the trip and current plant conditions are:

- RCS Pressure is 900 psia and rising
- Hot leg and CET temperatures indicate 530ºF
- Pressurizer level is 85% and rising
- 2 HPSI, 2 LPSI and all charging pumps are running
- EOP-00, "Standard Post Trip Actions" have been completed

What action should be directed in response to the rising pressurizer level?

A.✓ Enter EOP-03, "Loss of Coolant Accident," and allow HPSI and charging pumps to continue injecting at full capacity.

B. Enter EOP-03, "Loss of Coolant Accident," and reduce HPSI and charging pump flow per floating step A, "HPSI Stop and Throttle Criteria."

C. Enter EOP-20, "Functional Recovery Procedure," and allow HPSI and charging pumps to continue injecting at full capacity.

D. Enter EOP-20, "Functional Recovery Procedure," and reduce HPSI and charging pump flow per floating step A, "HPSI Stop and Throttle Criteria."

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 76 Rev 0

KA #: 000008 AA2.23 Tier 1 Group 1: Pressurizer Vapor Space Accident

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Criteria for throttling high-pressure injection after a small LOCA
Importance 3.6 / 4.3

CFR Number: 55.43(b)(5)

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:

EXPLAIN how the "Stop and Throttle" criteria is used to prevent reducing HPSI flow when full HPSI flow is required.

Question Pedigree

Modified question. Source question was used on 2005 NRC exam.

K/A Fit:

Question addresses reducing HPSI flow after a small LOCA

Choice A:

Correct answer: HPSI stop and throttle criteria not net due to inadequate subcooling. EOP-03 entry conditions met.

Choice B:

Distractor: Plausible because pressurizer level is high and rising. Incorrect because criteria are not met.

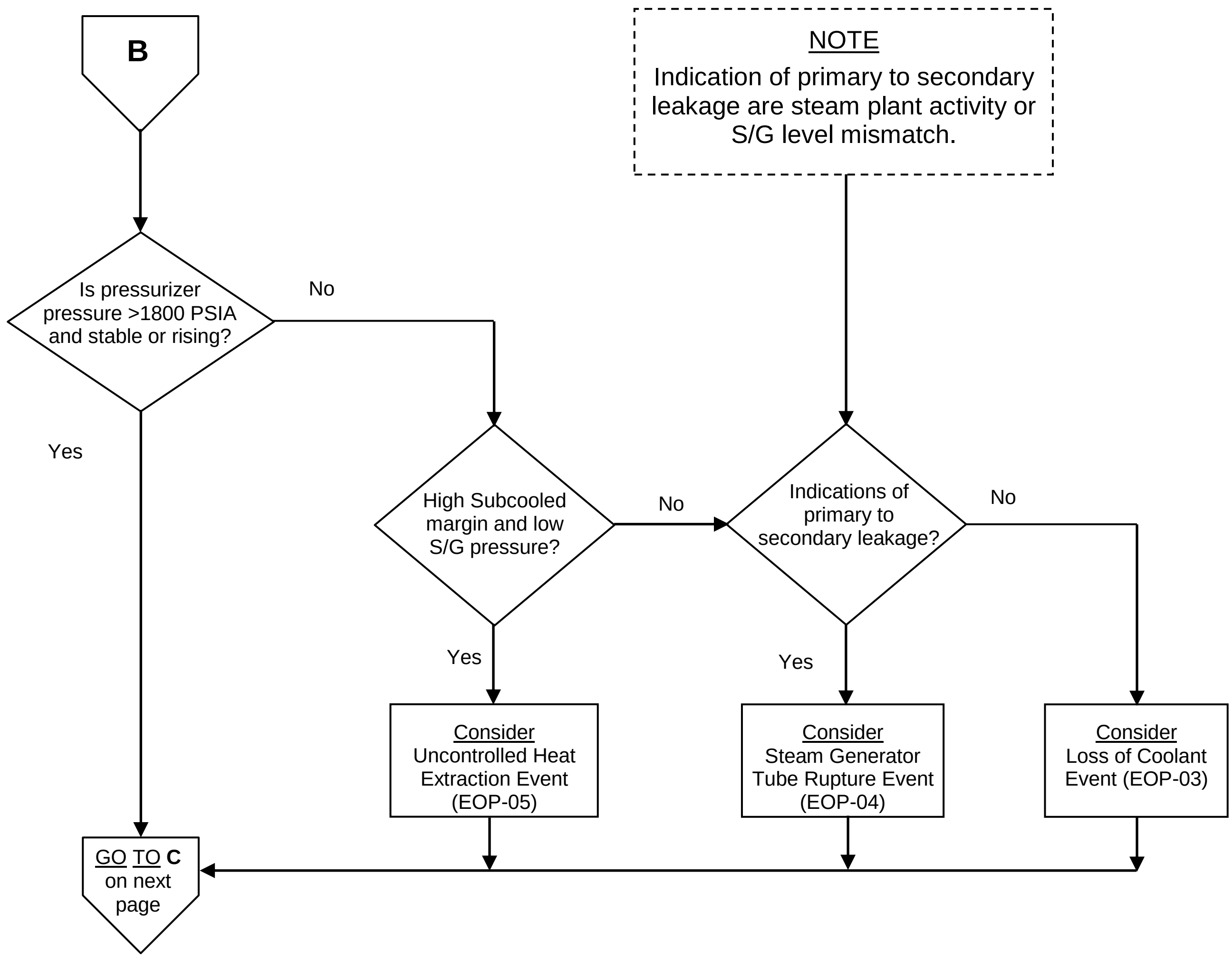
Choice C:

Distractor: Plausible because action is correct. Incorrect because EOP-20 entry conditions are not met.

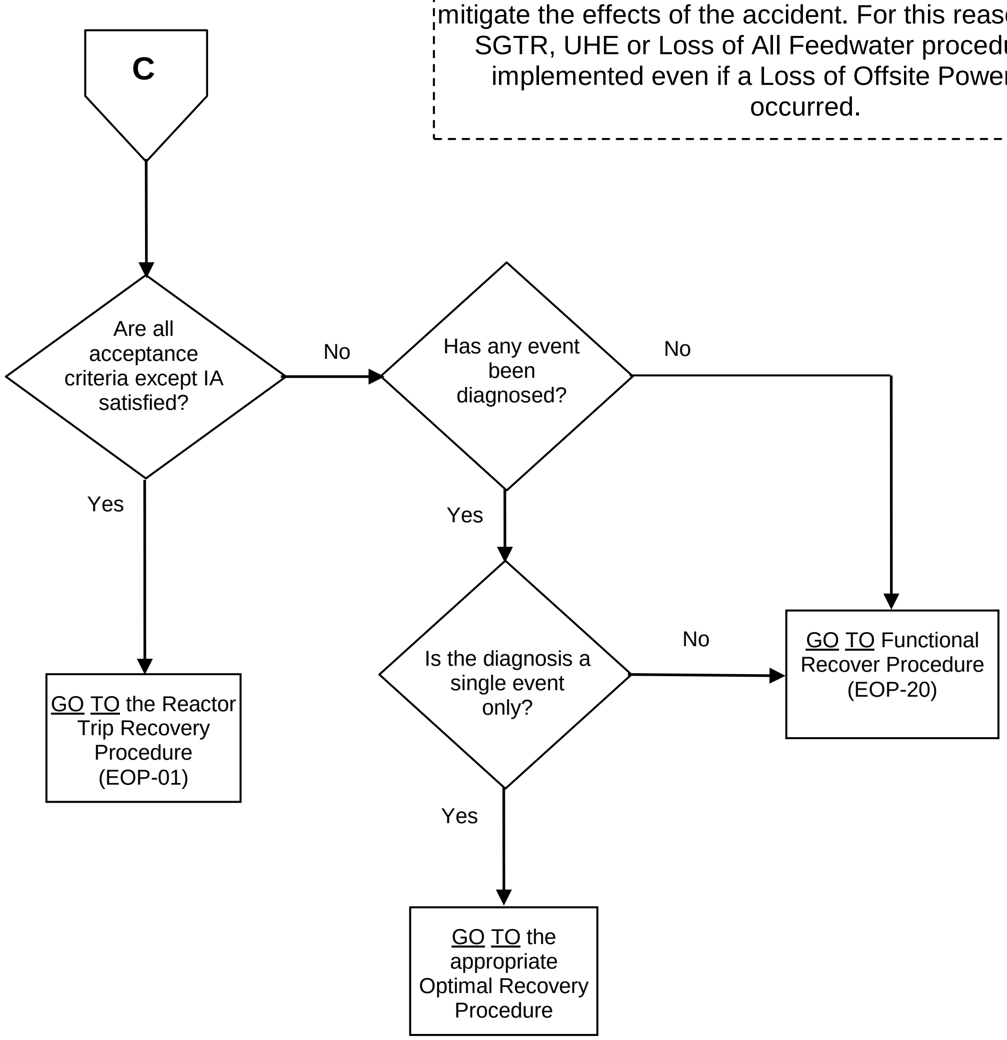
Choice D:

Distractor: Plausible because pressurizer level is high and rising. Incorrect because criteria are not met and EOP-20 entry conditions are not met.

KA#:	000008 AA2.23	Bank Ref #:	07-15-23 029
LP# / Objective:	0715-23 02.06	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	LP 07-15-23	Handout:	NONE


B
NOTE
Indication of primary to secondary leakage are steam plant activity or S/G level mismatch.
Is pressurizer pressure >1800 PSIA and stable or rising?
No
Yes
High Subcooled margin and low S/G pressure?
No
Indications of primary to secondary leakage?
No
Yes
Yes
Consider Uncontrolled Heat Extraction Event (EOP-05)
Consider Steam Generator Tube Rupture Event (EOP-04)
Consider Loss of Coolant Event (EOP-03)
GO TO C on next page

Certain events (i.e., LOCA, SGTR, UHE and Loss of All Feedwater) do not require offsite power in order to adequately, mitigate the effects of the accident. For this reason, the LOCA, SGTR, UHE or Loss of All Feedwater procedure may be implemented even if a Loss of Offsite Power has also occurred.



2.0 FLOATING STEPS

A. HPSI STOP AND THROTTLE CRITERIA

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. If emergency boration is required then at least one charging pump must remain running.

2. As natural circulation develops, the expected rise in T_H will reduce subcooling. This may jeopardize HPSI Stop and Throttle Criteria.

3. Reducing SI flow should be approached cautiously.

4. The purpose of HPSI stop and throttle is to prevent an over pressurization of the RCS and a solid PZR, however, maintaining RCS inventory is more important than pressure control.

****

1. Verify **ALL** of the following stop and throttle criteria are satisfied:

   - RCS subcooling is greater than or equal to 20°F
   - PZR level is greater than or equal to 10% and not lowering
   - At least one S/G is available for RCS heat removal
   - RVLMS indicates level is at or above the top of the Hot Leg (43%, ERF "I" display)

QUESTIONS REPORT
for ILO EXAM BANK

07-15-23 029

A plant transient has led to a high pressure reactor trip. Following the trip, a pressurizer safety valve opened and did not reclose. The RCS pressure then lowered and PPLS actuation started all of the ECCS pumps. The operators tripped all RCPs.

It is 20 minutes after the trip and current plant conditions are:

- RCS Pressure is steady at 900 psia
- Hot leg and CET temperatures indicate 530ºF
- Pressurizer level is 85% and rising
- All HPSI, LPSI and charging pumps are running.

What action should be taken in response to the rising pressurizer level?

A. ✓ Continue to provide full injection flow. Allow all ECCS pumps to continue injecting.

B. Trip one HPSI pump and all LPSI pumps. Continue to provide full charging flow.

C. Trip all HPSI pumps. Trip charging pumps as required to control pressurizer level.

D. Trip HPSI pumps as needed to control pressurizer level. Continue to provide full charging flow.

QUESTION NUMBER: 077

The plant was operating at full power with the following conditions:

- Containment pressure is 1.2 psig and slowly rising
- Containment temperature is 105°F and slowly rising
- Containment dew point is 125°F and slowly rising
- Pressurizer pressure and level are steady
- Charging flow is 40 gpm and steady
- Letdown flow is 32 gpm and lowering

What action should be taken as a result of these conditions?

A. Enter OI-VA-1, "Containment Heating, Cooling and Ventilation Systems Normal Operation," and place an additional containment cooling unit in service.

B. Enter OI-VA-1, "Containment Heating, Cooling and Ventilation Systems Normal Operation," and initiate containment pressure relief.

C.✓ Enter AOP-22, "Reactor Coolant Leak," isolate letdown and determine the leak rate.

D. Trip the reactor and enter EOP-00, "Standard Post Trip Actions. Then enter EOP-03, "Loss of Coolant Accident."

Question # 77 Rev 0

KA #: 000009 EA2.11 Tier 1 Group 1: Small Break LOCA
Ability to determine or interpret the following as they apply to a small break LOCA: Containment temperature, pressure, and humidity
Importance 3.8 / 4.1

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the Reactor Coolant Leak Procedure to mitigate the consequences of a leak in the Reactor Coolant System.

Question Pedigree
New question.

K/A Fit:
Question addresses using containment pressure, temperature and dew point to mitigate a small break in containment.

Choice A:
Distractor: Plausible because containment pressure and temperature are rising. Incorrect because AOP-22 entry conditions have been reached.

Choice B:
Distractor: Plausible because containment pressure is rising. Incorrect because AOP-22 entry conditions have been reached.

Choice C:
Correct answer. Indications of a 4 gpm leak in containment. AOP-22 entry conditions have been reached. AOP-22 will direct letdown isolation and leakrate determination.

Choice D:

Distractor: Plausible because a reactor trip is directed for RCS leakage greater than charging pump capacity. Incorrect because leakage is much less than 40 gpm.

KA#:	000009 EA2.11	Bank Ref #:	N/A
LP# / Objective:	0717-22 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	AOP-22	Handout:	NONE

Section I - Leak Rate Determination and Leak Isolation

1.0 PURPOSE

This procedure provides direction in the event of any size RCS leak from any initial RCS condition.

2.0 ENTRY CONDITIONS

RCS Leak Rate has increased as indicated by any one or more of the following conditions:

A. Lowering VCT level.

B. Unexplained mismatch between Charging and Letdown flow.

C. Abnormal change in PZR level.

D. Lowering RCS pressure.

E. Total RCS leak rate exceeds the limits of Technical Specification 2.1.4.

F. High temperature alarms on PORVs or PZR Code Safeties.

G. High Quench Tank level, pressure or temperature.

H. High RCDT level or pressure.

I. High Containment dew point, radiation, pressure or Sump level.

J. Rising S/G Blowdown or Condenser Off-Gas radiation.

K. High Auxiliary Building airborne activity.

L. Primary-to-Secondary leakage verified by Chemistry sampling.

Section I - Leak Rate Determination and Leak Isolation

INSTRUCTIONS | CONTINGENCY ACTIONS

5. (continued)

d. Verify PZR level is restored.

d.1 **IF** the Reactor is critical
AND PZR level continues to drop,
THEN initiate a Reactor Trip by performing the following steps:

1) Trip the Reactor.

2) GO TO EOP-00, Standard Post Trip Actions.

6. Determine the RCS leakage rate.

7. **IF** RCS leakage rate is greater than 40 gpm
AND the Reactor is critical,
THEN initiate a Reactor Trip by performing the following steps:

a. Trip the Reactor.

b. GO TO EOP-00, Standard Post Trip Actions.

Section I - Leak Rate Determination and Leak Isolation

INSTRUCTIONS	CONTINGENCY ACTIONS
8. Direct the Shift Chemist to verify primary to secondary leak rate less than 1 gpd PER CH-AD-0007, Primary to Secondary Leak Rate Determination.	8.1 **IF** primary to secondary leak rate is greater than 1 gpd, **THEN** IMPLEMENT Attachment B, Primary to Secondary Leak Rate Actions.

NOTES

1. If all Charging Pump auto-start signals are defeated and the Reactor is critical, the Plant must be placed in Hot Shutdown within six hours.

2. If Pressurizer Heater control is lost and the Reactor is critical, the Plant must be placed in Hot Shutdown within 12 hours.

9. Determine the source of leakage by performing the following steps:

   a. Isolate Letdown by closing **BOTH** Letdown Isolation Valves:

      - TCV-202
      - HCV-204

   b. Isolate CVCS by performing the following steps:

      1) Place the Charging Pump Control Switches in "PULL-TO-LOCK".

(continue)

QUESTION NUMBER: 078

Given the following plant conditions:

- A RCS cooldown is in progress
- T-cold is 310°F
- The cooldown rate is 75°F/hr
- A transient caused RCS pressure to increase to 1900 psia
- RCS pressure has been reduced to 600 psia

What actions are required as a result of these conditions?

A. No actions are required.

B.✓ Enter Technical Specification LCO 2.1.2, "Heatup and Cooldown Rate," and direct engineering to perform a fracture toughness evaluation.

C. Enter TDB-IX, "RCS Pressure and Temperature Limits Report" and direct engineering to perform an Appendix "E" evaluation.

D. Enter EOP/AOP Attachment 27, "P-T Limit Restoration," and maintain current temperature for 3 hours.

Question # 78 Rev 0

KA #: 000027 2.1.25 Tier 1 Group 1: Pressurizer Pressure Control System Malfunction
Ability to interpret reference materials such as graphs, curves, etc.
Importance 3.9 / 4.2

CFR Number: 55.43(b)(2)
Facility operating limitations in the technical specifications and their bases.

Fort Calhoun Objective:
EXPLAIN the basis for the RCS heatup and cooldown curves and STATE the limits.

Question Pedigree
New question.

K/A Fit:
Question addresses using graphs and curves to determine action required by technical specifications following a malfunction of the LTOP system.

Choice A:
Distractor: Plausible because all limits are currently being met.

Choice B:
Correct answer. The PTLR limit was exceeded. Tech Spec 2.1.2 requires a fracture toughness evaluation.

Choice C:
Distractor: Plausible because PTLR limits were exceeded. Incorrect because the Appendix E limits were not exceeded.

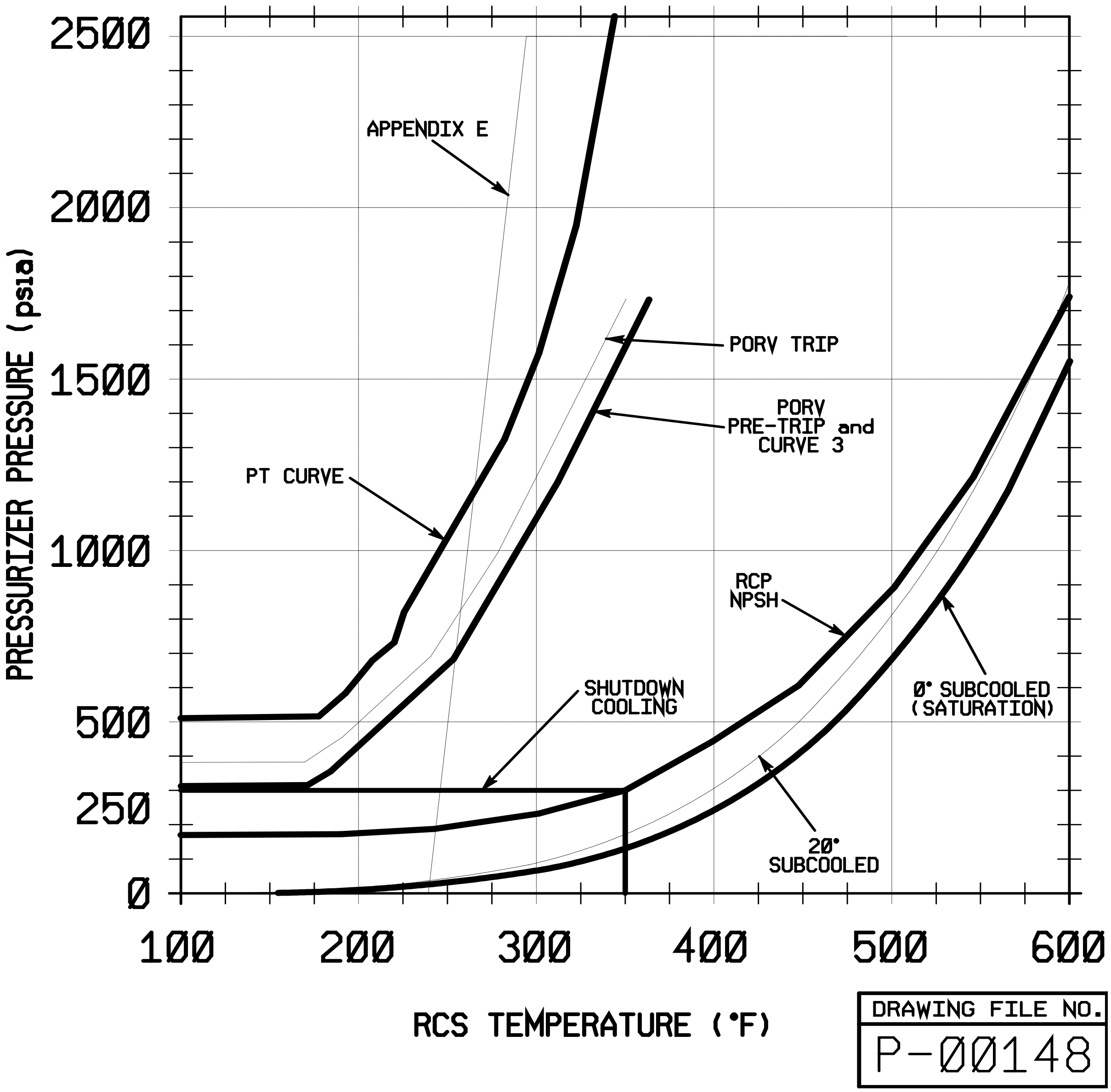
Choice D:
Distractor: Plausible because this is an action that would be taken if PTLR limits were exceeded during a steam line break. Incorrect because EOP entry conditions have not been met.

HANDOUTS: EOP/AOP Attachment 2 curve and TDB-IX Figure 5-1

KA#:	000027 2.1.25	Bank Ref #:	N/A
LP# / Objective:	0711-20 02.02	Exam Level:	SRO-2
Cognitive Level:	HIGH	Source:	NEW
Reference:	TS 2.1.2	Handout:	EOP/AOP ATT 2

Attachment 2

RCS Pressure-Temperature Limits for 40 EFPY

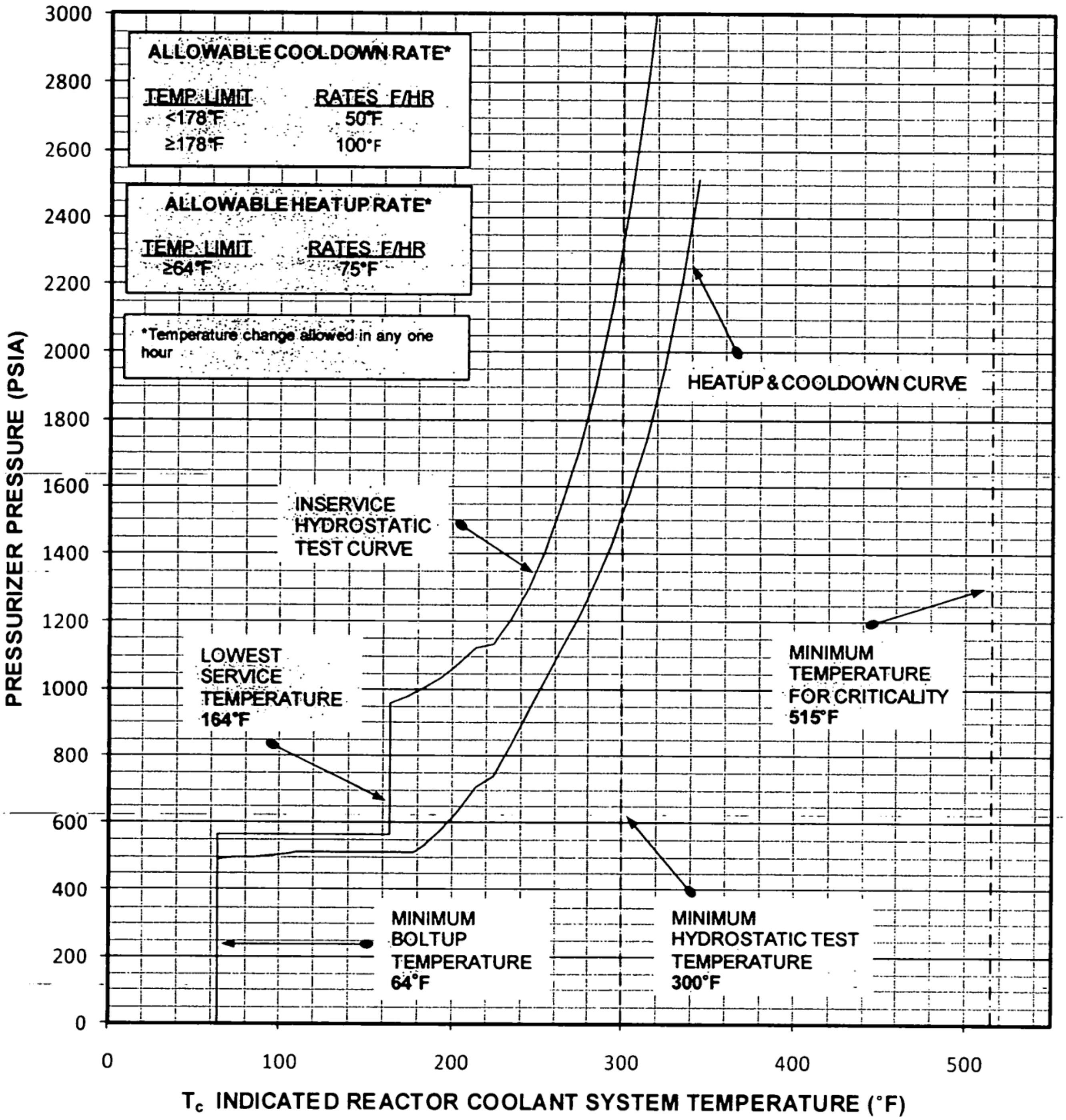


NOTES

1. During forced circulation, use T_H for the 20° F subcooled curve and saturation curve. During natural circulation, use CETs for the 20° F subcooled curve and the saturation curve. Use T_C for all other curves.
2. Curve 3 is the Maximum Pressure for First Start RCP curve.

   Reference: EA06-036

Figure 5-1 – Fort Calhoun Station Composite P-T Limits 40 EFPY



2.0 **LIMITING CONDITIONS FOR OPERATION**
2.1 Reactor Coolant System (Continued)
2.1.2 Heatup and Cooldown Rate

Applicability

Applies to the temperature change rates and pressure of the Reactor Coolant System (RCS).

Objective

To specify limiting conditions of the reactor coolant system heatup and cooldown rates.

Specification

The combination of RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR and as designated below:

a. Allowable combinations of pressure and temperature (T_c) for a specific heatup rate shall be below and to the right of the applicable limit lines as shown on the pressure and temperature (P-T) limit Figure(s) in the PTLR.

b. Allowable combinations of pressure and temperature (T_c) for a specific cooldown rate shall be below and to the right of the applicable limit lines as shown on the P-T limit Figure(s) in the PTLR.

c. The heatup rate of the pressurizer shall not exceed 100°F in any one hour period.

d. The cooldown rate of the pressurizer shall not exceed 200°F in any one hour period.

Required Actions

(1) When any of the above limits are exceeded, the following corrective actions shall be taken:

1. Immediately initiate action to restore the temperature or pressure to within the limit.

2. Perform an analysis to determine the effects of the out of limit condition on the fracture toughness properties of the reactor coolant system.

3. Determine that the reactor coolant system remains acceptable for continued operation or be in cold shutdown within 36 hours.

(2) Before the radiation exposure of the reactor vessel exceeds the exposure for which they apply, the P-T limit Figure(s) shown in the PTLR shall be updated in accordance with the following criteria and procedures:

QUESTION NUMBER: 079

Given the following plant conditions:

- The plant is operating at full power
- The “STEAM GEN FEED PUMP B OVERLOAD/TRIP” annunciator on CB-10,11/A12 is in alarm
- The control switch status for Feedwater Pump FW-4A is red light on / green flag
- The control switch status for Feedwater Pump FW-4B is green light on / red flag
- The control switch status for Feedwater Pump FW-4C is red light on / red flag
- Level in both S/G's is 61% narrow range
- DCS display indicates that both LCV-1101 and LCV-1102 are in three element mode

What action should be taken as a result of these conditions?

A. Trip the reactor, enter EOP-00, "Standard Post Trip Actions," and ensure an auxiliary FW pump is running.

B. Trip the reactor, complete EOP-00, "Standard Post Trip Actions," and enter EOP-06, "Loss of All Feedwater."

C. Enter ARP-DCS-FW and place control for LCV-1101 and LCV-1102 in single element mode

D.✓ Enter ARP-CB-10,11/A12 and ensure the 43-SIAS/FW4 switch is selected to FW-4C

Question # 79 Rev 0

KA #: 000054 AA2.02 Tier 1 Group 1: Loss of Main Feedwater
Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Differentiation between loss of all MFW and trip of one MFW pump
Importance 4.1 / 4.4

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
INTERPRET Feedwater System alarms and actions in the Control Room using the Alarm Response Procedures as a guide.

Question Pedigree
New question.

K/A Fit:
Question addresses differentiating between a pump trip and a loss of all feedwater and procedural action to be taken.

Choice A:
Distractor: Plausible if Applicant believes all Feedwater has been lost. Incorrect because only one pump tripped.

Choice B:
Distractor: Plausible if Applicant believes all main and auxiliary feedwater has been lost. Incorrect because only one pump tripped.

Choice C:
Distractor: Plausible if Applicant believes feedwater transient is due to DCS control malfunction. Incorrect because a FW pump tripped.

Choice D:
Correct answer. FW-4B tripped and FW-4C auto started. ARP directs that 43-SIAS/FW4 switch be selected to FW-4C.

KA#:	000054 AA2.02	Bank Ref #:	N/A
LP# / Objective:	0711-11 02.07	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	ARP-CB-10,11/A12	Handout:	NONE

Panel: **CB-10**	Annunciator: **A12**	Window: **A-6L**

STEAM GENERATOR FEEDWATER PUMP FW-4B
OVERLOAD OR TRIP

Page 1 of 2

SAFETY RELATED

STEAM GEN FEED PUMP B OVERLOAD/TRIP

Tech Spec References: None

Initiating Device 74 / FW-4B   Setpoint Breaker OPEN   Power DC BUS 2

OPERATOR ACTIONS

1. Dispatch an Operator to check the status of FW-4B, Feedwater Pump.

    1.1 IF FW-4B is operating, THEN check 83/FW-4B, Overload Relay at 1A2 (1A2-8).

2. IF FW-4B is tripped, THEN ensure that standby Feedwater Pump FW-4A or C has automatically started.

    2.1 IF two pumps are required, THEN ensure Feedwater Pumps FW-4A and C are running.

    2.2 Ensure 43-SIAS/FW4 Post SIAS Running Feedwater Pump is selected to FW-4C.

    2.3 IF no Feedwater Pump can be started, THEN trip the Reactor and GO TO EOP-00.

(continue)

PROBABLE CAUSES

- Steam Generator Feedwater Pump low oil pressure
- Feedwater Pump motor overload or trips on phase or differential overcurrent
- Loadshed on 4160 Volt Bus 1A2
- Feedwater Pump stopped from the local 69 permissive control switch

REFERENCES

136B3083 Sh 23 06305   0223R0454 Sh 9 09928

Panel: **CB-10**	Annunciator: **A12**	Window: **A-6L**

STEAM GENERATOR FEEDWATER PUMP FW-4B
OVERLOAD OR TRIP

SAFETY RELATED

OPERATOR ACTIONS (continued)

2.4 Check the following for the cause of the FW-4B trip:

- 49/FW-4B, Overcurrent Relay tripped at 1A2 (1A2-8)
- Motor stopped from local stop pushbutton
- Motor stopped from the 69 permissive switch (1A2-8)
- Low voltage on 4160V Bus 1A2
- Low oil pressure on Feedwater Pump
- 86/FW-4B, Feedwater Pump Motor Differential Overcurrent (AI-12)

2.5 Notify Electrical Maintenance of Feedwater Pump trip.

2.6 Place the running (standby) pump in AFTER-START.

NOTE: Prevent a pump in PULL-OUT from auto starting by first matching flags for the pumps not in PULL-OUT.

3. IF it is desired to align another pump to Standby, THEN perform the following:

3.1 Ensure the Control Switch for the running pump is in AFTER-START (Red Flag).

3.2 Place the Control Switch for the tripped pump in AFTER-STOP (Green Flag).

3.3 Place the Control Switch for the pump in PULL-OUT to AFTER-STOP (Green Flag).

QUESTION NUMBER: 080

The reactor was operating at full power, when the following annunciators came into alarm on CB-20, Panel A15:

- DC BUS #1 GROUND
- DC BUS #1 LOW VOLTAGE
- INVERTER #1 TROUBLE
- INVERTER A TROUBLE
- INVERTER C TROUBLE

A few seconds later, the reactor tripped. During the performance of EOP-00, Standard Post Trip Actions, the Balance of Plant Operator reported that DC Bus #1 is not energized.

What action should be taken following completion of EOP-00?

A. Enter AOP-16, "Loss of Instrument Bus Power," and transfer Instrument bus power to swing inverter EE-8T.

B. Enter AOP-16, "Loss of Instrument Bus Power," and transfer switchgear DC control power to DC bus #2.

C. Enter EOP-20, "Functional Recovery Procedure," and transfer Instrument bus power to swing inverter EE-8T.

D.✓ Enter EOP-20, "Functional Recovery Procedure," and transfer switchgear DC control power to DC bus #2.

Question # 80 Rev 0

KA #: 000058 2.1.07 Tier 1 Group 1: Loss of DC Power
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
Importance 4.4 / 4.7

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Given the Resource Assessment Trees, basically DESCRIBE the Method, Path and Acceptance Criteria for each success path.

Question Pedigree
New question.

K/A Fit:
Question addresses diagnosing symptoms of a loss of a DC bus and taking proper procedural actions.

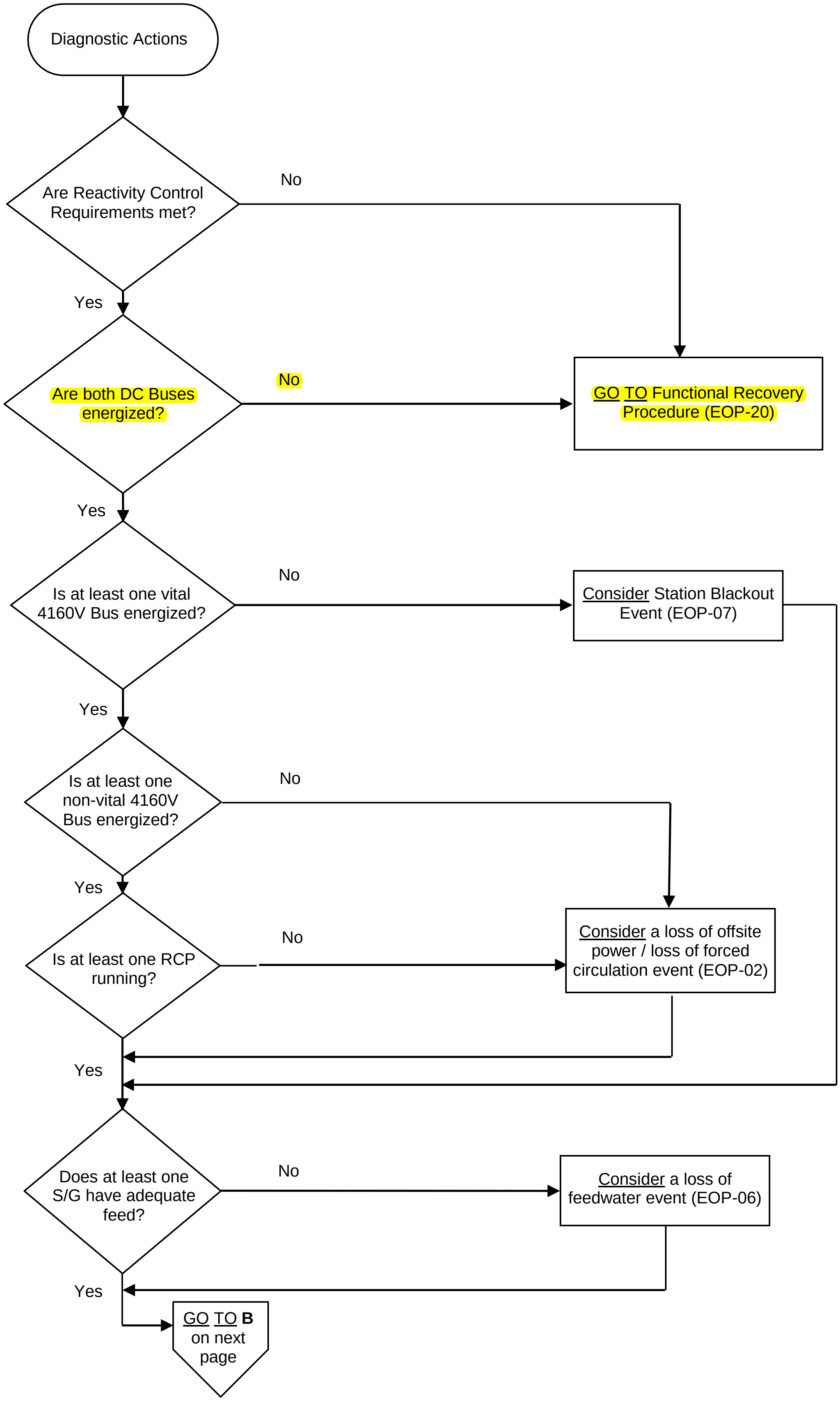
Choice A:
Distractor: Plausible because instrument inverter alarms are in. Incorrect because AOP-16 is not entered from EOP-00.

Choice B:
Distractor: Plausible because DC bus number 1 has been lost and AOP-16 does contain directions for transferring DC control power. Incorrect because AOP-16 is not entered from EOP-00.

Choice C:
Plausible because of instrument inverter trouble alarms and EOP-20 is the correct procedure. Incorrect because a DC bus has been lost.

Choice D:
Correct answer: EOP-00 directs transfer to EOP-20 and EOP-20 directs transferring DC control power.

KA#:	000058 2.1.07	Bank Ref #:	N/A
LP# / Objective:	0718-18 01.05	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-00, EOP-20	Handout:	NONE


Diagnostic Actions
Are Reactivity Control Requirements met?
No
GO TO Functional Recovery Procedure (EOP-20)
Yes
Are both DC Buses energized?
No
Yes
Is at least one vital 4160V Bus energized?
No
Consider Station Blackout Event (EOP-07)
Yes
Is at least one non-vital 4160V Bus energized?
No
Yes
Is at least one RCP running?
No
Consider a loss of offsite power / loss of forced circulation event (EOP-02)
Yes
Does at least one S/G have adequate feed?
No
Consider a loss of feedwater event (EOP-06)
Yes
GO TO **B** on next page

11.0 MAINTENANCE OF VITAL AUXILIARIES - DC

INSTRUCTIONS	CONTINGENCY ACTIONS
✘2. **IF** both DC Buses are energized, **THEN** GO TO Step 81.	2.1 **IF** either DC Bus is deenergized, **THEN** determine the appropriate recovery method by performing the following: a. **IF** DC Bus 1 is deenergized, **THEN** GO TO Step 3. b. **IF** DC Bus 2 is deenergized, **THEN** GO TO Step 42.

****

CAUTION

HCV-438A and C may go closed after control power is transferred on AI-41A.

****

✘3. Transfer DC control power to emergency by performing the following:

a. Press the "EMERG. SOURCE 125 VDC" pushbutton (AI-41A).

(continue)

11.0 MAINTENANCE OF VITAL AUXILIARIES - DC

MVA-DC

INSTRUCTIONS

CONTINGENCY ACTIONS

✘3. (continued)

b. Press PB-2/1B3A-4A-MTS, “MANUAL TRANSFER PUSHBUTTON 1B3A-4A-MTS EMERG. SOURCE" pushbutton (East Switchgear Room, 1B3A).

c. Press PB-2/1B3C-4C-MTS, “MANUAL TRANSFER PUSHBUTTON 1B3C-4C-MTS EMERG. SOURCE" pushbutton (East Switchgear Room, 1B3C).

d. Press PB-2/1A1-1A3-MTS, “MANUAL TRANSFER PUSHBUTTON 1A1-1A3-MTS EMERGENCY SOURCE” pushbutton ("1A1-1A3 AUX POWER COMPARTMENT").

(continue)

11.0 MAINTENANCE OF VITAL AUXILIARIES - DC

INSTRUCTIONS | CONTINGENCY ACTIONS

✘3. (continued)

e. Press ATD-D1, "DIESEL D1 125 VDC MANUAL TRANSFER SWITCH" "EMERGENCY" pushbutton, (D-1 Room, North Wall).

Time: ____________

NOTE

PCV-1849, Containment Instrument Air Inboard Isolation Valve may need to be held open while repressurizing the Containment Air header.

✘4. Restore Instrument Air to Containment by performing the following:

a. Open PCV-1849B, Containment Instrument Air Isolation Valve (Outboard).

b. Open PCV-1849A, Containment Instrument Air Isolation Valve (Inboard).

QUESTION NUMBER: 081

Given the following plant conditions:

- The plant was operating at full power
- The "PLANT AIR PRESS LO" annunciator alarmed 5 minutes ago
- The "INSTRUMENT AIR PRESS LO" annunciator alarmed 5 minutes ago
- The "FW REG SYS FCV-1101 TROUBLE" annunciator just alarmed
- The "FW REG SYS FCV-1102 TROUBLE" annunciator just alarmed
- The Balance of Plant Operator reports instrument air pressure is 48 psig and lowering

What SEQUENCE of procedures and actions should be used to mitigate this event.?

A✓ Enter AOP-17, "Loss of Instrument Air," trip the reactor, stop all main FW pumps then implement EOP-00, "Standard Post Trip Actions."

B. Trip the Reactor, Enter EOP-00, "Standard Post Trip Actions," implement AOP-17, "Loss of Instrument Air," then stop all main FW pumps.

C. Enter AOP-17, "Loss of Instrument Air," trip the reactor, reset and close FCV-1101 and FCV-1102 then implement EOP-00, "Standard Post Trip Actions."

D. Trip the Reactor, Enter EOP-00, "Standard Post Trip Actions," implement AOP-17, "Loss of Instrument Air," then reset and close FCV-1101 and FCV-1102.

Question # 81 Rev 0

KA #: 000065 2.1.32 Tier 1 Group 1: Loss of Instrument Air
Ability to explain and apply system limits and precautions.
Importance 3.8 / 4.0

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the Loss of Instrument Air Procedure to mitigate the consequences of a partial or complete loss of instrument air.

Question Pedigree
New question.

K/A Fit:
Question addresses FW control limitations during a loss of instrument air.

Choice A:
Correct answer: AOP-17 is initially entered. When instrument air pressure drops below 50 psig, AOP-17 directs the operators to trip the reactor, stop all main FW pumps and then implement EOP-00, "Standard Post Trip Actions.

Choice B:
Distractor: Plausible because the reactor is tripped. Incorrect because AOP-17 should have been entered on the initial alarms and AOP-17 directs that the reactor be tripped and FW pumps tripped before performing EOP-00.

Choice C:
Distractor: Plausible because AOP-17 is initially entered and directs that the reactor be tripped. Incorrect because AOP-17 directs that the FW pumps be tripped because FCV-1101 and FCV-1102 can not be closed due to the loss of instrument air pressure.

Choice D:
Distractor: Plausible because the reactor is tripped. Incorrect because AOP-17 directs that the FW pumps be tripped because FCV-1101 and FCV-1102 can not be closed due to the loss of instrument air pressure

KA#:	000065 2.1.32	Bank Ref #:	N/A
LP# / Objective:	0717-17 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	TBD-AOP-17	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

Extended operation with Compressed Air pressure less than 80 psig will result in depletion of Air Accumulator reserves.

****

13. **IF** Instrument Air pressure is greater than or equal to 50 psig,
THEN restore Instrument Air by performing the following steps:

    a. Determine the source of air leakage.

    b. Evaluate the need to shutdown the Reactor PER **ONE** of the following procedures:

    - OP-4, Load Change and Normal Power Operations
    - AOP-05, Emergency Shutdown

    c. Direct Maintenance to repair the source of air leakage.

    d. Ensure Service Air and Air Dryers have been returned to normal.

    e. **IF** air pressure is greater than or equal to 98 psig,
    THEN GO TO Section 5.0, Exit Conditions.

13.1 **IF** Instrument Air pressure is less than 50 psig,
THEN initiate a Reactor Shutdown by performing the following steps:

    a. Trip the Reactor.

    b. Stop all Main Feed Pumps to prevent an RCS cooldown.

    c. IMPLEMENT EOP-00, Standard Post Trip Actions.

    d. IMPLEMENT the Emergency Plan.

    e. Close **ONE** of the following valves to isolate LCV-1190 and prevent draining of Condensate Storage Tank to Condenser Hotwell (Turbine Mezzanine; West Side):

    - FW-269, “CONDENSATE MAKEUP VALVE LCV-1190 INLET VALVE”
    - FW-270, “CONDENSATE MAKEUP VALVE LCV-1190 OUTLET VALVE”

INSTRUCTIONS | CONTINGENCY ACTIONS

With Instrument Air pressure less than 50 psig, the pneumatic controllers will respond sporadically, if at all. Valves which are necessary for proper operation and control of the plant (i.e., Main Feedwater Reg and Bypass Valves) will not be operable at <50 psig. Therefore, the most conservative course of action would be to trip the plant to prevent more severe complications [(R8)].

The caution warns the operators that indefinite operation with abnormally low Instrument Air pressures is not possible. The contingency allows for short-term operation with low air pressure while the system is being restored to normal.

EPIP-OSC-1, Emergency Classification, provides instructions for emergency classification upon loss or degradation of the Instrument Air System.

Steps 13 through 25 provide instructions for actions to be taken in the event Instrument Air pressure is lost. This meets an NRC commitment [(C1)].

If instrument air can not be maintained above 50 psig, then either FW-269 or FW-270 is closed to isolate LCV-1190. LCV-1190 fails open on a loss of instrument air pressure. Backup N_2 will keep LCV-1190 closed for a limited time. Manual closure of FW-269 or FW-270 will isolate this line and prevent draining of the condensate storage tank to the condenser hotwell, thereby preserving a suction source for engine-driven AFW pump FW-54.

The Main Feedwater Pumps will cooldown the RCS due to the large amounts of "cold" water added to the Steam Generators. Securing the Main Feedwater Pumps will prevent and/or greatly decrease the RCS cooldown rate by slowing down the rate of cold water addition.

NOTES

1. YCV-1045A and YCV-1045B may open as their air accumulators bleed down, causing FW-10, Steam AFW Pump, to start.

2. Open and close operation of HCV-1107B and HCV-1108B is possible for a minimum of three cycles.

14. Feed S/Gs with AFW to the AFW Nozzles to maintain S/G levels 35-85% NR (73-94% WR) by performing the following steps:

    a. Ensure the "43/FW" Switch is in "OFF".

    b. Start FW-6, Electric AFW Pump.

QUESTION NUMBER: 082

The reactor tripped on low RCS flow following a loss of power to 4160V bus 1A3. During the performance of EOP-00, Standard Post Trip Actions, the At The Controls Operator reported that two trippable CEAs failed to insert.

Assuming bus 1A3 remains deenergized, which one of the following actions should be taken?

A.✓ Stay in EOP-00, "Standard Post Trip Actions," and dispatch the Aux Building Operator to manually open HCV-268, "Boric Acid Pump Discharge Header Isolation Valve."

B. Stay in EOP-00, "Standard Post Trip Actions," and dispatch the Aux Building Operator to manually open LCV-218-3, "Charging Pump Suction SIRWT Isolation Valve."

C. Enter AOP-03, "Emergency Boration," and dispatch the Aux Building Operator to manually open HCV-265, "CH-11A Gravity Feed Valve."

D. Enter AOP-03, "Emergency Boration," and dispatch the Aux Building Operator to manually close LCV-218-2, Charging Pump Suction VCT Isolation Valve."

Question # 82 Rev 0

KA #: 000024 2.2.37 Tier 1 Group 2: Emergency Boration
Ability to determine operability and/or availability of safety related equipment.
Importance 3.6 / 4.6

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the Emergency Boration AOP to mitigate the consequences of an uncontrollable or unexplained positive reactivity addition.

Question Pedigree
Bank question used on 2004 NRC exam. Reworded to SRO question. Not counted as a modified question.

K/A Fit:
Question addresses operability of equipment needed for emergency boration with a loss of bus 1A3.

Choice A:
Correct answer. HCV-268 is powered from 480 bus 1B3C which is powered from 4160 bus 1A3. When HCV-268 is manually opened, it will allow flow from the powered BA pump, CH-4B.

Choice B:
Distractor: Plausible because LCV-218-3 is used for emergency boration in EOP-20. Incorrect because it is not used in EOP-00.

Choice C:
Distractor: Plausible because HCV-265 is opened as part of emergency boration. Incorrect because AOP-03 is not entered and VCT pressure will prevent gravity feed unless LCV-218-2 is closed.

Choice D:
Distractor: Plausible because closing LCV-218-2 will allow emergency boration through the powered gravity feed valve. Incorrect because AOP-03 is not entered.

KA#:	000024 2.2.37	Bank Ref #:	07-17-03 003
LP# / Objective:	0717-03 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	BANK
Reference:	STM 12	Handout:	NONE

ning, and the concentrated boric acid tanks were the highest pressure suction source available to the charging pumps. In that case, the pumps would further borate the reactor.

•••Fail Position on Loss of Air or Power

2.586 On a loss of power, HCV-258 and -265 will "fail as is", and the respective indicating lights will go dark.

Boric Acid Pump to Charging Pump Suction Valve HCV-268

2.587 The following is a detailed description of HCV-268.

••Function

2.588 The boric acid pump to charging pump suction valve, HCV-268, is a normally closed, motor-operated valve that connects the combined discharges of the boric acid pumps directly to the suctions of the charging pumps via the VCT outlet line

••Design/Specification

2.589 This piping arrangement bypasses the boric acid filter to prevent a clogged filter from impeding boric acid flow. In all aspects of control and indication, this valve is identical to HCV-258 and HCV-265. The motor receives power from MCC-3C2.

2.590 HCV-268 has a 1/8-inch hole drilled in the upstream disk to prevent thermal binding and pressure locking (MR-FC-96-001).

••Location

2.591 HCV-268 is located on the 1016-foot elevation of Corridor 26 in the Auxiliary Building, by boric acid storage tanks CH-11A/B.

••Power Supplies

2.592 HCV-268 receives 480 VAC power from MCC-3C2.

••Instrumentation and Control

2.593 In all aspects of control and indication, (local, remote, interlocks, alarms and indications) HCV-268 is identical to HCV-258 and HCV-265.

••Failure Modes

2.594 The following paragraphs describe system response in the event HCV-268 was to fail open or closed.

2.595 In terms of operational characteristics, this valve is identical to HCV-258 and HCV-265. Therefore, its failure modes are also the same.

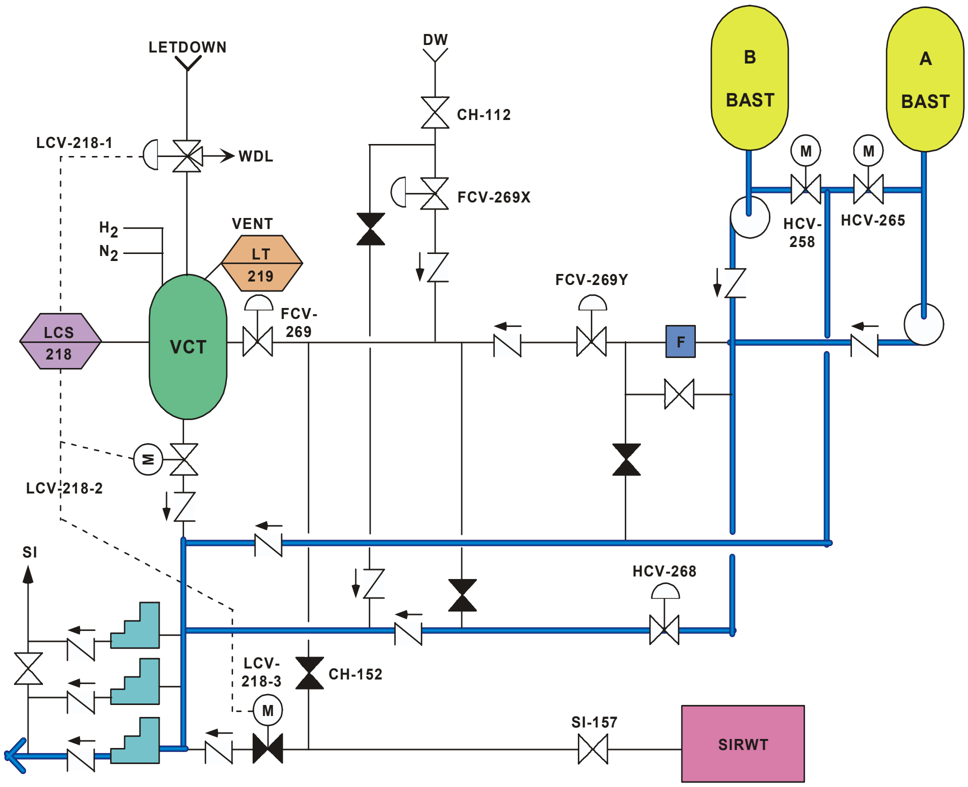


FIGURE 3-3: EMERGENCY BORATION

3.30 When the RCS is depressurized, all makeup from the CVCS is supplied to the charging pump suction. The boric acid pump and/or demineralized water header supplies sufficient head to provide flow through the idle charging pump and into the RCS, via the high-pressure safety injection header.

3.31 The manual blend mode is normally used when makeup to the SIRWT is required in order to restore level following the use of the SIRWT for safety injection or refueling.

Abnormal System Operation

3.32 The following describes system operation during a loss of letdown and a loss of charging.

••Loss of Letdown

3.33 Loss of letdown can be affected by the closing of any of the following valves:

- Letdown stop valve (TCV-202)
- Letdown flow control valves (LCV-101-1 and 101-2)

normally exist during makeup to the VCT. A locally read pressure indicating controller is connected to the outlet of each pump. These controllers provide the low-pressure condition input to the BORIC ACID PUMP CH-4A (CH-4B) AUTO MAKE-UP SIG ON AND DISCH PRESS LO alarm whenever discharge pressure drops below 40 psig.

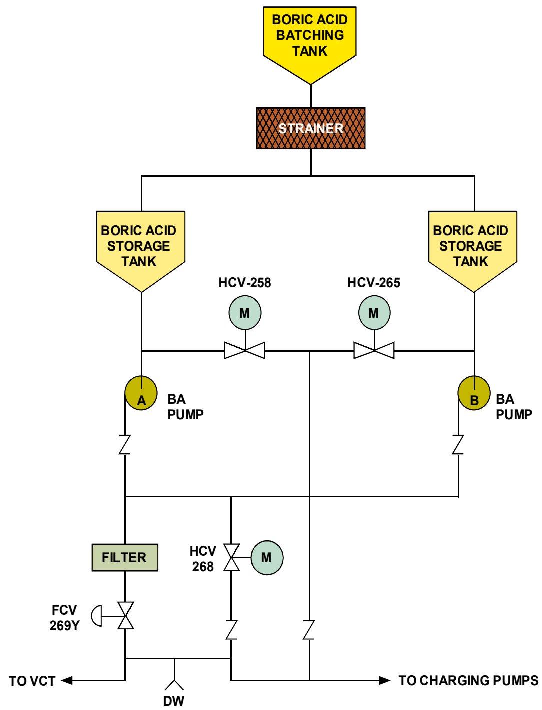


FIGURE 2-31: BORIC ACID SECTION

2.524 Each pump is driven by a 480 VAC, 30 horsepower motor. Pump CH-4A is powered from MCC-3C2 and CH-4B powered from MCC-4A2. Design head for each pump is 208 ft. (90 psig). Controls and indications for the motors are identical, therefore the following explanation will address the pumps generically, with differences being noted.

••Location

2.525 The boric acid pumps are located in Corridor 26, on the ground floor near the northwest corner of the Auxiliary Building.

••Power Supplies

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

Figure 1-11 - 480 VAC Distribution

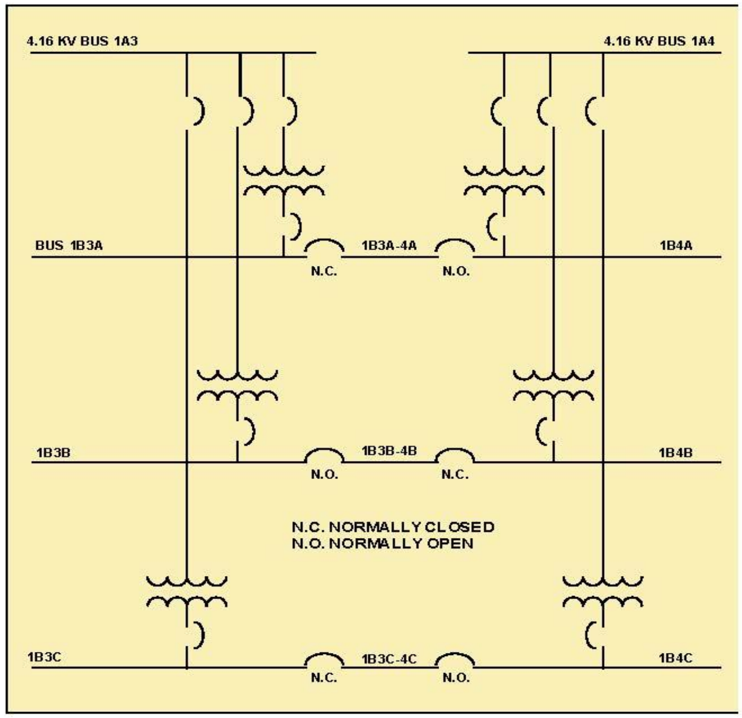
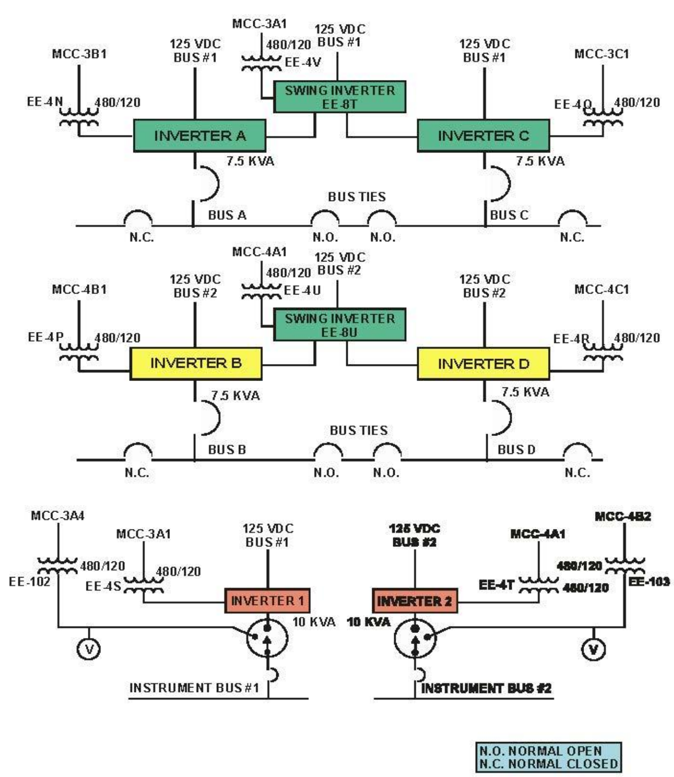


Figure 1-12 - Instrument AC Distribution



Source question for question 82

QUESTIONS REPORT
for ILO EXAM BANK

07-17-03 003

The plant was operating at 100% steady state power when the reactor tripped due to a loss of power to bus 1A3. All control room actions for Emergency Boration were taken.

Assuming bus 1A3 remains deenergized, which of the following local operations would result in emergency boration flow?

A. Opening HCV-258, "CH-11B Gravity Feed Valve"

B. Opening HCV-265, "CH-11A Gravity Feed Valve"

C.✓ Opening HCV-268, "Boric Acid Pump Discharge Header Isolation Valve"

D. Opening LCV-218-3, "Charging Pump Suction SIRWT Isolation Valve"

QUESTION NUMBER: 083

The reactor is operating at full power with the following conditions:

- Wide Range NI Channel "D" failed 72 hours ago.
- Wide Range NI Channel "C" failed 48 hours ago.
- Wide Range NI Channels "A" and "B" just failed

What action must be taken per Technical Specifications assuming all four Wide Range Channels remain inoperable?

A. Immediately, trip the reactor and enter EOP-00.

B. Immediately, begin a power reduction to less than 70% power.

C.✓ 4 days from now, initiate a plant shutdown.

D. 7 days from now, initiate a plant shutdown.

Question # 83 Rev 0

KA #: 000033 AA2.10 Tier 1 Group 2: Loss of Intermediate Range Nuclear Instrumentation
Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Tech-Spec limits if both intermediate-range channels have failed.
Importance 3.1 / 3.8

CFR Number: 55.43(b)(2)
Facility operating limitations in the technical specifications and their bases.

Fort Calhoun Objective:
Explain the indication provided on AI-212 provided by WR NIS.

Question Pedigree
New question.

K/A Fit:
Question addresses Tech Spec actions if all 4 wide range NI channels become inoperable.

Choice A:
Distractor: Plausible because this would be the action to take if all 4 power range NI channels failed. Incorrect because a trip is not required.

Choice B:
Distractor: Plausible because this would be the required action if 2 power range NI channels failed. Incorrect because a power reduction is not required at this time.

Choice C:
Correct answer: With reactor power above 15%, Tech Spec 2.15 requires that a reactor shutdown be initiated 7 days after wide range channel "D" becomes inoperable. Since it became inoperable 3 days ago, 4 days remain until a shutdown must be initiated.

Choice D:
Distractor: Plausible because Tech Spec 2.15 does have a 7 day LCO. Incorrect because 3 days of the allowable 7 have already elapsed.

KA#:	000033 AA2.10	Bank Ref #:	N/A
LP# / Objective:	0712-18 02.13	Exam Level:	SRO-2
Cognitive Level:	HIGH	Source:	NEW
Reference:	TS 2.15	Handout:	NONE

2.0 LIMITING CONDITIONS FOR OPERATION

* 2.15 Instrumentation and Control Systems (Continued)

If after 24 hours from time of initiating a hot shutdown procedure at least one inoperable engineered safety features or isolation functions channel has not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours. This specification applied to the high rate trip-wide range log channel when the plant is at or above 10^{-4}% power and is operating below 15% of rated power.

(3) In the event the number of channels on a particular engineered safety features (ESF) or isolation logic subsystem in service falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," sufficient channels shall be restored to OPERABLE status within 48 hours so as to meet the minimum limits or the reactor shall be placed in a hot shutdown condition within the following 12 hours; however, operation can continue without containment ventilation isolation signals available if the ventilation isolation valves are closed. If after 24 hours from time of initiating a hot shutdown procedure sufficient channels have not been restored to OPERABLE status, the reactor shall be placed in a cold shutdown condition within the following 24 hours.

(4) In the event the number of channels of those particular systems in service not described in (3) above falls below the limits given in the columns entitled "Minimum Operable Channels" or "Minimum Degree of Redundancy," except as conditioned by the column entitled "Permissible Bypass Conditions," the reactor shall be placed in a hot shutdown condition within 12 hours. If minimum conditions for engineered safety features or isolation functions are not met within 24 hours from time of discovering loss of operability, the reactor shall be placed in a cold shutdown condition within the following 24 hours. If the number of OPERABLE high rate trip-wide range log channels falls below that given in the column entitled "Minimum Operable Channels" in Table 2-2 and the reactor is at or above 10^{-4}% power and at or below 15% of rated power, reactor critical operation shall be discontinued and the plant placed in an operational mode allowing repair of the inoperable channels before startup or reactor critical operation may proceed.

If during power operation, the rod block function of the secondary CEA position indication system and rod block circuit are inoperable for more than 24 hours, or the plant computer PDIL alarm, CEA group deviation alarm and the CEA sequencing function are inoperable for more than 48 hours, the CEAs shall be withdrawn and maintained at fully withdrawn and the control rod drive system mode switch shall be maintained in the off position except when manual motion of CEA Group 4 is required to control axial power distribution.

(5) In the event that the number of operable channels of the listed Alternate Shutdown Panels or the Auxiliary Feedwater Panel instrumentation or control circuits falls below the required number of channels, either restore the required number of channels to OPERABLE status within seven (7) days, or be in hot shutdown (Mode 3) within the next twelve hours. This specification is applicable in Modes 1 and 2.

2.0 LIMITING CONDITIONS FOR OPERATION

*2.15 Instrumentation and Control Systems (Continued)

Function/Instrument or Control Parameter	Location	Required Number of Channels
1. Reactivity Control		
a. Source Range Power	AI-212	1
b. Reactor Wide Range Logarithmic Power	AI-212	1
2. Reactor Coolant System Pressure Control		
a. Pressurizer Wide Range Pressure (0-2500 psia)	AI-179	1
3. Decay Heat Removal via Steam Generators		
a. Reactor Coolant Hot Leg Temperature	AI-185	1 (Note 1)
b. Reactor Coolant Cold Leg Temperature	AI-185	1 (Note 1)
c. Steam Generator Pressure	AI-179	1 per Steam Generator
d. Steam Generator Narrow Range Level	AI-179	1 per Steam Generator
e. Steam Generator Wide Range Level	AI-179	1 per Steam Generator
4. Reactor Coolant System Inventory Controls		
a. Pressurizer Level	AI-185	1
b. Volume Control Tank Level	AI-185	1
c. Charging Pump CH-1B and its associated controls	AI-185	1
d. Charging Isolation Valve Control	AI-185	1
5. Transfer Functions		
a. All Transfer Switches/Lockout Relays	AI-185	1
b. All Transfer Switches/Lockout Relays	AI-179	1

Note 1: One reactor coolant hot leg temperature indication and one reactor coolant cold leg temperature indication channel must both be operable on the same steam generator (i.e., RC-2A or RC-2B).

* TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition	Test, Maintenance and Inoperable Bypass
1	Manual (Trip Buttons)	1	None	None	N/A
2	High Power Level	2[(b)(c)]	1[(c)]	Thermal Power Input Bypassed below 10^{-4}% of Rated Power[(a)(d)]	(e)
3	Thermal Margin/Low Pressurizer Pressure	2[(b)]	1	Below 10^{-4}% of Rated Power[(a)(d)]	(e)
4	High Pressurizer Pressure	2[(b)]	1	None	(e)
5	Low R.C. Flow	2[(b)]	1	Below 10^{-4}% of Rated Power [(a)(d)]	(e)
6	Low Steam Generator Water Level	2/Steam Gen[(b)]	1/Steam Gen	None	(e)
7	Low Steam Generator Pressure	2/Steam Gen[(b)]	1/Steam Gen	Below 600 psia[(a)(d)]	(e)
8	Containment High Pressure	2[(b)]	1	During Leak Test	(e)
9	Axial Power Distribution	2[(b)(c)]	1[(c)]	Below 15% of Rated Power[(g)]	(e)
10	High Rate Trip-wide Range Log Channels	2[(b)]	1	Below 10^{-4}% and above 15% of Rated Power[(a)(g)]	(e)
11	Loss of Load	2[(b)]	1	Below 15% of Rated Power[(g)]	(e)
12	Steam Generator Differential Pressure	2[(b)]	1	None	(e)

a. Bypass automatically removed.

b. Specification 2.15(2) is applicable.

* **TABLE 2-2**
(Continued)

c. If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d. For low power physics testing this trip may be bypassed up to 10^{-1}% of rated power.

e. Specification 2.15(1) is applicable.

f. Deleted.

g. For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

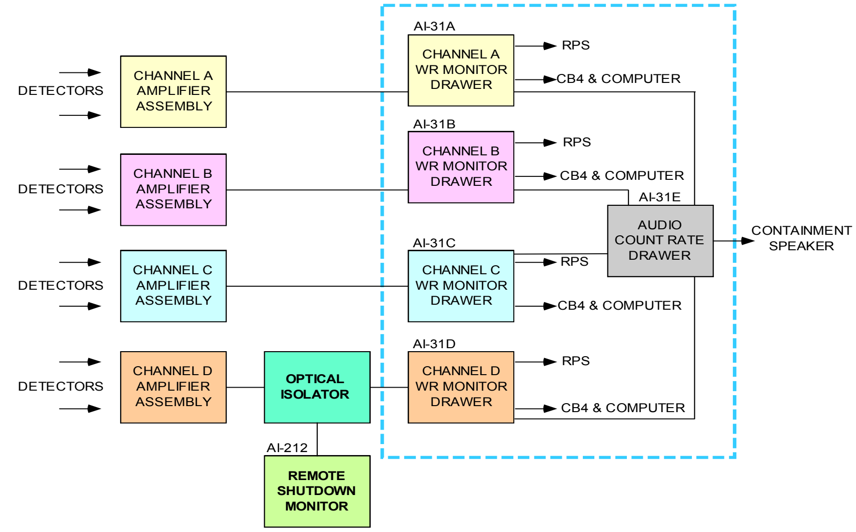


FIGURE 2-1: WIDE RANGE NI SYSTEM OVERVIEW

2.5 In addition, an Appendix R channel (channel D) consists of an optical isolator and wide range monitor. A single audible count rate chassis is provided selectable to any one of the four channels.

2.6 Wide Range NI System channel D (NI-004) provides input to an independent signal processor at alternate shutdown panel AI-212. For this channel the amplifier signals are fed to an optical isolator NA-004 where the signals are split to two identical wide range monitors. This channel is designed so that a failure of either wide range monitor cannot affect the operation of the other.

••Location

2.7 Four cabinets (AI-31A, B, C and D) each house one wide range NI channel along with the power range channels. The cabinets are located side by side in the Control Room.

2.8 Alternate shutdown panel AI-212 houses a display for channel D and is located in the upper electrical penetration area (Room 57) of the Auxiliary Building.

••Power Supplies

2.9 Wide range NI channels A, B, C and D are powered from their respective 120 VAC instrument buses, AI-40A/B/C/D.

QUESTION NUMBER: 084

A fire was detected and confirmed in Room 19.

What procedure should be used by the Control Room Operators to identify safe shutdown equipment potentially threatened by the fire and operator actions to be taken?

A. AOP-06-01, "Fire Emergency Auxiliary Building Radiation Controlled Areas and Containment."

B✓ AOP-06-02, "Fire Emergency Uncontrolled Areas of Auxiliary Building."

C. Standing Order G-28, "Station Fire Plan."

D. Standing Order G-102, "Fire Protection Program Plan."

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 84 Rev 0

KA #: 000067 AA2.12 Tier 1 Group 2: Plant Fire on Site
Ability to determine and interpret the following as they apply to the Plant Fire on Site:
Location of vital equipment within fire zone
Importance 2.9 / 3.9

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the Fire Emergency AOP to mitigate the consequences of a fire on the plant site.

Question Pedigree
New question.

K/A Fit:
Question addresses procedure used to identify safe shutdown equipment within a fire zone.

Choice A:
Distractor: Plausible because AOP-06-01 is used to identify safe shutdown equipment in the RCA and containment. Incorrect because Room 19 is in the uncontrolled area of the Auxiliary Building.

Choice B:
Correct answer: AOP-06-02 is used to identify safe shutdown equipment in Room 19.

Choice C:
Distractor. Plausible because SO-G-28 does contain information about safe shutdown equipment located in various areas of the plant. Incorrect because SO-G-28 provides information to the Fire Brigade Incident Commander and Fire Fighters.

Choice D:
Distractor: Plausible because SO-G-102 does address the Fire Protection Program. Incorrect because it does not identify safe shutdown equipment in specific plant areas.

KA#:	000067 AA2.12	Bank Ref #:	N/A
LP# / Objective:	0717-06 01.00	Exam Level:	SRO-5
Cognitive Level:	LOW	Source:	NEW
Reference:	AOP-06-02	Handout:	NONE

Section XII - Fire in Room 19 (AFW/Air Compressor Area)

1.0 PURPOSE

This procedure provides steps to be followed in the event of a fire in Room 19 (AFW/Air Compressor Area). It provides a Control Room strategy that will protect credited Fire Safe Shutdown equipment. The following information is provided to assist in understanding the strategy employed by this procedure:

Area Name: Room 19 (AFW/Air Compressor Area)
Location: Auxiliary Building - 989' elev.
Fire Area No: 32

Credited Safe Shutdown Equipment Threatened by This Fire:

4160 Volt Electrical Distribution System
480 Volt Electrical Distribution System
SI-1A/B/C, HPSI Pumps
Charging System
PCV-103-1, Pressurizer Spray Valve
PCV-103-2, Pressurizer Spray Valve
HCV-150, PORV Block Valve
HCV-151, PORV Block Valve
PCV-102-1, PORV
PCV-102-2, PORV
FW-6, Motor Driven AFW Pump
FW-10, Steam Driven AFW Pump

Electrical Supply Panels Located in This Area: None

Instrumentation Available for Fire Area:

- A/LT-911
- A/LT-912
- A/PT-902
- A/PT-905
- A/TE-112C
- A/TE-112H
- B/LT-911
- B/LT-912
- B/PT-902
- B/PT-905
- B/TE-112C
- B/TE-112H
- LT-101X
- LT-101Y
- NE-003
- NE-004
- PT-105
- PT-115

QUESTION NUMBER: 085

Given the following plant conditions:

- Fuel movement is in progress inside containment and at the spent fuel pool
- The containment equipment hatch is open
- Both PAL doors are closed
- All other containment penetrations open to the atmosphere are closed
- VIAS and CRHS actuations have occurred
- Radiation Monitors RM-050 and RM-051 are in alarm
- The refueling SRO reports that a fuel assembly has been dropped

Which one of the following actions should be taken as a result of these conditions?

A.✓ Enter AOP-08, "Fuel Handling Incident," and direct securing of the Room 66 openings to isolate any release to the atmosphere.

B. Enter AOP-08, "Fuel Handling Incident," and ensure that the AB supply and exhaust fans are operating as required to maintain a negative pressure.

C. Enter AOP-09, "High Radioactivity," and ensure that the AB supply and exhaust fans are operating as required to maintain a negative pressure.

D. Enter AOP-12, "Loss of Containment Integrity," and direct securing of the room 66 openings to isolate any release to the atmosphere.

Question # 85 Rev 0

KA #: 000036 2.4.18 Tier 1 Group 2: Fuel Handling Incidents
Knowledge of the specific bases for EOPs.
Importance 3.3 / 4.0

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the Fuel Handling Incident Procedure to mitigate the consequences in the event an irradiated fuel assembly is dropped or damaged.

Question Pedigree
New question.

K/A Fit:
Question addresses procedures, procedural action and their bases during a fuel handling incident.

Choice A:
Correct answer: AOP-08 is entered and it directs securing the Room 66 openings.

Choice B:
Distractor: Plausible because AOP-08 is entered and fuel movement is in progress in the Auxiliary Building. Incorrect because RM-050 and RM-051 monitor the containment atmosphere and AOP-08 only realigns AB ventilation if the incident is in the AB.

Choice C:
Distractor: Plausible because RM-050 and RM-051 alarming are entry conditions for AOP-09. Incorrect because AOP-09 contains a not that AOP-08 provides guidance for dropped fuel assembly.

Choice D:
Distractor: Plausible because the containment equipment hatch is open. Incorrect because this is not an AOP-12 action.

KA#:	000036 2.4.18	Bank Ref #:	N/A
LP# / Objective:	0717-08 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	TBD-AOP-08	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

NOTE

It is required to close the following within one hour: Room 66 Roll-up Doors and Room 66 Construction Access Opening or the Equipment Hatch, all containment penetrations open to the outside atmosphere and one door in the PAL.

5. Secure Room 66 Openings by performing the following:
   - Direct Security Shift Manager to close Room 66 Roll-up Doors 1009-1, 1013-3, and 1013-4.
   - Direct Maintenance to install cover for the Room 66 Construction Access Opening.

5.1 **IF** unable to close Room 66 Roll-up Doors and Room 66 Construction Access Opening,
THEN direct Shift Outage Manager to close the Equipment hatch.

6. Direct Shift Outage Manager to close all containment penetrations open to the outside atmosphere.

7. **WHEN** Room 66 Roll-up Doors and Room 66 Construction Access Opening **OR** Equipment Hatch and all applicable containment penetrations have been closed,
THEN direct the EONA to close at least one PAL door.

INSTRUCTIONS

8. Verify VIAS actuation PER Attachment A, VIAS Actuation.

9. **IF** the incident was in the Auxiliary Building,
THEN align ventilating equipment by performing the following steps (AI-44):

a. Ensure VA-66, Spent Fuel Area Charcoal Filter, is in the filtered mode.

(continue)

CONTINGENCY ACTIONS

8.1 **IF** VIAS did **NOT** actuate,
THEN ensure VIAS actuates by performing the following steps:

a. Trip VIAS using CRHS test switches.

b. Verify VIAS actuation PER Attachment A, VIAS Actuation.

INSTRUCTIONS | CONTINGENCY ACTIONS

9. (continued)

b. Verify at least one of the Stack Radiation Monitors has power and the pump is energized (AI-33C).

b.1 **IF** Stack Radiation Monitors are not powered or pumps are not energized,
THEN align a power source and energize the associated pump:

- RM-062 and RM-063 (MCC-4C2)
- RM-052 (MCC-4C2) (normal)
- RM-052 (MCC-3B1), Attachment 2 of OI-RM-1, Radiation Monitoring

c. Ensure a negative pressure exists in the Auxiliary Building by performing the following steps:

1) Ensure **ANY** two of the following Auxiliary Building Exhaust Fans are in service:

- VA-40A
- VA-40B
- VA-40C

2) Ensure **ONE** Auxiliary Building Supply Fan is in service:

- VA-35A
- VA-35B

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. **CAUTIONS**

None

B. **NOTES**

1. AOP-08, Fuel Handling Incident, provides guidance in the event an irradiated fuel assembly is dropped or otherwise damaged.

QUESTION NUMBER: 086

The following plant conditions exist:

- The reactor is at 100% power
- The RC-3A "SEAL LEAKAGE FLOW HI" annunciator is in alarm
- VCT pressure is 50 psia.
- RC-3A upper seal inlet pressure is 110 psia
- RC-3A middle seal inlet pressure is 1095 psia
- Pressurizer pressure is 2100 psia

What action should be taken as a result of these conditions per AOP-35, Reactor Coolant Pump Malfunctions?"

A.✓ Stay in AOP-35 and continue to monitor seal parameters.

B. Perform a plant shutdown using OP-4, "Load Change and Normal Power Operation" and OP-3A, "Plant Shutdown."

C. Perform a plant shutdown using AOP-5, "Emergency Shutdown."

D. Trip the reactor and enter EOP-00, "Standard Post-Trip Actions" then trip RCP RC-3A.

Question # 86 Rev 0

KA #: 003000 2.1.28 Tier 2 Group 1: Reactor Coolant Pump System
Knowledge of the purpose and function of major system components and controls.
Importance 4.1 / 4.1

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
EXPLAIN the operation of the RCP seal package.

Question Pedigree
Modified question. Stem conditions modified to make another choice correct and selection of appropriate procedure added.

K/A Fit:
Question addresses loss of a reactor coolant pump seal.

Choice A:
Correct answer: The upper RCP seal has failed. AOP-35 directs the operators to continue to monitor seal parameters.

Choice B:
Distractor: Plausible because AOP-35 directs a plant shutdown if two seals have failed. Incorrect because only one seal has failed and AOP-05 would be used for the shutdown if two seals failed.

Choice C:
Distractor: Plausible because this is the correct answer if two seals failed. Incorrect because only one seal failed.

Choice D:
Distractor: Plausible because this would be the correct answer if more than two seals failed. Incorrect because only one seal failed.

KA#:	003000 2.1.28	Bank Ref #:	07-11-20 154
LP# / Objective:	0711-20 01.07D	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	AOP-35	Handout:	NONE

Section I - RCP Seal Failure

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. **CAUTIONS**

1. All indications shall be verified by independent instrumentation (where available).

B. **NOTES**

1. RCP operating parameters are available on the ERF computer on Page 342 in a summary format for comparison between pumps and on pages 440, 441, 442 and 443 for individual pumps.

2. A differential pressure across a seal stage of less than 100 psid indicates a seal stage failure.

3. Middle seal inlet pressure less than 1000 psig with Controlled Bleedoff less than 0.75 gpm, indicates blockage of the lower pressure breakdown device.

Section I - RCP Seal Failure

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS

1. Verify **NONE** of the following conditions exist for any RCP:
   - Lower seal cavity temperature exceeds 200°F
   - Vapor seal pressure equals RCS pressure
   - More than two seals have failed

CONTINGENCY ACTIONS

1.1 **IF** any condition exists **AND** the Reactor is critical, **THEN** stop the affected RCPs by performing the following steps:

   a. Trip the Reactor.

   b. IMPLEMENT EOP-00, Standard Post Trip Actions.

   c. Stop the affected RCPs.

   d. GO TO Section 5.0, Exit Conditions.

1.2 **IF** any condition exists **AND** the Reactor is shutdown, **THEN** stop the affected RCPs.

Section I - RCP Seal Failure

INSTRUCTIONS	CONTINGENCY ACTIONS

2. Verify proper RCP seal operation by **ALL** of the following conditions:
   - ΔP across each RCP seal is greater than 100 psid
   - Middle seal inlet pressure is greater than 500 psig with Controlled Bleedoff flow greater than 0.5 gpm
   - Bleedoff temperature is less than 250°F

(continue)

2.1 **IF** only one seal per RCP has failed, **THEN** continue to monitor seal parameters for the affected RCPs.

2.2 **IF ANY** of the following conditions exist:
   - Two seals have failed
   - Middle seal inlet pressure is less than or equal to 500 psig and Controlled Bleedoff flow is less than 0.5 gpm
   - Bleedoff temperature is greater than or equal to 250°F

   AND the Reactor is critical,
   THEN stop the affected RCPs by performing the following steps:

   a. Commence an immediate Plant Shutdown PER AOP-05, Emergency Shutdown.

   b. Continue to monitor seal parameters for the affected RCPs.

   c. Monitor for entry conditions of AOP-22, Reactor Coolant Leak.

(continue)

QUESTIONS REPORT
for ILO EXAM BANK

07-11-20 154

The RO reports the following plant conditions:

- The reactor is at 100% power
- RCS pressure is 2100 psia
- The RC-3A "SEAL LEAKAGE FLOW HI" annunciator is in alarm
- VCT pressure is 50 psia.
- RC-3A middle seal inlet pressure is 110 psia
- RC-3A upper seal inlet pressure is 80 psia

What actions should you direct your crew to take?

A. Monitor the RC-3A's seals on the ERF computer. Continue full power operation.

B. Perform a normal plant shutdown using OP-3A, then shutdown RC-3A.

C✓ Perform an emergency plant shutdown using AOP-05, then shutdown RC-3A.

D. Trip the reactor, then shutdown RC-3A and enter EOP-00.

QUESTION NUMBER: 087

Five days ago the Army Corps of engineers notified the plant that the level of the Missouri River at the plant was expected to rise to 1007 feet MSL over the next 24-36 hours.

Current plant conditions:

- The plant has been placed in cold shutdown with shutdown cooling in service
- The river level at the plant is 1004 feet MSL and rising
- All 4160 volt buses are being supplied by 161 KV offsite power

Which one of the following actions should be taken per FCS procedures?

A.✔ Transfer loads on one vital bus to a diesel generator per AOP-01, "Acts of Nature."

B. Transfer loads on both vital buses to the diesel generators per AOP-01, "Acts of Nature."

C. Transfer loads on 480 V Buses 1B3A, 1B4A and 1B3A-4A to 13.8 KV supply per EOP/AOP Attachment 5, "Energizing 480 V Buses From 13.8 KV."

D. Transfer loads on 480 V Buses 1B3C, 1B4C and 1B3C-4C to 13.8 KV supply per EOP/AOP Attachment 5, "Energizing 480 V Buses From 13.8 KV."

Question # 87 Rev 0

KA #: 064000 2.4.20 Tier 2 Group 1: Emergency Diesel Generators
Knowledge of operational implications of EOP warnings, cautions, and notes.
Importance 3.8 / 4.3

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Describe how the plant responds to the following: Flood.

Question Pedigree
New question.

K/A Fit:
Question addresses procedural actions for diesel generators related to several notes in the AOP for external flooding

Choice A:
Correct answer: AOP-01 directs transferring loads to one diesel generator if the river level is expected to exceed 1006 feet 6 inches. Only one diesel generator is used to conserve fuel.

Choice B:
Distractor: Plausible because vital loads are transferred to the diesel generators. Incorrect because only one diesel generator is used.

Choice C:
Distractor: Plausible because there is an EOP/AOP Attachment for powering 480 volt buses from 13.8 KV. Incorrect because the attachment is not used with AOP-01 and does not provide power to the listed 480 volt buses.

Choice D:
Distractor: Plausible because there is an EOP/AOP Attachment for powering the listed 480 volt buses from 13.8 KV. Incorrect because the attachment is not used with AOP-01.

KA#:	064000 2.4.20	Bank Ref #:	N/A
LP# / Objective:	0717-01 01.02A	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	AOP-01	Handout:	NONE

Section I - Flood

3.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. CAUTIONS

None

B. NOTES

1. The National Weather Service and the U.S. Army Corps of Engineers provide continuous forecasts regarding the weather and the Missouri River flood stage.

2. 1004 feet is the elevation of the plant site and the Security Building floor.

3. Terminals in the 161KV Switchyard Control Building are flooded at an elevation of 1005 feet 6 inches.

4. Terminals in the West 345KV Switchyard Control Building are flooded at an elevation of 1005 feet 9 inches.

5. 1007 feet 6 inches is the elevation of the Intake Structure floor and the elevation at which the Turbine Building Truck Bay overflows and where the terminals in the East 345KV Switchyard Control Building are flooded.

6. Installation of Flood Barriers in the Intake Structure and Auxiliary Building will protect those areas to at least 1014 feet. Internal building leakage will need to be controlled by temporary dewatering pumps.

7. Maintenance is responsible for the installation and removal of flood control barriers and sandbagging operations.

8. Technical Specification 2.16, River Level, addresses operating limitations associated with river level.

9. On an upstream dam failure river level at the station is automatically assumed to exceed 1004 feet. A failure of the nearest upstream dam (i.e., Gavins Point Dam) would result in the flood surge reaching the Station in approximately 44 hours. The estimated rate of rise is two inches per hour once it arrives.

3.0 **PRECAUTIONS** (continued)

B. NOTES (continued)

10. AOP-01 directs flood protection of vital SSCs. Non-vital protection is directed by FCSG-64, External Flooding of the Site.

11. To convert the Blair NE river level gauge to the elevation at Fort Calhoun Station add 975.74 ft.

12. Following the loss of off-site power, utilizing a single Emergency Diesel Generator to power station loads minimizes Diesel Fuel Oil consumption.

13. River level is the water level as measured at the FCS Intake Structure by L1900 (ERF) or approved equivalent level instrumentation.

Section I – Flood

INSTRUCTIONS	CONTINGENCY ACTIONS

NOTES

1. The goal of transferring 4160V loads to a diesel-generator before losing off-site power should be balanced against the goal of maximizing fuel oil inventory. The decision of when to transfer loads should consider the rate at which the flood level is rising.
2. Bus loading should be less than 1550kw for the first 24 hours and 1381kw thereafter to ensure a 7 day supply of Diesel Fuel without an available offsite source.

11. **IF** river level reaches 1004 feet and is expected to exceed 1005 feet 6 inches **THEN** transfer 4160 volt power to the Emergency Diesel Generators by performing the following:

    a. Ensure the Reactor is Tripped.

    b. **IF** Bus 1A3 is to remain energized **THEN** perform the following:

    1) Deenergize Bus 1A3 by opening **BOTH** of the following breakers:

    - 1A13
    - 1A33

    (continue)

    b.1 **IF** Bus 1A4 is to remain energized, **THEN** perform the following:

    1) Deenergize Bus 1A4 by opening **BOTH** of the following breakers:

    - 1A24
    - 1A44

    (continue)

Section I – Flood

INSTRUCTIONS

11.b. (continued)

2) Ensure that Bus 1A3 is powered from Diesel Generator 1.

(continue)

CONTINGENCY ACTIONS

b.1 (continued)

2) Ensure that Bus 1A4 is powered from Diesel Generator 2.

3) **IF TWO** CCW Pumps are running,
THEN stop **ONE** CCW Pump, AC-3A or AC-3C.

(continue)

INSTRUCTIONS

CONTINGENCY ACTIONS

NOTE

A LPSI pump is required to be started only if on Shutdown Cooling

11.b. (continued)

3) Attempt to start **ALL** of the following equipment powered from Bus 1A3:

- **ONE** CCW Pump, AC-3A/C
- **ONE** Raw Water Pump, AC-10A/C
- AC-9A, Bearing Water Pump
- **ONE** Air Compressor CA-1A/C
- CH-1A, Charging Pump
- VA-7C, Containment Vent Fan
- SI-1A, LPSI Pump

(continue)

b.1 (continued)

4) Attempt to start **ALL** of the following equipment powered from Bus 1A4:

- AC-3B, CCW Pump
- **ONE** Raw Water Pump, AC-10B/D
- AC-9B, Bearing Water Pump
- Air Compressor CA-1B
- **ONE** Charging Pump, CH-1B/C
- VA-7D, Containment Vent Fan
- SI-1B, LPSI Pump

(continue)

QUESTION NUMBER: 088

The plant was in hot shutdown with T-cold at 532°F and all CEAs inserted when the high containment trip bistables all tripped on high containment pressure

- PPLS,CPHS and SGLS actuations all occurred
- Three charging pumps, Two HPSI pumps, 2 LPSI pumps and Containment Spray Pump SI-3B are operating
- Containment Spray Pump SI-3A Failed to start and cannot be manually started
- Containment Cooling Fans, VA-3A and VA-3B, are running
- Containment Cooling Fans, VA-7C and VA-7D, are not running
- Containment pressure is 35 psig and slowly lowering

What action should be taken in response to these conditions?

A. Enter AOP-40, "Overcooling/Excessive Steam Demand," and manually start Containment Spray Pump SI-3C.

B. Enter AOP-40, "Overcooling/Excessive Steam Demand," and manually start VA-7C and VA-7D.

C. Enter EOP-00, "Standard Post Trip Actions," and manually start Containment Spray Pump SI-3C.

D.✓ Enter EOP-00, "Standard Post Trip Actions," and manually start VA-7C and VA-7D.

Question # 88 Rev 0

KA #: 013000 A2.02 Tier 2 Group 1: Engineered Safety Features Actuation System
Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Excess steam demand
Importance 4.3 / 4.5

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
STATE plant conditions that require manual initiation of engineered safeguards.

Question Pedigree
New question.

K/A Fit:
Question address using procedures to mitigate a steam line break.

Choice A:
Distractor: Plausible because the plant is shutdown and only one containment spray pump is running. Incorrect because SO-O-1 requires entry into EOP-00 if T-cold is 525°F or greater at the start of the event and EOP-05 contains a caution that says "Do not run SI-3B and SI-3C at the same time."

Choice B:
Distractor: Plausible because the plant is shutdown. Incorrect because SO-O-1 requires entry into EOP-00 if T-cold is above 525°F at the start of the event.

Choice C:
Distractor: Plausible because EOP-00 should be entered and only one containment spray pump is operating. Incorrect because EOP-05 contains a caution that says "Do not run SI-3B and SI-3C at the same time."

Choice D:
Correct answer: EOP-00 should be entered because T-cold was above 525°F at the start of the event. EOP-00 directs starting VA-7C and VA-7D.

KA#:	013000 A2.02	Bank Ref #:	N/A
LP# / Objective:	0712-14 03.01	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-00	Handout:	NONE

INSTRUCTIONS

15. (continued)

f. Verify that **BOTH** of the following Containment conditions exist:

- Containment pressure is less than 3.0 psig
- Containment temperature is less than 120°F (ERF "CI" display)

(continue)

CONTINGENCY ACTIONS

f.1 **IF** Containment pressure is greater than or equal to 3.0 psig, **OR** Containment temperature is greater than or equal to 120°F, **THEN** initiate Containment Cooling by performing the following:

1) Start **ALL** of the following available Containment Vent Fans:

- VA-3A
- VA-7C
- VA-3B
- VA-7D

2) Ensure CCW is supplied to the coils of the operating Containment Vent Fans.

(continue)

4.0 PRECAUTIONS

The following specific cautions and notes apply prior to or throughout this procedure.

A. **CAUTIONS**

1. UHEs inside Containment can magnify instrument inaccuracies. A single indication may be unreliable. All available indications and actual Plant response should be used to evaluate actions and Plant conditions. TSC staff should assist in evaluating these uncertainties through use of the Electrical Equipment Qualification Manual.

2. Do not allow Diesel Generator loads to exceed power and current rating limits.

3. Do not run SI-3B and SI-3C at the same time.

4. No more than three RCP's shall be in operation when RCS temperature is less than 500°F.

B. **NOTES**

1. Floating Step BB, Minimizing DC Loads, requires operator action within 15 minutes of loss of either battery charger.

5.19.3 Guidance on Terminating a Procedure Before it is Completed

5.19.3.A If the procedure (OI, OP, ST etc.) being performed must be terminated prior to completion, the steps or sections between the point of termination and the steps for restoration shall be marked N/A. The authority to N/A shall be as specified in SO-G-7 Operating Manual, or SO-G-23 Surveillance Test Program.

The system may then be returned to normal lineup in accordance with the restoration steps of the procedure being used. If the procedure modified the plant system (e.g., installed test equipment) and it is preferred not to remove the modification, then the requirements of SO-O-25 Temporary Modification Control, shall be followed by indicating the modification in the Temporary Modification Index.

5.19.4 Procedural Guidance during Off-Normal Conditions

5.19.4.A The following information is offered to enable consistent, conservative response to abnormal events or conditions whereby implementation of a procedure, EOP, AOP or ARP is deemed appropriate.

5.19.4.A.1) **When the generator is on-line,** the Operator shall enter the Standard Post-Trip Action procedure (EOP-00) whenever an automatic or manual RPS Actuation/Reactor Trip occurs or if an EOP-00 entry condition is met.

5.19.4.A.2) **When $T_c \geq 525°F$ and the generator is off-line**, the Operator shall enter EOP-00 and subsequent EOP optimal recovery or functional recovery procedures, and follow the procedure as written to the extent possible, whenever an RPS actuation/reactor trip occurs (automatic or manual) or EOP-00 (Standard Post Trip Actions) entry condition is met. It should be noted that verbatim compliance is not required as it will be necessary to use operational judgment in applying the instructions of the EOP since plant configuration may be somewhat different than the configuration assumed when the EOPs were designed. For example, the instruction in EOP-00 which requires the Operator to verify output breakers 3451-4 and 3451-5 are open following a reactor/turbine trip would be inappropriate if 3451-4 or 3451-5 were closed for 345KV backfeed.

The EOPs do however offer sound guidance to an Operator when events occur in plant modes not assumed in current station procedures. Specific attention should be paid to the maintenance of safety functions.

QUESTION NUMBER: 089

Given the following plant conditions:

- A loss of load event resulted in a high pressure reactor trip
- Both of the Power-Operated Relief Valves (PORVs) opened
- PORV, PCV-102-1, failed to close when pressurizer pressure lowered below its setpoint
- Power was not available to PORV block valve, HCV-151, during performance of EOP-00, "Standard Post Trip Actions"
- Pressurizer level is 100%
- Power has now been restored to HCV-151

What Action should be taken as a result of these conditions?

A. Close HCV-151 and go to EOP-01, "Reactor Trip Recovery."

B. Enter EOP-03, "Loss of Coolant Accident," and close HCV-151.

C.✓ Enter EOP-03, "Loss of Coolant Accident," and begin a RCS cooldown.

D. Enter EOP-20, "Functional Recovery Procedure," and begin a RCS cooldown.

Question # 89 Rev 0

KA #: 010000 A2.03 Tier 2 Group 1: Pressurizer Pressure Control System
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: PORV failures
Importance 4.1 / 4.2

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
EXPLAIN the major strategy used to mitigate the consequences of a LOCA.

Question Pedigree
Modified question. The source question was used on the 2010 NRC exam.

K/A Fit:
Question addresses procedural actions to mitigate a stuck open PORV.

Choice A:
Distractor: Plausible because EOP-00 does have instructions for isolating a stuck open PORV. Incorrect because EOP-03 will be entered for a LOCA, even if isolated.

Choice B:
Distractor: Plausible because EOP-03 is the correct procedure and does contain steps to isolate the PORV. Incorrect because EOP-03 has a caution that says "DO NOT isolate a PORV if the pressurizer is water solid."

Choice C:
Correct answer: EOP-03 entry conditions are met and the caution in EOP-03 states "DO NOT isolate a PORV if the pressurizer is water solid. A cooldown must be commenced before isolation."

Choice D:
Distractor: Plausible because the cooldown is correct. Incorrect because EOP-20 entry conditions are not met.

KA#:	010000 A2.03	Bank Ref #:	2010 NRC Q 76
LP# / Objective:	0718-13 01.01	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	MODIFIED
Reference:	EOP-03	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

Do **NOT** isolate a PORV if the pressurizer is water solid. A cooldown must be commenced before isolation.

****

12. Verify PORVs and PZR Code Safety Valves are closed by performing the following:

    a. Check that **ALL** of the following PZR Quench Tank Alarms are clear (CB-1,2,3; A4):

    - "QUENCH TANK TEMP HI"
    - "QUENCH TANK PRESS HI"
    - "QUENCH TANK LEVEL HI-LO"

    b. Check that **ALL** of the following Relief Header Temperature Alarms are clear (CB-1,2,3; A4):

    - "PRESSURIZER PWR OPERATED RELIEF VALVE DISCH TEMP HI"
    - "PRESSURIZER SAFETY VALVE RC-141 DISCH TEMP HI"
    - "PRESSURIZER SAFETY VALVE RC-142 DISCH TEMP HI"

(continue)

12.1 **IF** RCS pressure is less than 2300 psia,
AND a PORV is open,
THEN isolate the PORV(s) by closing **ONE** PORV Block Valve for the associated PORV:

- PCV-102-1
  HCV-151

- PCV-102-2
  HCV-150

(continue)

QUESTIONS REPORT

for 2010 FCS NRC WRITTEN EXAM Rev 0

QUESTION NUMBER: 076

Given the following plant conditions:

- The plant has tripped from full power due to a High Pressurizer Pressure Trip
- EOP-00, STANDARD POST TRIP ACTIONS, were completed and EOP-01, REACTOR TRIP RECOVERY, was entered
- Several minutes later PPLS actuated
- All Engineered Safety Features operated as designed
- Pressurizer Pressure is now 931 psia
- The ATCO has tripped all Reactor Coolant Pumps
- Pressurizer Level is 100% (solid)
- Representative CET Temperature is 536°F
- PORV, PCV-102-2, is discovered to be open

Which one of the following actions should be taken?

A. Go to Diagnostic Actions of EOP-00, STANDARD POST TRIP ACTIONS, then enter EOP-03, LOSS OF COOLANT ACCIDENT, and close HCV-151, the block valve for PCV-102-2.

B. Stay in EOP-01, REACTOR TRIP RECOVERY, and reestablish letdown using EOP/AOP Attachment 23, RESTORATION OF LETDOWN.

C.✓ Go to Diagnostic Actions of EOP-00, STANDARD POST TRIP ACTIONS, then enter EOP-03, LOSS OF COOLANT ACCIDENT, and begin a plant cooldown.

D. Go to EOP-20, FUNCTIONAL RECOVERY PROCEDURE, implement EOP/AOP Floating Step A, HPSI STOP AND THROTTLE CRITERIA.

QUESTION NUMBER: 090

Given the following plant conditions:

- The reactor is operating at full power
- Instrument Air Dryer, CA-12, is in service
- The following annunciators are in alarm:
  - AIR DRYER TROUBLE
  - PLANT AIR PRESS LO
  - INSTRUMENT AIR PRESS LO
- Instrument air pressure is steady at 78 psig

What action should be taken in response to these conditions?

A.✓ Enter AOP-17,"Loss of Instrument Air," direct a building operator to open CA-197, "Bypass Control Valve PCV-1752 Bypass Valve," then place Instrument Air Dryer, CA-31, in service per OI-CA-1.

B. Enter AOP-17,"Loss of Instrument Air," Close PCV-1849A/B, Instrument Air Containment Isolation Valves then direct a building operator to place Instrument Air Dryer, CA-31, in service per OI-CA-1.

C. Trip the reactor, complete EOP-00,"Standard Post Trip Actions." Then enter EOP-20,"Functional Recovery Procedure, MVA-IA." direct a building operator to open CA-197, "Bypass Control Valve PCV-1752 Bypass Valve," then place Instrument Air Dryer, CA-31, in service per OI-CA-1.

D. Trip the reactor, complete EOP-00,"Standard Post Trip Actions. Then enter EOP-20,"Functional Recovery Procedure, MVA-IA." Close PCV-1849A/B, Instrument Air Containment Isolation Valves, then direct a building operator to place Instrument Air Dryer, CA-31, in service per OI-CA-1.

Question # 90 Rev 0

KA #: 078000 A2.01 Tier 2 Group 1: Instrument Air System
Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions
Importance 2.4 / 2.9

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Explain the principles of Abnormal operation of the Compressed Air System in terms of flow paths, major parameters, (temperature, pressure, flow, etc.), alarms and control devices.

Question Pedigree
New question.

K/A Fit:
Question addresses using procedures to mitigate an air dryer malfunction.

Choice A:
Correct answer: Instrument air pressure will be holding at 78 psig due to automatic operation of PCV-1752. AOP-17 directs that CA-197 be opened and CA-31 be placed in service.

Choice B:
Distractor: Plausible because AOP-17 is entered and closing PCV-1849A/B is an action in AOP-17. Incorrect because that step in AOP17 will not be reached for a dryer failure.

Choice C:
Distractor: Plausible because AOP-17 does direct tripping the plant if instrument air pressure gets low enough. Incorrect because instrument air pressure must fall below 50 psig before the plant is tripped.

Choice D:
Distractor: Plausible because AOP-17 does direct tripping the plant if instrument air pressure gets low enough. Incorrect because instrument air pressure must fall below 50 psig before the plant is tripped.

KA#:	078000 A2.01	Bank Ref #:	N/A
LP# / Objective:	0711-07 01.05	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	AOP-17	Handout:	NONE

INSTRUCTIONS		CONTINGENCY ACTIONS	
3.	**IF** the "AIR DRYER TROUBLE" alarm (CB-10,11; A11) annunciates, **THEN** direct an operator to proceed to AI-79 to determine the cause of the alarm (Room 19).		
4.	Direct all available operators to search for the source of the air leakage.		
5.	**IF** Instrument Air pressure is less than 80 psig, **THEN** verify PCV-1753, "SERVICE AIR SYSTEM AUTOMATIC ISOLATION VALVE", is closed (Room 19).	5.1	**IF** PCV-1753 is **NOT** closed, **THEN** close CA-121, "SERVICE AIR SUPPLY SYSTEM MANUAL ISOLATION VALVE" (Room 19).
6.	**IF** Instrument Air pressure returns to a normal 98-108 psig after isolating Service Air, **THEN** GO TO Section 5.0, Exit Conditions.		
7.	**IF** Instrument Air pressure is less than 78 psig, **THEN** verify that PCV-1752, "AIR DRYERS CA-31 & CA-12 BYPASS VALVE", is open (Room 19).	7.1	**IF** Instrument Air pressure is less than 78 psig **AND** PCV-1752 is closed, **THEN** open CA-197, "BYPASS CONTROL VALVE PCV-1752 BYPASS VALVE" (Room 19).

INSTRUCTIONS | CONTINGENCY ACTIONS

8. **IF ONE** of the following valves is open (Room 19):

- PCV-1752, "AIR DRYERS CA-31 & CA-12 BYPASS VALVE"
- CA-197, "BYPASS CONTROL VALVE PCV-1752 BYPASS VALVE"

AND Instrument Air pressure returns to a normal 98-108 psig,
THEN restore Instrument Air by performing the following steps:

a. Place CA-31, "BACKUP HEATLESS AIR DRYER", in service PER the Startup of Instrument Air Dryer CA-31 section of OI-CA-1, Compressed Air System Normal Operation.

b. GO TO Section 5.0, Exit Conditions.

QUESTION NUMBER: 091

Given the following sequence of events:

- The plant was operating at full power
- A group 4 CEA dropped into the core
- AOP-02, "CEA and Control System Malfunctions," was entered
- One minute later, another group 4 CEA dropped into the core

What action should be taken as a result of these events?

A. Reduce power to less than 70% using AOP-05, "Emergency Shutdown."

B. Realign the dropped CEAs to within 12 inches of all other CEAs in group 4 or place the reactor within hot shutdown using OP-4, "Load Change and Normal Power Operations."

C. Trip the reactor and perform EOP-00, "Standard Post Trip Actions" then re-enter AOP-02, "CEA and Control System Malfunctions."

D.✓ Trip the reactor and perform EOP-00, "Standard Post Trip Actions" then enter EOP-01, "Reactor Trip Recovery."

Question # 91 Rev 0

KA #: 014000 2.4.08 Tier 2 Group 2 ; Rod Position Indication System
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
Importance 3.8 / 4.5

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
Use the CEA and Control System Malfunctions Procedure to mitigate the consequences of a malfunction of a CEA, the CEA control system or CEA position indication.

Question Pedigree
New question.

K/A Fit:
Question addresses transition from an AOP to an EOP during a dropped CEA event.

Choice A:
Distractor: Plausible because this is the action in AOP-02 for one dropped CEA. Incorrect because dropping two CEA;s requires a reactor trip.

Choice B:
Distractor: Plausible because this is the action in AOP-02 after power is lowered to 70%.

Choice C:
Distractor: Plausible because AOP-02 directs a reactor trip for two dropped CEAs. Incorrect because AOP-02 says "GO TO" EOP-00 and all trippable rods will be fully inserted after the trip.

Choice D:
Correct answer: AOP-02 directs the operators to trip the reactor and go to EOP-00 if more than one CEA is misaligned greater than 18 inches from any other CEA in its group.

KA#:	014000 2.4.08	Bank Ref #:	N/A
LP# / Objective:	0717-02 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	TBD-AOP-02	Handout:	NONE

Section III - Misaligned Group 4 CEA

4.0 INSTRUCTIONS/CONTINGENCY ACTIONS

INSTRUCTIONS | CONTINGENCY ACTIONS

1. **IF ANY** of the following conditions exist:

   - A dropped CEA occurs during a Reactor startup

   - More than one CEA is misaligned greater than 18 inches from any other CEA in its group

   THEN initiate a Reactor shutdown by performing the following steps:

   a. Trip the Reactor.

   b. GO TO EOP-00, Standard Post Trip Actions.

The requirement to trip the Reactor if a CEA drops during startup complies with Operating Procedures. The requirement to trip the Reactor if more than one CEA is misaligned >18" is in compliance with PED-FC-89-2482 [R1].

---

2. Stop all CEA movement.

All CEA movement is stopped to preclude any further power peaking and CEA movement during any power reduction.

---

3. Adjust Turbine load to match Reactor power.

4. Establish steady Reactor power.

5. Ensure RCS pressure is 2075 to 2150 psia.

QUESTION NUMBER: 092

The plant was operating at 50% power due to condensate pumps FW-2A and FW-2C being out of service when the following sequence of events occurred:

- The running feedwater pump tripped and the standby feedwater pumps could not be started
- The reactor tripped on low S/G level
- Auxiliary feedwater pumps, FW-6, FW-10 and FW-54 could not be started
- EOP-00, “Standard Post Trip Actions,” were completed and EOP-06, “Loss of All Feedwater,” was entered
- S/G pressure was reduced to less than 550 psia and condensate pump, FW-2B, was used to feed the S/G’s
- Condensate pump, FW-2B, tripped and could not be restarted.

Assuming no further equipment failures occur, what actions should be taken as a result of these events?

A.✔ Stay in EOP-06, "Loss of All Feedwater," and initiate once-through-cooling when required.

B. Enter AOP-28, "Auxiliary Feedwater System Malfunctions," and establish flow to the S/G's from the Blair Water System.

C. Enter EOP-20, "Functional Recovery Procedure," and establish flow to the S/G's using demineralized water.

D. Enter OCAG-1, "Operational Contingency Action Guidelines," and establish flow to the S/G's from the FCS fire truck.

Question # 92 Rev 0

KA #: 056000 A2.04 Tier 2 Group 2: Condensate System
Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of condensate pumps
Importance 2.6 / 2.8

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
DEMONSTRATE the knowledge required to use EOP-06, Loss of All Feedwater (LOAF), to mitigate the consequences of a loss of all feedwater.

Question Pedigree
New question.

K/A Fit:
Question addresses using procedures to mitigate the loss of a condensate pump.

Choice A:
Correct answer: EOP-06 contains instruction for initiating once-through-cooling.

Choice B:
Distractor: Plausible because all AFW has been lost and Blair Water can be used to feed the emergency feedwater storage tank. Incorrect because AOP-28 does not address a loss of all feedwater and condensate.

Choice C:
Distractor: Plausible because EOP-20 does contain directions for feeding the S/G's using demineralized water. Incorrect because this would only be performed if Once-through-cooling failed.

Choice D:
Distractor: Plausible because OCAG-1 does establish flow to the S/G's using the fire truck. Incorrect because this event is mitigated by the EOPs.

KA#:	056000 A2.04	Bank Ref #:	N/A
LP# / Objective:	0718-16 01.00	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-06	Handout:	NONE

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. Depressurization of S/Gs to establish a low pressure Feedwater source will result in cooldown of the RCS and the potential for SGLS actuation.
2. RCS pressure and inventory control will be affected and SGIS may occur if SGLS is not blocked as the depressurization proceeds.
3. When T_C is 178°F or greater, the maximum RCS cooldown rate is 100°F/hr. When T_C is less than 178°F, the maximum RCS cooldown rate is 50°F/hr.

****

✘14. **IF** any Condensate Pumps are operating
AND a flow path to at least one S/G is available,
THEN perform the following steps to maintain S/G levels 35-85% NR (73-94% WR):

a. Place all of the Feed Pumps Control Switches, FW-4A/B/C, in "PULL-TO-LOCK".

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

✘14. (continued)

b. Locally open **ALL** of the following Feed Pump Discharge Valves (North wall, Turbine Building Basement):

- "DISCH VALVE HCV-1150A"
- "DISCH VALVE HCV-1150B"
- "DISCH VALVE HCV-1150C"

c. Ensure all of the Steam Generator Feed Pump Recirc valves, FCV-1151A/B/C are closed.

d. Start all of the Feed Pump Lube Oil Pumps, FW-30A/B/C.

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

✘ 14. (continued)

e. Reduce S/G pressure to less than 550 psia using **ANY** of the following methods:

- Steam Dump and Bypass Valves
- HCV-1040, Atmospheric Dump Valve
- MS-291 and MS-292, Air Assisted Main Steam Safety Valves

f. Maintain PZR level 10-70% by control of Charging and Letdown.

g. Maintain RCS pressure PER Attachment 2, RCS Pressure-Temperature Limits, by control of PZR Heaters and Spray.

h. Locally ensure FCV-1172, Condensate Pump Recirc Valve, is closed (Turbine Building Mezzanine).

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

✘14. (continued)

i. **WHEN** S/G pressure is less than 550 psia,
THEN ensure SGLS is blocked by performing the following steps:

1) Block SGLS-A and SGLS-B.

2) Verify at least one of the following SGLS Blocked alarms annunciates (CB-4; A8):

- "SGLS "A" BLOCKED"
- "SGLS "B" BLOCKED"

2).1 **IF** neither SGLS Blocked alarm annunciates,
THEN continue attempts to block SGLS until at least one alarm annunciates.

j. Ensure **BOTH** of the Feed Header Isolation Valves are open:

- HCV-1386
- HCV-1385

k. Ensure **BOTH** of the Feed Reg Block Valves are closed:

- HCV-1103
- HCV-1104

(continue)

INSTRUCTIONS | CONTINGENCY ACTIONS

✘14. (continued)

l. Reduce secondary pressure to achieve feed flow via **BOTH** of the Feed Reg Bypass Valves:

- HCV-1105
- HCV-1106

Time: ________

m. Continue efforts to restore Main or Auxiliary Feedwater.

✘15. Verify adequate RCS heat removal via the S/G(s) by **BOTH** of the following conditions:

- At least one S/G has wide range level greater than or equal to 27%
- RCS T_C temperatures are stable or lowering

(continue)

15.1 **IF ANY** of the following criteria are satisfied:

- The least affected S/G wide range level less than 27%
- An uncontrolled rise in RCS T_C of greater than 5°F

THEN perform step a or b:

a. **IF BOTH** Vital 4160V buses are energized,
THEN perform the following:

1) Stop all RCPs

2) Deenergize all PZR Heaters.

(continue)

INSTRUCTIONS	CONTINGENCY ACTIONS
✘ 15. (continued)	15.1.a (continued)
	3) Initiate PPLS by placing **BOTH** of the following switches in "TEST": • 86A/PPLS TEST SWITCH • 86B/PPLS TEST SWITCH
	4) Ensure two HPSI pumps SI-2A/B or SI-2B/C start.
(continue)	5) Ensure all of the HPSI Loop Injection Valves are open.
	6) Verify all Charging Pumps CH-1A/B/C start.
	7) Ensure **BOTH** of the PORV Block Valves are open: • HCV-150 • HCV-151
	8) Open the PORVs. Time: ___________
	(continue)

QUESTION NUMBER: 093

Given the following plant conditions:

- A LOCA occurred inside containment
- EOP-03, “Loss of Coolant Accident,” was entered after completion of EOP-00 standard post trip actions
- All Reactor Coolant Pumps have been tripped
- All charging pumps, HPSI Pumps SI-2A and SI-2B and LPSI Pumps SI-1A and SI-1B are all operating
- Reactor Vessel Level is 0%
- SI flow is less than Attachment 3, Safety Injection Flow vs. Pressurizer Pressure
- The countrates on all of the source range NI's are rising
- WR NIs indicate 4 x 10^{-7}% power
- A RCS cooldown is in progress
- RCS Pressure is 600 psia
- CETs indicate 510°F
- RAS has NOT occurred

What action should be taken in response to these conditions?

A. Enter OI-RC-12, "Post Accident Venting of Non-Condensable Gases from the Reactor Coolant System," and vent RV head to Containment.

B. Enter OI-RC-12, "Post Accident Venting of Non-Condensable Gases from the Reactor Coolant System," and vent RV head to the Pressurizer Quench Tank

C. Enter EOP-20, RC-1,"Reactivity Control," and insert the group "N" CEAs

D.✓ Enter EOP-20, IC-2, "RCS Inventory Control" and start HPSI Pump, SI-2C

QUESTIONS REPORT

for 2012-2 FCS NRC WRITTEN EXAM Rev 0

Question # 93 Rev 0

KA #: 015000 A2.05 Tier 2 Group 2: Nuclear Instrumentation System

Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b)based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Core void formation
Importance 3.3 / 3.8

CFR Number: 55.43(b)(5)

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:

Explain the effects on the WR NIS with voiding in the core.

Question Pedigree

New question.

K/A Fit:

Question address procedural action to mitigate a core uncovery event that is affecting nuclear instrumentation.

Choice A:

Distractor:Plausible because core uncovery has occured and non-condensible gases have most likely formed. Incorrect because EOPs direct starting another HPSI pump.

Choice B:

Distractor:Plausible because core uncovery has occured and non-condensible gases have most likely formed. Incorrect because EOPs direct starting another HPSI pump.

Choice C:

Distractor: Plausible because source range NI's show rising counts. Incorrect because WR NI's are less that 10^{-5} % power.

Choice D:

Correct answer: EOP-20, IC-2 is entered because the Inventory Control Safety Function is not met with reactor vessel level at 0%. EOP-20 directs starting all idle pumps if SI flow is not maximized. SI flow is less than required by attachment 3.

KA#:	015000 A2.05	Bank Ref #:	N/A
LP# / Objective:	0712-18 02.10	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-03	Handout:	NONE

14.0 RCS INVENTORY CONTROL

SAFETY FUNCTION: RCS Inventory Control

SUCCESS PATH: Safety Injection: IC-2

RESOURCE TREE: Tree C

IC-2

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTIONS

1. LOCAs inside Containment can greatly raise instrument inaccuracies. A single indication may be unreliable. All available indications and actual plant response should be used to evaluate actions and plant conditions. TSC staff should assist in evaluating these uncertainties through use of the Electrical Equipment Qualification Manual.

2. Do not allow Diesel Generator loads to exceed power and current rating limits.

****

✘1. **IF** RCS pressure is less than or equal to 1600 psia, **THEN** verify Engineered Safeguards are actuated by performing the following:

a. Ensure Emergency Boration is in progress.

b. Ensure SI flow is acceptable PER Attachment 3, Safety Injection Flow vs. Pressurizer Pressure.

14.0 RCS INVENTORY CONTROL

IC-2

INSTRUCTIONS | CONTINGENCY ACTIONS

****

CAUTION

Do not allow Diesel Generator loads to exceed power and current rating limits.

****

✘3. **IF** SIAS has actuated, **THEN** maximize Safety Injection flow and Charging flow to the RCS by operating **ALL** of the following available pumps:

- Either HPSI Pumps, SI-2A/B or SI-2B/C
- LPSI Pumps, SI-1A/B
- Charging Pumps, CH-1A/B/C

(continue)

3.1 **IF** Safety Injection and Charging flow is **NOT** maximized, **THEN** restore Safety Injection and Charging flow by performing the following:

a. Restore electrical power to valves and pumps.

b. Ensure **ALL** o. the following SI valves are open to align SI for injection:

- HPSI Loop Injection Valves
- LPSI Loop Injection Valves
- HPSI Discharge Cross-Connect Valves
- HPSI Header Isolation Valves
- SI Pump Suction SIRWT Isolation Valves

(continue)

14.0 RCS INVENTORY CONTROL

IC-2

INSTRUCTIONS	CONTINGENCY ACTIONS
✘3. (continued)	3.1 (continued) c. Start **ALL** of the following idle pumps: • HPSI Pumps, SI-2A/B/C • LPSI Pumps, SI-1A/B • Charging Pumps, CH-1A/B/C d. Verify SI flow is acceptable PER Attachment 3, Safety Injection Flow vs. Pressurizer Pressure.
✘4. **IF** high RCS pressure is preventing adequate SI flow, **THEN** depressurize the RCS to less than 1200 psia by performing **ANY** of the following: a. Control RCS heat removal. b. Control PZR Heaters. c. Control PZR Main or Auxiliary Spray. d. Operate the PORVs.	

QUESTION NUMBER: 094

With the plant in hot shutdown, the Shift Technical Advisor slipped on a stairway and injured his leg 2 hours into the shift. An EMT was called and the STA was transported to the Blair Hospital. The CRS is also STA qualified.

What action should be taken due to this event?

A. No action is required.

B.✔ Have a qualified STA on-site within two hours.

C. Have a qualified STA on-site within four hours.

D. Have the CRS fill the STA position for the remainder of the shift.

Question # 94 Rev 0

KA #: 000000 2.1.05 Tier 3 Group 4: Generic Knowledges and Abilities
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc..
Importance 2.9 / 3.9

CFR Number: 55.43(b)(1)
Conditions and limitations in the facility license.

Fort Calhoun Objective:
Operations and STA staffing requirements (SRO only)

Question Pedigree
Bank question used on the 1997 exam with slight rewording. Not counted as a modified question.

K/A Fit:
Question addresses shift staffing requirements per Technical Specifications.

Choice A:
Distractor:Plausible because reactor is shutdown. Incorrect because a STA is required in hot shutdown.

Choice B:
Correct answer: A STA is required in hot shutdown and there is a 2 hour time period.

Choice C:
Distractor: Plausible because a STA is required. Incorrect because the time period is 2 hours.

Choice D:
Distractor: Plausible because CRS is STA qualified. Incorrect because CRS can not fill two positions.

KA#:	000000 2.1.05	Bank Ref #:	ADM-OPS 008
LP# / Objective:	0762-08 10.01	Exam Level:	SRO-1
Cognitive Level:	LOW	Source:	BANK
Reference:	TS TABLE 5.2-1	Handout:	NONE

TABLE 5.2-1

MINIMUM SHIFT CREW COMPOSITION[ii]

License Category	**Core Alteration**	**Cold Shutdown or Refueling Shutdown**	**Operating or Hot Shutdown Modes**
Senior Operator License	2[i]	1	2[iii]
Operator License	2	1	2[iv]
Non-Licensed	(As required)	1	2
Shift Technical Advisor	None	None	1

(i) This includes the individual with Senior Operator License supervising Core Alterations.

(ii) Shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 5.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewmember being late or absent.

(iii) At least one of these individuals must be in the control room at all times.

(iv) At least one of these individuals (or the second senior licensed operator, if both senior licensed operators are in the control room) must be present at the controls at all times.

QUESTIONS REPORT
for ILO EXAM BANK

ADM-OPS 008

With the plant in hot shutdown, the Shift Technical Advisor slipped on a stairway and injured his ankle. An off-duty Shift Manager leaving the site has taken him to the emergency room. Which of the following actions should be taken due to this event?

A. Make a four hour report to the NRC.

B. Have a System Engineer who is training to be an STA fill the position.

C.✓ Have a qualified STA on-site within two hours.

D. Declare a Notification of Unusual Event based on transport of an injured person.

QUESTION NUMBER: 095

The performance of SE-PM-AE-1008, "Inspection of Station Blackout (SBO) Restoration Structural Components," is being planned. Access to the switchyard will be required. What is the Shift Manager's responsibility for pre-planning this work?

A. Inform the Security Shift Supervisor to provide an escort for the personnel performing the inspection while they are in the switchyard per FCSG-32, "Work Week Management."

B. Determine if proposed work can be accomplished without adversely affecting plant operation per FCSG-32, "Work Week Management."

C. Inform the Security Shift Supervisor to provide an escort for the personnel performing the inspection while they are in the switchyard per NOD-QP-36, "Control of Switchyard Activities at Fort Calhoun Station"?

D.✓ Determine if proposed work can be accomplished without adversely affecting plant operation per NOD-QP-36, "Control of Switchyard Activities at Fort Calhoun Station"?

Question # 95 Rev 0

KA #: 000000 2.1.13 Tier 3 Group 4
Generic Knowledges and Abilities
Knowledge of facility requirements for controlling vital / controlled access.
Importance 2.5 / 3.2

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective
Given a copy of the procedure NOD-QP-36, Control of SWYD Activities at FCS, the student will be able to perform the following: Describe the responsibilities associated with NOD-QP-36.

Question Pedigree
New question.

K/A Fit:
Question addresses SRO responsibility for access to the switchyard.

Choice A:
Distractor: Plausible if Applicant believes security must supply escort. Incorrect because the Shift Manager does not inform the Security Shift Supervisor and switchyard access is not controlled by FCSG-32.

Choice B:
Distractor: Plausible because this the Shift Managers responsibility. Incorrect because switchyard access in not controlled by FCSG-32.

Choice C:
Distractor: Plausible because NOD-QP-36 is used to control switchyard access. Incorrect because the Shift Manager does not inform the Security Shift Supervisor.

Choice D:
Correct answer: The Shift Manager determines if proposed work can be accomplished without adversely affecting plant operation per NOD-QP-36, "Control of Switchyard Activities at Fort Calhoun Station"

KA#:	000000 2.1.13	Bank Ref #:	ADM-CONTROL 017
LP# / Objective:	0713-01 01.06C	Exam Level:	SRO-5
Cognitive Level:	LOW	Source:	NEW
Reference:	NOD-QP-36	Handout:	NONE

4.3.1 (continued)

F. The Shift Manager will:

1. Determine if the activity can be accomplished without adversely impacting plant operation.

2. Ensure conditions have not changed since initial draft of the activity.

3. Review and give final approval to the substation activity.

G. The Transmission Operator will notify the Shift Manager just prior to commencing the stated activity to obtain final approval.

4.3.2 Emergent activities affecting the availability of FCS substations circuits

A. The T&D System Scheduler or Transmission Operator will notify the Shift Manager at least one day in advance, if possible, via OPPD normal telephone lines (preferred) or the emergency radio prior to performing any manipulations affecting the FCS Switchyard.

B. The Shift Manager will contact the Systems Analysis Department, System Engineer and the Work Week Manager for assistance in evaluating and resolving any schedule or plant risk conflicts if the grid is in a degraded condition. Reference Standing Order SO-M-100 for additional risk assessment information.

C. The Transmission Operator will notify the Shift Manager just prior to commencing the stated activity to obtain final approval.

4.3.3 Emergency Activities affecting the availability of FCS Switchyard circuits

A. The Transmission Operator will notify the Shift Manager, as soon as practical, when unanticipated conditions require any manipulation of circuits affecting FCS Switchyard circuits.

B. The Shift Manager will evaluate out-of-service safety related equipment and determine the need to expedite the return of equipment that may have an impact on the plant risk assessment. The Shift Manager will continue to monitor conditions and will direct the necessary actions to maintain FCS in a safe condition. Transmission Operator will provide high priority support in the event of a failure that results in a loss of operability of FCS house service (T1A-3 and T1A-4) or auxiliary transformers (T1A-1 and T1A-2) or the loss of transmission line supplies (Circuits 1587 and 3423) to these transformers. If these are not energized within 72 hours, a plant shutdown is required.

QUESTION NUMBER: 096

The performance of RE-ST-RX-0003, "Moderator Temperature Coefficient Determination Using Center CEA," requires the department manager or above to attend the pre-job briefing.

Where is that requirement found?

A. SO-G-3, "Special Procedures"

B. SO-G-23, Surveillance Test Program"

C.✓ SO-G-92, "Conduct of Infrequently Performed Procedures"

D. SO-G-117, "Reactivity Management"

Question # 96 Rev 0

KA #: 000000 2.2.07 Tier 3 Group 4:Generic Knowledges and Abilities
Knowledge of the process for conducting special or infrequent tests.
Importance 2.9 / 3.6

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
STATE some of the activities, covered by Standing Orders, which require written procedures per Regulatory Guide 1.33.

Question Pedigree
New question.

K/A Fit:
Question addresses the process for conducting an infrequent test.

Choice A:
Distractor: Plausible if the Applicant believes this is covered by SO-G-3. Incorrect because the requirement is found in SO-G-92.

Choice B:
Distractor: Plausible if the Applicant believes this is covered by SO-G-23. Incorrect because the requirement is found in SO-G-92.

Choice C:
Correct answer. This requirement is found in SO-G-92.

Choice D:
Distractor: Plausible if the Applicant believes this is covered by SO-G-117. Incorrect because the requirement is found in SO-G-92.

KA#:	000000 2.2.07	Bank Ref #:	N/A
LP# / Objective:	0762-01 02.00	Exam Level:	SRO-5
Cognitive Level:	LOW	Source:	NEW
Reference:	SO-G-92	Handout:	NONE

Attachment 1 - Infrequently Performed Procedures

NUMBER	DESCRIPTION
MD-RR-MX-1002	Establishing and Maintaining Freeze Seals by Vendor Personnel
MD-RR-MX-1003	Establishing and Maintaining Freeze Seals on Stainless Steel Piping
MD-RR-MX-1004	Establishing and Maintaining Freeze Seals on Carbon Steel Piping
MD-RR-MX-1005	Establishing and Maintaining Freeze Seals Using Portable Refrigerant Units
MM-RR-CEDM-1201	CEDM Coupling (Note 1)
OI-CC-1	CCW System Normal Operations; any operation or maintenance activity that directs removal of CCW system overpressure
OI-RC-2A	RCS Fill and Drain Operations (Lowered/Reduced Inventory Operations Only) (Note 1)
OI-SFP-7	Temporary Spent Fuel Pool Cooling via Chillers (Note 1)
OP-2A	Plant Startup (Approach to Critical Operation Only) (Note 1)
OP-ST-ESF-0002	Diesel Generator No. 1 and No. 2 Auto Operation (Note 1)
OP-ST-RC-3004	Power Operated Relief Valves (PORVs) Low Temperature Low Pressure Exercise Test (PCV-102-1 and PCV-102-2) (Note 1)
PE-HT-MX-1000	Hydrostatic Testing of Safety-Related Plant System and Components
PE-OT-SFP-0001	Temporary Spent Fuel Cooling System Installation/Removal (Note 1)
RE-CPT-RX-0001	Post Refueling Core Physics Testing and Power Ascension (Note 1)
RE-RR-DFS-0001	DSC/TC Prep for Fuel Loading Operations
RE-RR-FE-0001	Reconstitution of Irradiated Fuel Assemblies (Note 1)
RE-RR-FE-0002	Modification of Fuel Assemblies (Note 1)
RE-ST-RX-0003	Moderator Temperature Coefficient Determination Using Center CEA (Note 1)
RP-208	Radiography
RP-212	Diving Operations Within Radiologically Controlled Areas
SE-ST-ILRT-0001	Containment Integrated Leak Rate Test (ILRT)

Note 1: Department Manager or above must attend activity pre-job brief [SOER 10-2].

QUESTION NUMBER: 097

With the reactor operating at full power, an engineering evaluation determined that several pipe support snubbers in both trains of the emergency core cooling system are unable to perform their associated support function.

What action is required by Technical Specifications?

A. Place the plant in hot shutdown within 6 hours.

B.✔ Restore the snubbers to operable status within 12 hours or place the plant in hot shutdown within an additional 6 hours.

C. Restore the snubbers to operable status within 24 hours or place the plant in hot shutdown within an additional 6 hours.

D. Restore the snubbers to operable status within 72 hours or place the plant in hot shutdown within an additional 6 hours.

Question # 97 Rev 0

KA #: 000000 2.2.22 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of limiting conditions for operations and safety limits.
Importance 4.0 / 4.7

CFR Number: 55.43(b)(2)
Facility operating limitations in the technical specifications and their bases.

Fort Calhoun Objective:
Given a copy of Technical Specifications, APPLY the requirements to a given condition covered by an LCO.

Question Pedigree
New question.

K/A Fit:
Question address LCO that applies to numerous plant systems.

Choice A:
Distractor: Plausible if Applicant does not realize Tech Specs allow 12 hours to restore snubbers to operable. Incorrect because we have 12 hours.

Choice B:
Correct answer: Tech Spec 2.0.1 requires that we restore the snubbers to operable status within 12 hours or place the plant in hot shutdown within an additional 6 hours.

Choice C:
Distractor: Plausible if Applicant does not know the time limit. Incorrect because we have 12 hours.

Choice D:
Distractor: Plausible because this would be correct if only one train was affected. Incorrect because snubbers in both trains are affected.

KA#:	000000 2.2.22	Bank Ref #:	N/A
LP# / Objective:	0762-08 05.00	Exam Level:	SRO-2
Cognitive Level:	LOW	Source:	NEW
Reference:	TS 2.0.1	Handout:	NONE

2.0 **LIMITING CONDITIONS FOR OPERATION**

2.0.1 General Requirements

Applicability

Applies to the operable status of all systems, subsystems, trains, components, or devices covered by the Limiting Conditions for Operation.

Objective

To specify corrective measures to be employed for system conditions not covered by or in excess of the Limiting Conditions for Operation.

Specification

(1) In the event a Limiting Condition for Operation and/or associated action requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT SHUTDOWN within 6 hours, in at least subcritical and < 300°F within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours, unless corrective measures are completed that permit operation under the permissible action requirements for the specified time interval as measured from initial discovery or until the reactor is placed in an Operating Mode in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

(2) When one or more required snubbers are unable to perform their associated support function(s), any affected supported LCO(s) are not required to be declared not met solely for this reason if risk is assessed and managed, and:

a. the snubbers not able to perform their associated support function(s) are associated with only one train or subsystem of a multiple train or subsystem supported system or are associated with a single train or subsystem supported system and are able to perform their associated support function within 72 hours; or

b. the snubbers not able to perform their associated support function(s) are associated with more than one train or subsystem of a multiple train or subsystem supported system and are able to perform their associated support function within 12 hours.

At the end of the specified period the required snubbers must be able to perform their associated support function(s), or the affected supported system LCO(s) shall be declared not met.

QUESTION NUMBER: 098

Given the following conditions:
- Spent Fuel is being moved from the Spent Fuel Pool to the Independent Spent Fuel Storage Installation (ISFSI)
- A Transfer Cask/Dry Shielded Canister (TC/DSC) is being removed from the Spent Fuel Pool and transferred to the work platform.
- Anticipated radiation levels are 1200 mrem/hr at 30 cm from the radiation source

How will the Spent Fuel Pool Area be posted and access controlled during this operation?

A. The area will be posted as a Radiation Area; available gates and doors to the area will be locked.

B. The area will be posted as a Radiation Area; available gates and doors to the area will not be locked.

C.✓ The area will be posted as a Restricted High Radiation Area; available gates and doors to the area will be locked.

D. The area will be posted as a Restricted High Radiation Area; available gates and doors to the area will not be locked.

Question # 98 Rev 0

KA #: 000000 2.3.12 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
Importance 3.2 / 3.7

CFR Number: 55.43(b)(4)
Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Fort Calhoun Objective:
DEFINE a restricted high radiation area and LIST the posting requirements.

Question Pedigree
Bank question used on the 2009 NRC exam.

K/A Fit:
Question addresses access to a restricted high radiation area during spent fuel transfer.

Choice A:
Distractor: Plausible if area was posted as a Radiation Area, Incorrect because it is posted as a Restricted High Radiation Area.

Choice B:
Distractor: Plausible if area was posted as a Radiation Area, Incorrect because it is posted as a Restricted High Radiation Area.

Choice C:
Correct answer: RP-222 states that the area will be posted as a RHRA and doors and gates will be locked.

Choice D:
Distractor: Plausible because it is posted as a Restricted High Radiation Area. Incorrect because access controls to a RHRA include locking doors and gates.

KA#:	000000 2.3.12	Bank Ref #:	19-24-03 003
LP# / Objective:	1924-03 01.06	Exam Level:	SRO-4
Cognitive Level:	HIGH	Source:	BANK
Reference:	RP-222	Handout:	NONE

7.15 Area postings and controls:

7.15.1 If conditions warrant access shall be controlled as a RHRA when the TC/DSC is being removed from the spent fuel pool and transferred to the work platform or moving from work platform to transport trailer in railroad siding by using combination of one or more of the following:

7.15.1.A Lock or guard and post the gate from room 69 to the SFP.

7.15.1.B Lock or guard and post the doors in Rooms 25A, 67 and 68.

7.15.1.C Station RHRA guards in the BAST Area and at the Stairs to Room 69.

7.15.1.D Suspend all work in room 69 and the BAST area to RRS hallway.

7.15.2 After the TC/DSC is placed in room 68, RP will re-survey the area and update postings accordingly.

7.15.3 After the following steps have been completed:

- The DSC is inside room 68
- Up to 750 gallons Water Drained from DSC
- TC/DSC annulus full (within approximately 1 foot of top)
- TC neutron shield full
- Top Shield Plug in place and included in axial shielding
- Inner Top Cover Plate in place and included in axial shielding
- Automated Welding System (AWS) with integral shield in place and included in axial shielding

Obtain a gamma and neutron dose rate at the vertical centerline of DSC, 3 feet and 30 cm from AWS shield. Ensure the sum of the dose rates obtained does <u>not</u> exceed **200 mrem/hr at 3 feet**.

7.15.4 DSC Removable Surface Contamination Limits

7.15.4.A.1) 2,200 dpm/100 cm^2 for beta-gamma sources

7.15.4.A.2) 220 dpm/100 cm^2 for alpha sources

2.12 Posting - A conspicuous sign bearing a magenta or black radiation symbol on a yellow background and the words "CAUTION", or "DANGER" or "GRAVE DANGER" as described in 10CFR20, which is used to post areas or rooms due to radiation or radioactive materials. Posting shall identify the presence of radiation or radioactive materials. Extra information which may aid individuals in minimizing their exposure to radiation or radioactive materials may also be provided.

2.13 Radiation Area - An Area, accessible to individuals, in which radiation levels could result in an individual receiving a deep dose equivalent in excess of 0.005 rem (0.05 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

2.14 Radiologically Controlled Area - An area posted and controlled for the purpose of protecting personnel from radiation or radioactive materials. Access to any RCA requires a Radiation Work Permit.

2.15 Restricted Area - An area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. At FCS this area is normally established by the security fence and gate house.

2.16 Restricted High Radiation Area - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving a deep dose equivalent in excess of 1 rem (10 mSv) in one hour at 30 centimeters from the radiation source or from any surface that the radiation penetrates.

2.17 Step-Off Pad (SOP) - A delineated area on a surface outside the boundary of a Contaminated or Highly Contaminated Area, which establishes access/egress points for such areas.

2.18 Temporarily Accessible - For the purpose of identifying, demarcating and surveying, temporarily accessible is defined to be any location that is greater than eight feet off the floor and is in such a location that an individual can, access within 30 centimeters, for less than seven calendar days (via a ladder, scaffolding or climbing on pipes).

2.19 Very High Radiation Area - An area, accessible to individuals, in which radiation levels from radiation sources external to the body could result in an individual receiving an absorbed deep dose in excess of 500 rads (5 grays) in one hour at 1 meter from a radiation source or from any surface that the radiation penetrates.

2.20 Whole Body - For purposes of external exposure, head, trunk (including male gonads), arms above elbow, or legs above the knee.

3.0 RESPONSIBILITIES

3.1 The Manager-Radiation Protection is responsible for:

3.1.1 Enforcing compliance with radiological postings by all personnel at FCS.

QUESTION NUMBER: 099

A large steam line break inside containment occurred on steam generator RC-2A coincident with a 50 gpm tube rupture on steam generator RC-2B.

What action should be taken after EOP-00, "Standard Post Trip Actions have been completed?

A. Enter EOP-04, "Steam Generator Tube Rupture," and isolate S/G RC-2B

B. Enter EOP-05, "Uncontrolled Heat Extraction," and isolate S/G RC-2A

C.✓ Enter EOP-20, "Functional Recovery Procedure," and isolate S/G RC-2A

D. Enter EOP-20, "Functional Recovery Procedure," and isolate S/G RC-2B

Question # 99 Rev 0

KA #: 000000 2.4.06 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge EOP mitigation strategies.
Importance 3.7 / 4.7

CFR Number: 55.43(b)(5)
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal,and emergency situations.

Fort Calhoun Objective:
EXPLAIN how an Excessive Heat Removal Event in conjunction with another event would be handled by the operators.

Question Pedigree
New question.

K/A Fit:
Question addresses EOP mitigation strategies for multiple simultaneous events.

Choice A:
Distractor: Plausible because a tube rupture has occurred. Incorrect because EOP-20 is used to mitigate simultaneous events.

Choice B:
Distractor: Plausible because an uncontrolled heat extraction has occurred. Incorrect because EOP-20 is used to mitigate simultaneous events.

Choice C:
Correct answer: EOP-20 is entered for multiple events. The most affected steam generator is isolated.

Choice D:
Distractor: Plausible because EOP-20 is entered for multiple events. Incorrect because RC-2A, the most affected S/G, should be isolated.

KA#:	000000 2.4.06	Bank Ref #:	07-15-20 022
LP# / Objective:	0715-20 02.07	Exam Level:	SRO-5
Cognitive Level:	HIGH	Source:	NEW
Reference:	EOP-20	Handout:	NONE

16.0 RCS AND CORE HEAT REMOVAL

HR-3

INSTRUCTIONS	CONTINGENCY ACTIONS
✘10. Determine the S/G with the Tube Rupture by performing the following:	
a. Review the S/G analysis for activity.	
b. Monitor Steam Line radiation levels.	
c. Monitor S/G Blowdown radiation levels.	
d. Monitor S/G levels.	
✘11. **WHEN** RCS T_H is less than or equal to 510°F, **THEN** isolate the most affected S/G by performing the following:	
a. **IF** RC-2A is most affected, **THEN** isolate RC-2A by performing the following:	a.1 **IF** RC-2B is most affected, **THEN** isolate RC-2B by performing the following:
(continue)	(continue)

QUESTION NUMBER: 100

A General Emergency has been declared at Fort Calhoun Station. The TSC and EOF are both fully staffed and operational. The Shift Manager is in the Command and Control position at this time.

Which of the following duties can the Shift Manager delegate to the CRS?

A. Ensuring that the event is classified properly.

B. Ensuring appropriate Protective Action Recommendations are provided to offsite officials.

C. Authorizing the issuance of Potassium Iodide to OPPD emergency workers.

D.✓ Authorizing deviations from Technical Specifications needed to mitigate the event.

Question # 100 Rev 0

KA #: 000000 2.4.40 Tier 3 Group 4: Generic Knowledges and Abilities
Knowledge of SRO responsibilities in emergency plan implementation.
Importance 2.7 / 4.5

CFR Number: 55.43(b)(1)
Conditions and limitations in the facility license.

Fort Calhoun Objective
IDENTIFY the ERO positions that can have overall Command and Control.

Question Pedigree
Bank question used on the 2005 NRC exam.

K/A Fit:
Question addresses SRO responsibilities during an emergency plan event.

Choice A:
Distractor: Plausible if Applicant does not know the Command and Control Responsibilities that can not be delegated. Incorrect because this responsibility can not be delegated.

Choice B:
Distractor: Plausible if Applicant does not know the Command and Control Responsibilities that can not be delegated. Incorrect because this responsibility can not be delegated.

Choice C:
Distractor: Plausible if Applicant does not know the Command and Control Responsibilities that can not be delegated. Incorrect because this responsibility can not be delegated.

Choice D:
Correct answer: This is not one of the responsibilities that can not be delegated.

KA#:	000000 2.4.40	Bank Ref #:	ADM-EP
LP# / Objective:	1070-101 01.13	Exam Level:	SRO-1
Cognitive Level:	LOW	Source:	BANK
Reference:	EPIP-OSC-2	Handout:	NONE

Attachment 6.8 - Command and Control Position Responsibilities

The following responsibilities **CAN NOT BE DELEGATED** by the Command and Control position. The responsibility of their completion rests with the Command and Control position until relieved by another qualified individual or the emergency is terminated. The Command and Control position may assign other personnel to assist in conducting the actions necessary.

1. Overall **COMMAND AND CONTROL** of the Emergency Response Organization.

2. Ensuring the proper **CLASSIFICATION AND DECLARATION** of the emergency situation is made in accordance with EPIP-OSC-1 and is periodically reviewed to determine if the classification should be upgraded, downgraded or terminated.

3. Ensuring all required **NOTIFICATIONS** are made to appropriate state, local and federal officials.

4. Ensuring any appropriate **PROTECTIVE ACTION RECOMMENDATIONS** (PARs) are provided to offsite officials.

5. Authorizing OPPD emergency worker exposure extensions beyond the Federal Radiation Protection Guidance.

6. Authorizing issuance of Potassium Iodide for OPPD emergency workers.

THE COMMAND AND CONTROL POSITION ALSO HAS THE FOLLOWING RESPONSIBILITIES WHICH MAY BE DELEGATED TO OTHER PERSONNEL, AS NECESSARY.

7. Request for assistance from federal agencies.

8. Authorizing any emergency information to be released to the media or the general public.

9. Coordinating the transfer of emergency information from the Emergency Response Organization (ERO) to other OPPD and outside organizations called upon to assist.

10. Ensuring a timely and complete turnover of information to any qualified relief.

11. Providing information to authorized representatives of the states of Nebraska, and Iowa, and associated local governments.

12. Ensuring plant operations are in compliance with Technical Specifications and other license conditions. If deviations are necessary to protect the public health and safety, they must be evaluated with respect to 10CFR50.54(x) and (y) and approved, as a minimum, by a senior licensed operator, prior to taking the action.