NAME:________________________ EMPLOYEE #______________

COURSE NO.__STP LOT 26 NRC Written Exam__

EXAM NO.__**KEY RO 1-75 / SRO 1-100**__

1	[A]	[B]	[C]	🟥
2	🟥	[B]	[C]	[D]
3	[A]	[B]	[C]	🟥
4	🟥	[B]	[C]	[D]
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100	[A]	[B]	[C]	🟥

CONFIDENTIAL
D56/D43

I signify that all answers in this examination are my own. I have not given or received any unauthorized assistance and I have not used any unauthorized references. I understand that I am not to remove any of the examination materials from this room.

_____________________ __________
Student Signature Date

I signify that I have had the opportunity to review my graded exam and am satisfied that I understand the correct answer for those questions I missed. This signature does not remove my right to appeal questions with the NRC per NUREG-1021.

_____________________ __________
Student Signature Date

KEY

Points Scored: ____________________

Points Possible:___________________

Grade %:_________________________

Exam Bank No.: 3118 **Last used on an NRC exam:** Never

RO Sequence Number: 1

Unit 1 is at 100% power.

- An automatic reactor trip signal fails to trip the reactor.
- The RO attempts to trip the reactor using both Reactor Trip Switches.
- The RO is directed in 0POP05-E0-EO00, Reactor Trip or Safety Injection to "OPEN 480V LC 1K1 and 1L1 feeder breakers".

This step will de-energize the ______(1)______.

___(2)___ feeder breaker(s) must open to trip the reactor.

A. (1) Reactor Trip Breakers shunt trip coils
(2) One

B. (1) Reactor Trip Breakers shunt trip coils
(2) Both

C. (1) Rod Drive MG Set Motors
(2) One

D. (1) Rod Drive MG Set Motors
(2) Both

Answer: D (1) Rod Drive MG Set Motors
(2) Both

Exam Bank No.: 3118 **Source:** Bank **Modified from**

K/A Catalog Number: EPE 007 EK2.02 Knowledge of the interrelations of a reactor trip and the following: Breakers, relays, and disconnects.

RO Importance: 2.6 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(6)

STP Lesson: LOT 505.05 **Objective Number:** 80843

Given a copy of a subsequent step or from memory an immediate operator action step from 0POP05-EO-EO00, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose, and result. LOT 201.18 (Obj. 30444) - List the power supplies for the major components of the Rod Control System.

Reference: LOT 201.18 slides 121, 127, 128; 0POP02-RS-0001, Rod Control, Lineup 1 pg 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the shunt trip coils will trip open the Reactor Trip Breakersif der-energized but are powered from a different source and only one opening will cause a Reactor trip.

B: INCORRECT. Plausible because the shunt trip coils will trip open the Reactor Trip Breakers is denergized but are powered from a different source..

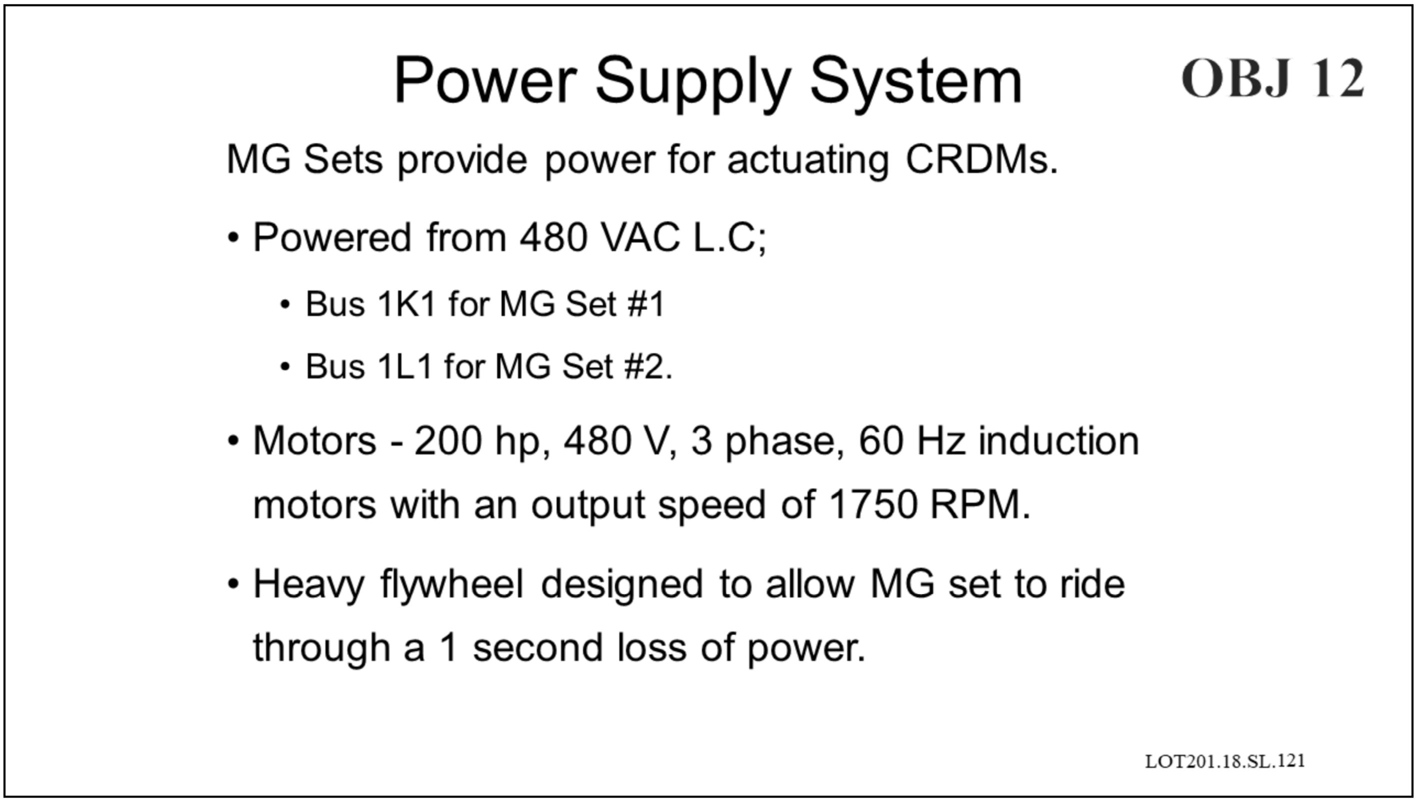
C: INCORRECT. Plausible because the Rod Drive MG Set Motors are de-energized by opening 1K1 and 1L1 loads centers; however, both must be de-energized to initiate a reactor trip.

D: CORRECT.1K1 and 1L1 load centers supply power to individual Rod Drive MG Set motors. Both are normally running. Either MG Set providing power to the Rod Control System will be sufficient to power the rod drives (design redundancy) so both MG Sets must be de-energized to trip the reactor.

Question Level: F **Question Difficulty** 3

Justification:

The student must know the interrelationship of the components in the power supply lineup to the reactor trip breakers.



Power Supply System

OBJ 12

MG Sets provide power for actuating CRDMs.

- Powered from 480 VAC L.C;
  - Bus 1K1 for MG Set #1
  - Bus 1L1 for MG Set #2.
- Motors - 200 hp, 480 V, 3 phase, 60 Hz induction motors with an output speed of 1750 RPM.
- Heavy flywheel designed to allow MG set to ride through a 1 second loss of power.

LOT201.18.SL.121

The system san withstand the loss of one power supplying MG set without adverse effects, as only one MG set is necessary to power the Rod Control System. However a power failure at both buses would trip the reactor.

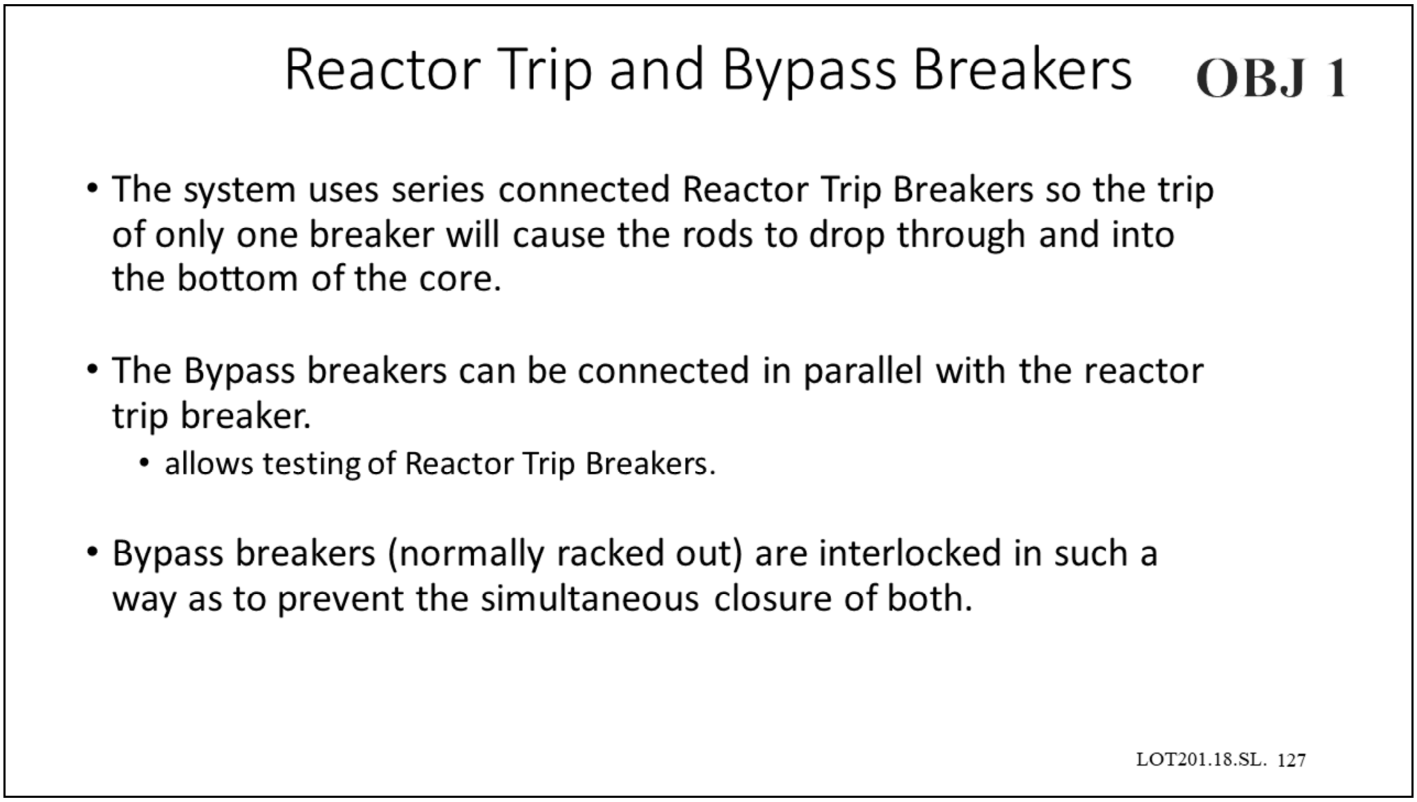
The Motor Generator Sets are located in the MG set room on the 60’ level of the Mechanical and Electrical Aux Bldg (west end).

Motor end - 200 hp, 480 v, 3 phase, 60 Hz, 1750 RPM

Induction motor with finite speed slip

Motor Supply breakers are at the buses.

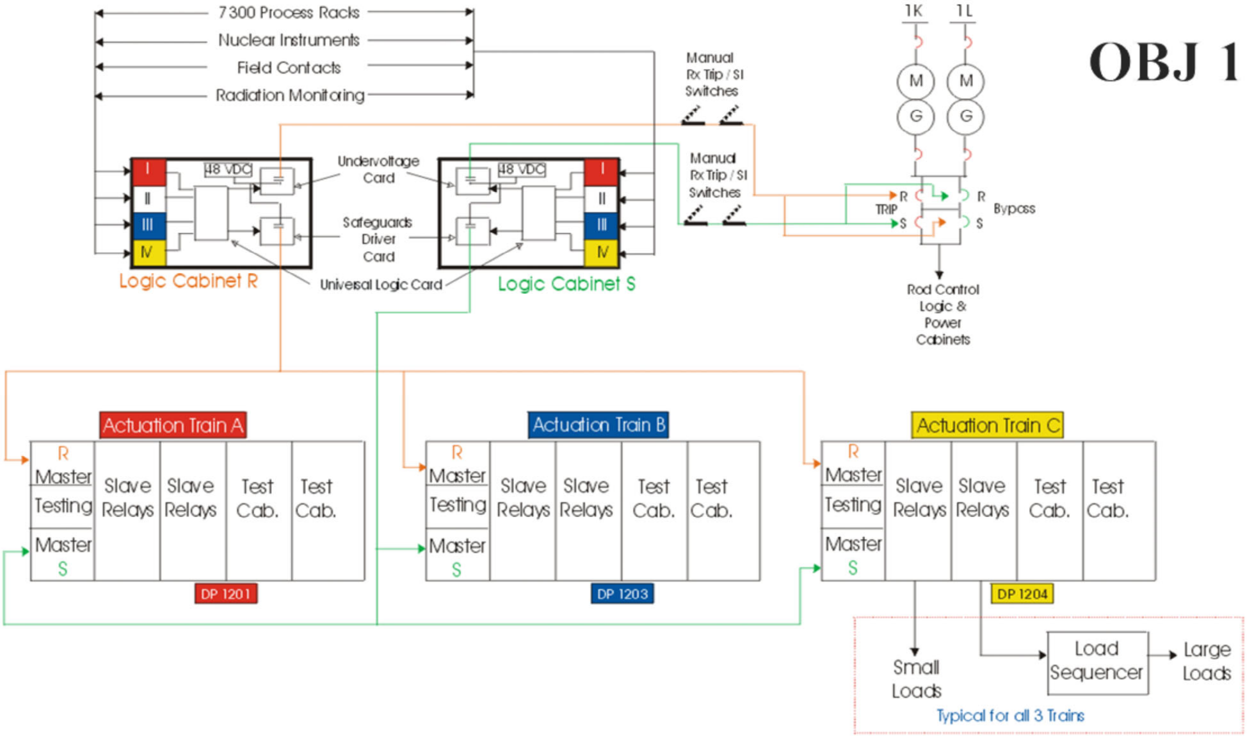
Control switches are at the MG set panel.



Reactor Trip and Bypass Breakers **OBJ 1**

- The system uses series connected Reactor Trip Breakers so the trip of only one breaker will cause the rods to drop through and into the bottom of the core.
- The Bypass breakers can be connected in parallel with the reactor trip breaker.
  - allows testing of Reactor Trip Breakers.
- Bypass breakers (normally racked out) are interlocked in such a way as to prevent the simultaneous closure of both.

OBJ 1



Shows link between reactor protection system and MG set output/Reactor Trip Breakers.

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY	NEW TAG NEEDED
E1A11(E2A11)/4A	TRAIN A RX TRIP SWGR CONTROL PWR	EAB 10' Distribution Room 007 125V DC SWBD E1A11(E2A11)	ON			
1K1(2K1)/3D	CONTROL ROD DRIVE MOTOR/GENERATOR SET 1	EAB 35' Train B Switchgear Room 212 480V Load Center 1K1(2K1)	RACKED IN OPEN			
E1B11(E2B11)/5C	TRAIN B RX TRIP SWGR CONTROL PWR	EAB 35' Distribution Room 213 125V DC SWBD E1B11(E2B11)	ON			
PL125H BKR 21	MG 1 ROD DRIVE CONT CAB	EAB 60' Hallway 125 VDC Distribution Panel PL125H	ON			
PL125H BKR 22	MG 2 ROD DRIVE CONT CAB	EAB 60' Hallway 125 VDC Distribution Panel PL125H	ON			
1L1(2L1)/3D	CONTROL ROD DRIVE MOTOR/GENERATOR SET 2	EAB 60' Train C Switchgear Room 318 480V Load Center 1L1(2L1)	RACKED IN OPEN			
1L1(2L1)/M4L	ROD HOLDOUT POWER RECEPTACLE	EAB 60' Electrical Penetration Space	ON			
S1	(TOGGLE SWITCH)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control DC Hold Supply Cabinet	OFF			
S1	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			
S2	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			
S3	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			
S4	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			
S5	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			
S6	(THUMBWHEEL)	EAB 60' Rod Drive Power Cab Room 323 Inside Rod Control Logic Cab	(1)			

(1) Record current settings from Plant Curve Book Table 1.1 in Position Required block.

Exam Bank No.: 3123 **Last used on an NRC exam:** Never

RO Sequence Number: 2

Unit 2 is at 100% power.

- Reactor Trip and Safety Injection occurs.
- Containment Pressure is 1.5 psig
- Intact S/G Pressures are 1195 psig
- Operators have transitioned to 0POP05-E0-E010, Loss of Reactor or Secondary Coolant in response to a stuck open PZR PORV.

When directed in 0POP05-EO-E010 to depressurize intact SGs to less than 1000 psig, the steam dumps will **INITIALLY** be operated in ___(1)___ with the steam dump "MODE SEL" switch in the ___(2)___ position.

A. (1) manual
(2) STEAM PRESS

B. (1) automatic
(2) STEAM PRESS

C. (1) manual
(2) T-AVG

D. (1) automatic
(2) T-AVG

Answer: A (1) manual, (2) Steam Pressure

Exam Bank No.: 3123 **Source:** New **Modified from** N/A

K/A Catalog Number: APE 008 AA1.03 Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: Turbine bypass in manual control to maintain header pressure

RO Importance: 2.8 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 504.09 **Objective Number:** 81084

Given a copy of a step from 0POP05-EO-EO10, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose, and the result.

Reference: 0POP05-EO-EO10, steps 2e-2i, LOT 504.09 slide 22

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. For a vapor space LOCA, 0POP05-E0-E010, Step 2 initially depressurizes the SGs using steam dumps in MANUAL in the Steam Pressure mode to lower SG pressure to 980 to 994 psig and then will be operated in automatic with a setting between 7.0 and 7.1 to maintain pressure < 994 psig.

B: INCORRECT. Plausible if the student incorrectly believes pressure is lowered by adjusting the setpoint to between 7.0 and 7.1 in automatic.

C: INCORRECT. Plausible if the student thinks the procedure directs operating the Steam Dump System in T-AVG mode.

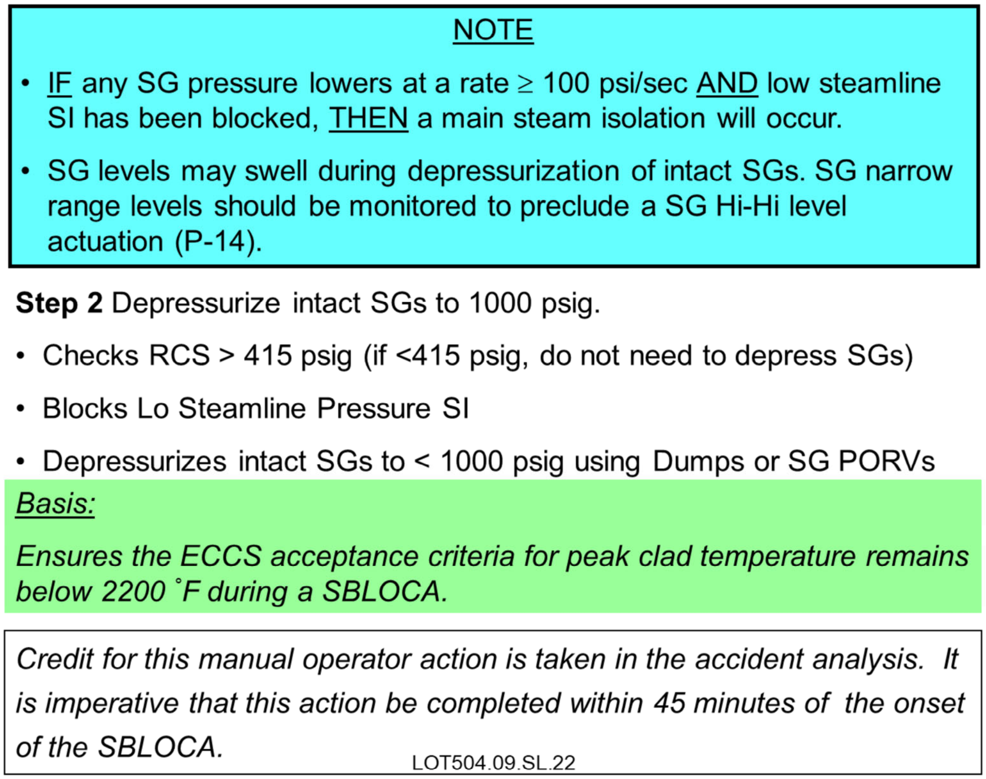
D: INCORRECT. Plausible if the student believes pressure is lowered by adjusting the setpoint to between 7.0 and 7.1 in automatic and the procedure directs operating the Steam Dump System in T-AVG mode.

Question Level: F **Question Difficulty** 3

Justification:

Student must know the proper operation of the Steam Dump System in accordance with procedure.

Obj. 4



NOTE

- IF any SG pressure lowers at a rate ≥ 100 psi/sec AND low steamline SI has been blocked, THEN a main steam isolation will occur.
- SG levels may swell during depressurization of intact SGs. SG narrow range levels should be monitored to preclude a SG Hi-Hi level actuation (P-14).

Step 2 Depressurize intact SGs to 1000 psig.

- Checks RCS > 415 psig (if <415 psig, do not need to depress SGs)
- Blocks Lo Steamline Pressure SI
- Depressurizes intact SGs to < 1000 psig using Dumps or SG PORVs

Basis:

Ensures the ECCS acceptance criteria for peak clad temperature remains below 2200 ˚F during a SBLOCA.

Credit for this manual operator action is taken in the accident analysis. It is imperative that this action be completed within 45 minutes of the onset of the SBLOCA.

LOT504.09.SL.22

This operator action helps to rapidly cooldown the RCS and lower RCS pressure. This lowers break flow and raises SI flow to help recover the core. The lower RCS pressure also results in the crossover leg flashing to steam sooner and forced out the cold leg break by the cooler, denser SG cold leg.

Without forcing this “clearing of the loop seal” the RCS pressure would remain high for too long, more RCS mass would be lost out the break (net loss in RCS mass), and the core would be further uncovered and for a longer time. The resulting clad temperatures may exceed analysis and design limits. Lowering SG pressure will lower pressure ion the RCS and allow more SI flow.

Initiating the lowering of RCS pressure by depressurizing the SGs within 45 minutes of the event will ensure SI flow can refill the downcomer and the core and keep peak clad temperatures well within 10CFR50.46 limits.

Step 2 continued from previous page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
	____e. CHECK steam dump in steam pressure mode	e. PERFORM the following: 1) PLACE steam dump controller in MANUAL with zero demand. 2) ADJUST "HDR PRESS CONT PK-0557" setpoint to BETWEEN 7.0 (980 PSIG) and 7.1 (994 PSIG). 3) PLACE steam dump "MODE SEL" switch in the STEAM PRESS position. 4) DEPRESSURIZE intact SGs to BETWEEN 980 PSIG and 994 PSIG using steam dumps in MANUAL. 5) GO TO Step 2.i. -------------------------------------
	____f. ENSURE "HDR PRESS CONT PK-0557" in MANUAL	
	____g. ADJUST "HDR PRESS CONT PK-0557" setpoint to BETWEEN 7.0 (980 PSIG) and 7.1 (994 PSIG)	
	____h. DEPRESSURIZE intact SGs to BETWEEN 980 PSIG and 994 PSIG using steam dumps in MANUAL	
	____i. CHECK RCS TAVG - LESS THAN 563°F	i. PERFORM the following: 1) WHEN RCS TAVG lowers to 563°F, THEN PLACE steam dump "INTLK SEL" switches to BYPASS INTERLCK. 2) GO TO Step 2.k. -------------------------------------
	____j. PLACE steam dump "INTLK SEL" switches to BYPASS INTERLCK.	
	____k. ENSURE "HDR PRESS CONT PK-0557" in AUTO	

Step 2 continued on next page.

Exam Bank No.: 3124 **Last used on an NRC exam:** Never

RO Sequence Number: 3

Unit 1 was at 100% power.

- A small break LOCA and a loss of offsite power occurred 30 minutes ago.
- The following conditions exist:
  - RCS pressure is 1876 psig and lowering.
  - CET temperatures are 600ºF and rising.
  - SG pressures are 1030 psig and lowering.
  - RCS Thots are 575ºF and rising.
  - RCS Tcolds are 570ºF and rising.

Natural circulation flow ___(1)___ currently providing core cooling.

Another method of core cooling post LOCA is by reflux boiling (cooling) which condenses steam in the SGs and returns the fluid to the core via the RCS ___(2)___.

A. (1) is
(2) cold legs

B. (1) is
(2) hot legs

C. (1) is NOT
(2) cold legs

D. (1) is NOT
(2) hot legs

Answer: D (1) is NOT
(2) hot legs

Exam Bank No.: 3124 **Source:** New **Modified from** N/A

K/A Catalog Number: EPE 009 EK1.01 Knowledge of the operational implications of the following components as they apply to the small break LOCA: Natural circulation and cooling, including reflux cooling

RO Importance: 4.2 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(3)

STP Lesson: LOT 501.21 **Objective Number:** 32255

GIVEN a set of plant conditions or event description, be able to PREDICT the sequence of events and trends of plant parameters for a transient or accident involving a decrease in reactor coolant inventory.

Reference: LOT 501.21 PPT slide 70, 0POP05-EO-ES12, Step 15 RNO a.4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as the conditions given support establishing natural circulation but do not meet the criteria to verify it is occurring and it is reasonable to believe that the condensed steam for reflux cooling returns via the cold legs.

B: INCORRECT. Plausible as the conditions given support establishing natural circulation but do not meet the criteria to verify it is occurring.

C: INCORRECT. Plausible as it is reasonable to believe that the condensed steam for reflux cooling returns via the cold legs.

D: CORRECT. With RCS pressure at 1876 psig (1891 psia) natural circulation flow is NOT occurring as subcooling is less than 35ºF (28ºF) and RCS temperatures are rising.. By definition Reflux boiling (cooling) occurs when the steam condensed in the SGs is returned to the core via the hot legs.

Question Level: H **Question Difficulty** 3

Justification:

This question requires analyzing conditions to determine if natural circulation flow is occurring and the how reflux boiling takes place.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 15 continued from previous page.

3) START RCP(s) per 0POP02-RC-0004, OPERATION OF REACTOR COOLANT PUMP as necessary to provide normal pressurizer spray.

4) IF an RCP can NOT be started, THEN VERIFY natural circulation:

- o RCS subcooling based on core exit T/Cs - GREATER THAN 35°F [45°F]
- o SG pressures - STABLE OR LOWERING
- o RCS hot leg temperatures - STABLE OR LOWERING
- o Core exit T/Cs - STABLE OR LOWERING
- o RCS cold leg temperatures - AT SATURATION TEMPERATURE FOR SG PRESSURE, REFER TO ADDENDUM 5, SATURATION CURVE

5) IF natural circulation NOT verified, THEN RAISE steam dumping rate.

___b. STOP RCP(s) NOT required for normal pressurizer spray

Decreased RCS Inventory
sbloca

What is Reflux Boiling (Cooling)?

- **Steam from the core rises via the hot legs to the steam generators where it is condensed and returned to the core via the hot legs**

- Natural Circulation
  - **Remember that noncondensables can impede natural circulation
  - Heat Source (core) must be below the Heat Sink (SG)
    - Sets up the required thermal driving head ('delta density')
  - Coolable Geometry
  - Th and CET temperatures stable or lowering
  - Tc at saturation temperature for SG pressure
  - SG pressure stable or lowering (steam flow – demand)
  - AFW flow required to maintain SG level

Exam Bank No.: 3125 **Last used on an NRC exam:** Never

RO Sequence Number: 4

A LOCA has occurred on Unit 1.

- The crew is at step 14 of 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant "CHECK if Charging Flow Has Been Established".

The basis for establishing charging flow is to ________(1)________.

A Charging Pump should **NOT** be started if __________(2)_________.

A. (1) try to reestablish adequate conditions to terminate Safety Injection
   (2) charging pumps were secured per 0POP05-EO-ES13, Transfer to Cold Leg Recirculation

B. (1) try to reestablish adequate conditions to terminate Safety Injection
   (2) seal injection isolation valves are isolated

C. (1) allow differentiation between a small break and large break LOCA
   (2) charging pumps were secured per 0POP05-EO-ES13, Transfer to Cold Leg Recirculation

D. (1) allow differentiation between a small break and large break LOCA
   (2) seal injection isolation valves are isolated

Answer: A (1) try to reestablish adequate condtions to terminate Safety Injection (2) charging pumps were secured per 0POP05-EO-ES13, Transfer to Cold Leg Recirculation

Exam Bank No.: 3125 **Source:** New **Modified from** N/A

K/A Catalog Number: EPE 011 EA2.05 Ability to determine or interpret the following as they apply to a Large Break LOCA: Significance of charging pump operation

RO Importance: 3.3 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(8)

STP Lesson: LOT 504.09 **Objective Number:** 81084

Given a copy of a step from 0POP05-EO-E010, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose and the result

Reference: WOG ERG LPBG-E-1 Rev. 3, LOT 504.09 slides 25 and 27, 0POP05-EO-EO10, step 14

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Charging flow is restored to the RCS in Step 14 to establish conditions to allow Safety Injection termination. If charging pumps were secured in ES-13 the RNO sends you directly to the step to determine if SI should be terminated and a charging pump is not started.

B: INCORRECT. Plausible if the student thinks restoring charging flow is contingent on being able to restore seal injection flow and the RNO does have a step involving those valves.

C: INCORRECT. Plausible as starting a charging pump and providing flow in addition to the HHSI pumps may hold RCS pressure above 415 psig which is used to determine whether you have a small or large break LOCA and determines whether LHSI is needed..

D: INCORRECT. Plausible if the student thinks restoring charging flow is contingent on being able to restore seal injection flow, the RNO has a step involving those valves.

Question Level: F **Question Difficulty** 3

Justification:

The question requires recall of an EOP step and its basis.

STEP DESCRIPTION TABLE FOR E-1 Step 10

STEP: Check If Charging Flow Has Been Established

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

BASIS:

Charging flow to the RCS is established to try to reestablish adequate conditions to terminate SI. If charging pumps are already running, then seal cooling is adequate and charging flow can be established as necessary. If pumps are not running, then seal injection flow is lost, and the only remaining source of seal cooling is CCW. If CCW flow to the RCP thermal barrier is also lost, then the seal is assumed to be already heated up. Rather than initiate the slow and tedious process of reestablishing seal cooling at this time, the seal injection flow path is isolated to allow the charging pumps to be started and charging flow to be established.

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

Means for establishing necessary charging flow

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___14	CHECK If Charging Flow Has Been Established:	
___a.	CCPs - AT LEAST ONE RUNNING	a. PERFORM the following: 1) IF CCPs are secured per 0POP05-EO-ES13, TRANSFER TO COLD LEG RECIRCULATION, THEN GO TO Step 15. 2) CLOSE seal injection isolation valves: o "SEAL INJ ISOL MOV-0033A" o "SEAL INJ ISOL MOV-0033B" o "SEAL INJ ISOL MOV-0033C" o "SEAL INJ ISOL MOV-0033D" 3) CLOSE the CCP discharge valve for the CCP to be started. 4) CLOSE the charging flow control valve. 5) IF charging flow control valve will NOT close, THEN PERFORM the following: a) ESTABLISH charging flow per ADDENDUM 2, ESTABLISHING ALTERNATE CHARGING FLOW CONTROL. b) GO TO Step 14.c. 6) OPEN the recirculation valve for the CCP to be started. 7) START one CCP. 8) OPEN the CCP discharge valve for the pump that was started.

Step 14 continued on next page.

Step 10 Monitor Pressurizer PORV's and Isolation Valves.

Basis:

To ensure operability of the block valves, we verify power available. PZR PORV's are closed to preclude the possibility of an undetected stuck open valve. Both block valves should be open unless one was closed to isolate a stuck-open PZR PORV. At least one block valve is left open for pressure excursions in the RCS.

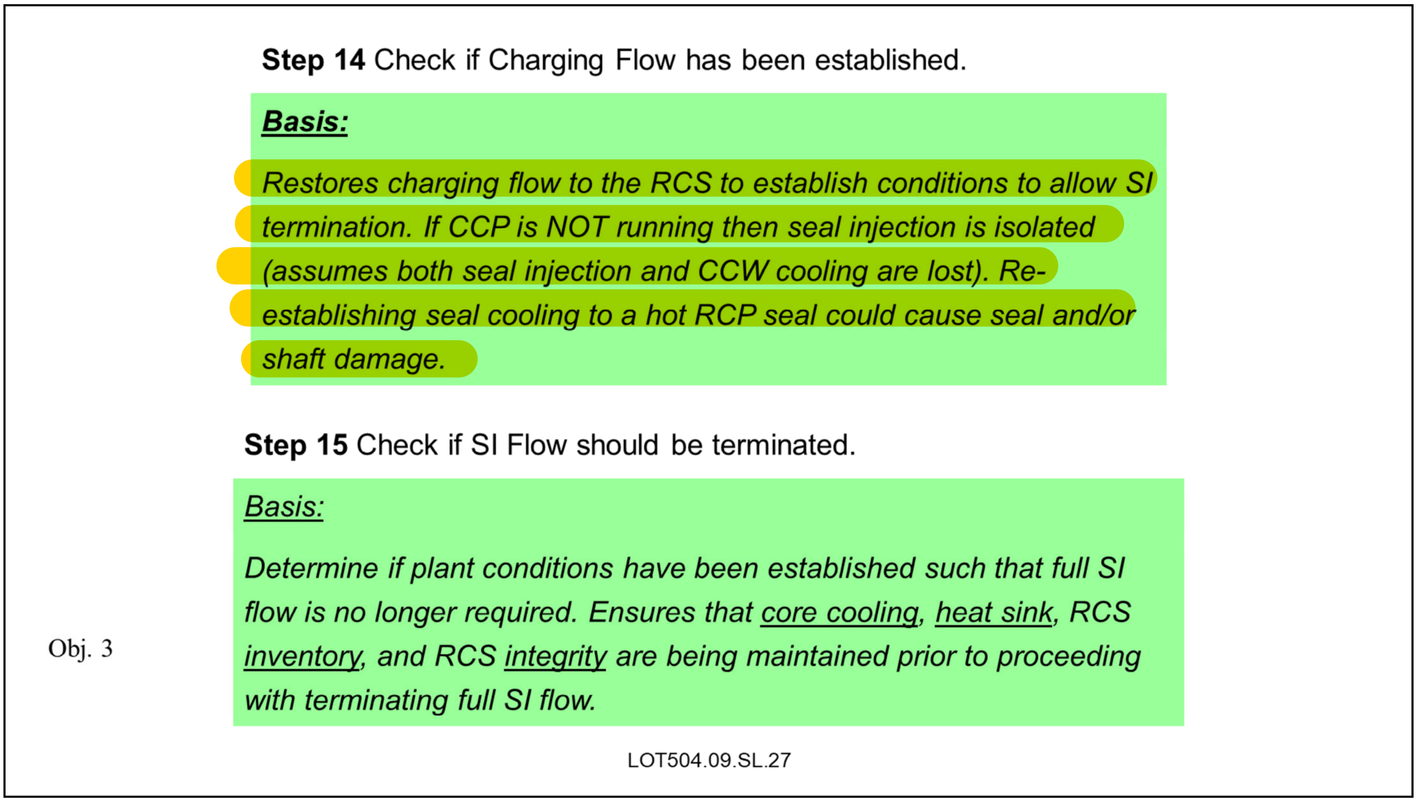
Step 11 Establish IA To Containment

Basis:

Obj. 3

Restores control of air-operated valves inside CTMT. The reason for establishing IA at this time is to enable the operator to establish normal charging to aid in meeting SI termination criteria and also provide seal cooling (WOG Background Document). Allows operation of normal and auxiliary spray valves and RHR system operation.

LOT504.09.SL.25



Step 14 Check if Charging Flow has been established.

Basis:

Restores charging flow to the RCS to establish conditions to allow SI termination. If CCP is NOT running then seal injection is isolated (assumes both seal injection and CCW cooling are lost). Re-establishing seal cooling to a hot RCP seal could cause seal and/or shaft damage.

Step 15 Check if SI Flow should be terminated.

Basis:

Determine if plant conditions have been established such that full SI flow is no longer required. Ensures that core cooling, heat sink, RCS inventory, and RCS integrity are being maintained prior to proceeding with terminating full SI flow.

Obj. 3

LOT504.09.SL.27

Step 14:

Also, rather than initiate the slow and tedious process of re-establishing seal cooling to a hot seal package, the seal injection flowpath is isolated to allow starting CCP and establishing charging flow.

Exam Bank No.: 3126 **Last used on an NRC exam:** Never

RO Sequence Number: 5

Unit 1 is at 100% power with Tavg and PZR level on program.

- CV-FCV-0205, Charging Flow Control Valve fails closed.
- The RO immediately closes CV-FV-0011, Letdown Orifice Header Isolation Valve.
- RCP Number One Seal Leakoff is 3 gpm per pump.
- RCP Seal Injection Flow is 8 gpm per pump.

With NO operator action, pressurizer backup heaters will energize in about ___(1)___ minutes.

Per 0POP04-RP-0002, Loss of Automatic Pressurizer Level Control, the crew should ________(2)_________.

A. (1) 11
(2) place Excess Letdown in service

B. (1) 17
(2) place Excess Letdown in service

C. (1) 11
(2) de-energize the pressurizer backup heaters

D. (1) 17
(2) de-energize the pressurizer backup heaters

Answer: B (1) 17; (2) place Excess Letdown in service

Exam Bank No.: 3126 **Source:** New **Modified from** N/A

K/A Catalog Number: APE 022 AA1.03 Ability to operate and /or monitor the following as they apply to the Loss of Reactor Coolant Makeup: PZR level trend

RO Importance: 3.2 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.06 **Objective Number:** 98084

Given plant conditions with a Reactor Coolant leak in progress, Estimate the leakage rate by performing a CVCS flow balance or an inventory balance, determine their effects on the Pressurizer pressure and level control system. LOT 201.05 4288 - DESCRIBE the RCP flow paths. Include seals, normal rates and direction.
LOT 201.05 50757 - Describe the normal range of the following: B. Seal Injection flow, D. No. 1 Seal Leakoff Flow.

Reference: LOT 201.06 slides 84, LOT 201.05 slides 51-52, and POP09-AN-04M8 window C-6 origin and setpoint, 0PSP03-RC-0006A, p. 9 0POP04-RC-0002

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because this is the number calculated if the student fails to subtract the total seal leakoff flow from the seal injection flow. (68.3 gals/% X 5%)/4(8 gpm) = 10.672 minutes. The second part is correct.

B: CORRECT. Time to change from setpoint to the point the PZR Backup Heaters energize on level deviation (5% above setpoint) = (68.3 gals/% X 5%)/4(8 gpm-3 gpm) = 17.075 minutes. 0POP04-RP-0002 has guidance to place excess letdown in service to control pressurizer level.

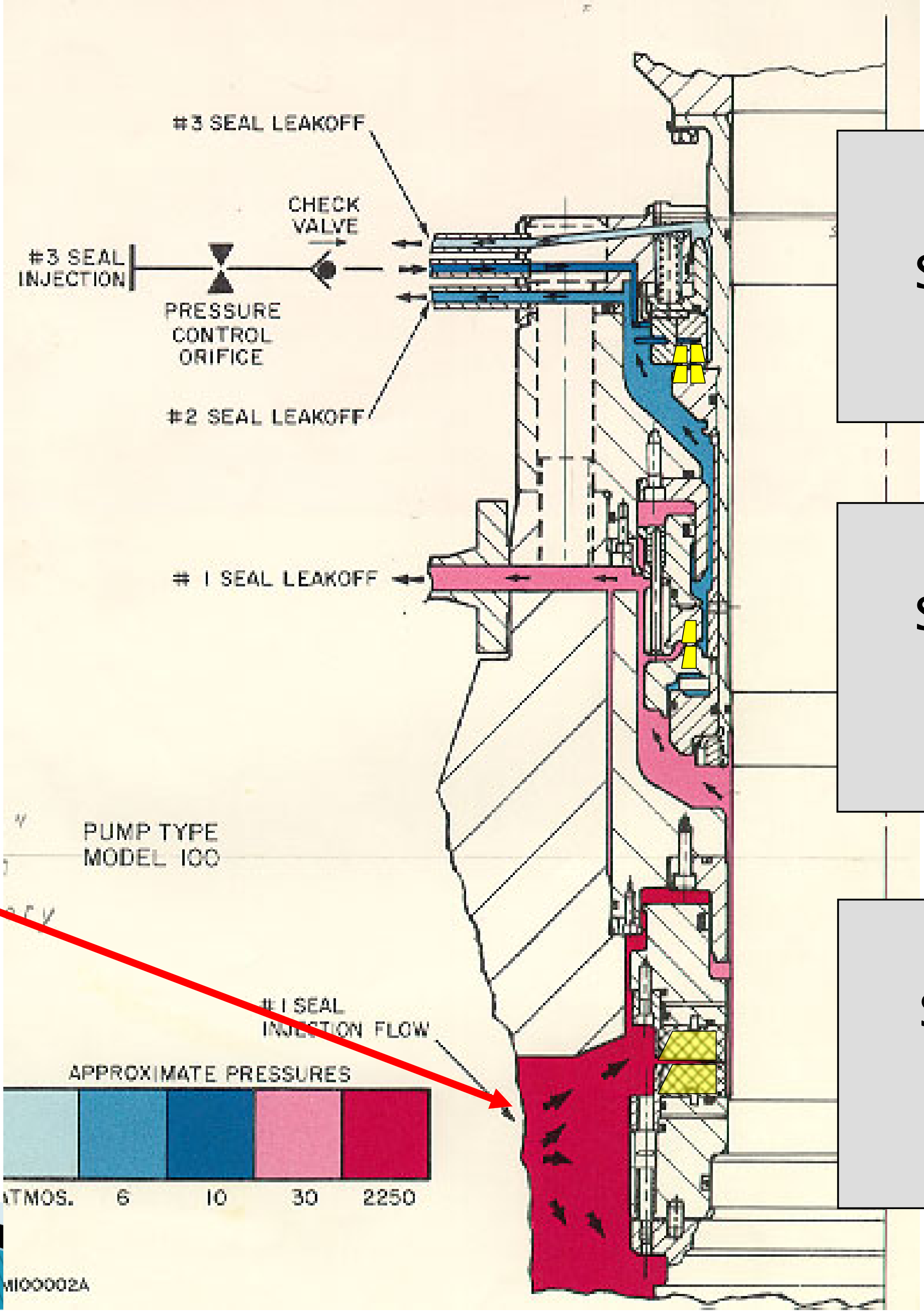
C: INCORRECT. Plausible because this is the number calculated if the student fails to subtract the total seal leakoff flow from the seal injection flow. (68.3 gals/% X 5%)/4(8 gpm) = 10.672 minutes and because backup pressurizer heaters energizing will start an RCS pressurization until the spray valves open and the student could think that it is desirable to secure them.

D: INCORRECT. . Plausible because backup pressurizer heaters energizing will start an RCS pressurization until the spray valves open and the student could think that it is desirable to secure them. The first part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student must analyze the data given in the stem of the question along with knowing the gals/% change in pzr level and the % deviation of pzr level to setpoint and correct methodology per 0POP04-RC-0002.



No.1 SEAL INJ

8-13 GPM

2250 PSIG

110-130 F

No.3

SEAL LKOFF

400 cc/hr

No.2

SEAL LKOFF

3 GPH

10 PSIG

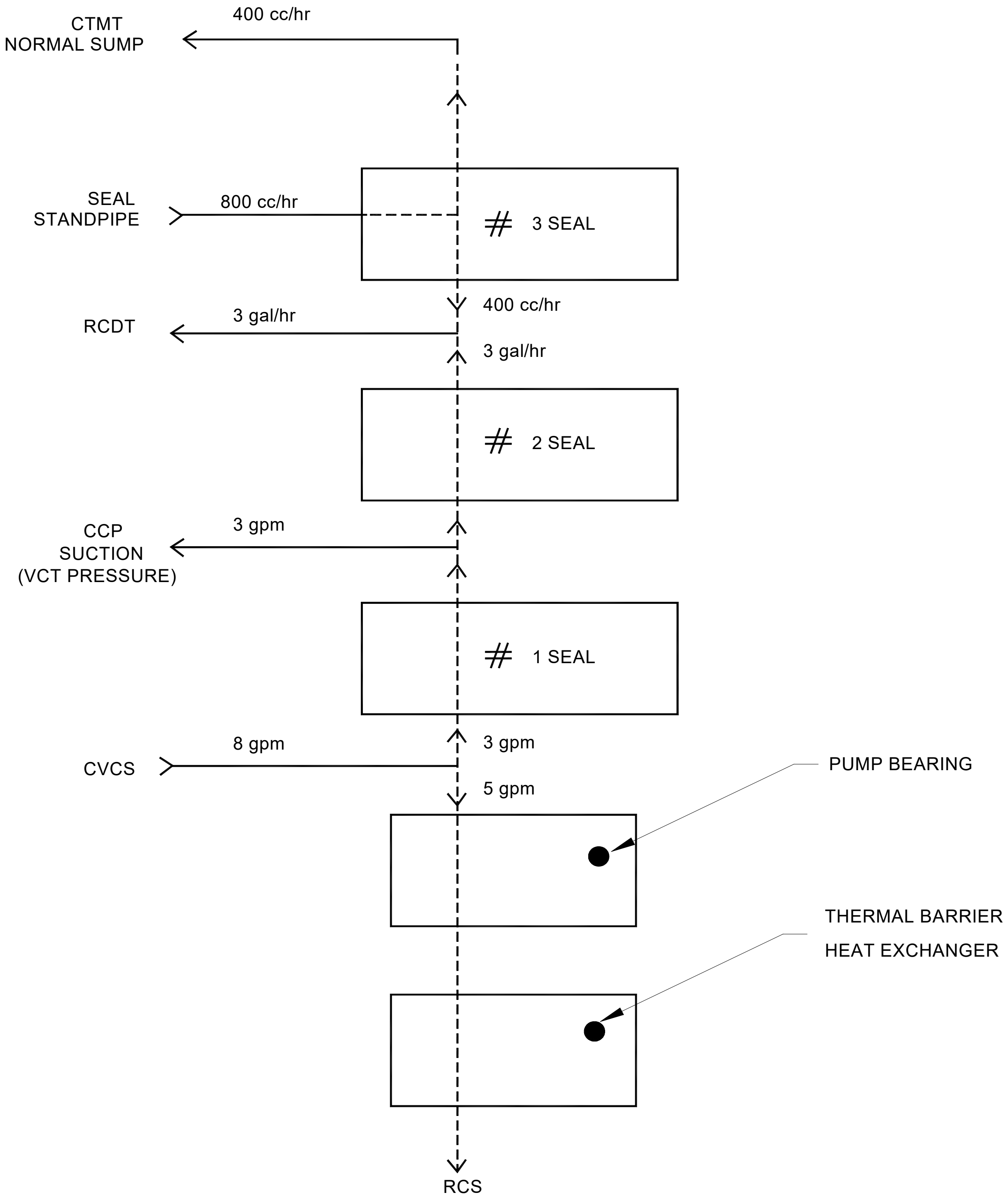
No.1

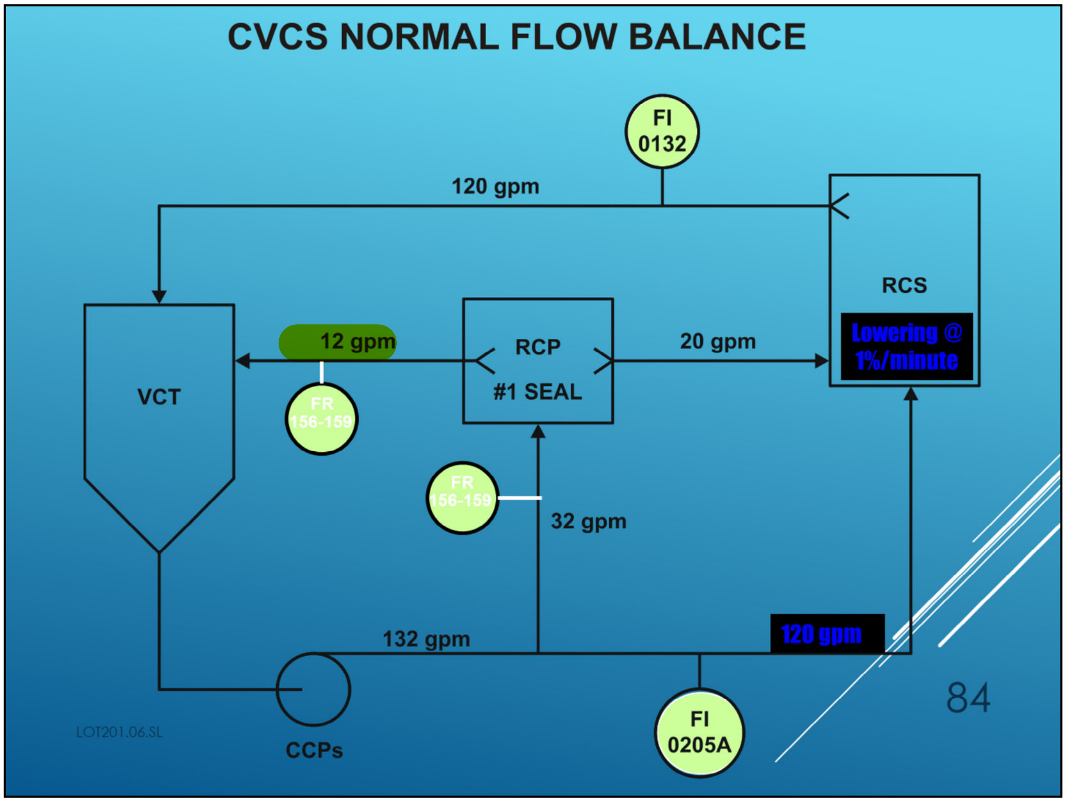
SEAL LKOFF

3 GPM

30 PSIG

RCP Seal Injection Simplified Flow Diagram





CVCS NORMAL FLOW BALANCE

Best way to accurately estimate leakage is mark Pzr level & VCT level, start stopwatch, 2 min later, mark both again. Do the calc. Get the leakrate

PZR = 68.3 Gal/%

VCT=33.9gal/%

PRZR LEVEL DEV HI B/U HTRS ON

Subsequent Actions: (continued)

______ 4) ENSURE RCS Tavg within 1.5°F of Tref.

______ 5) ADJUST "CHG FLOW CONT FK-0205" as necessary to maintain pressurizer level on "PRZR PROG LEVEL LI-0665". (CP004)

Probable Causes:

1) Pressurizer insurge
2) Pressurizer level control malfunction

Origin and Setpoint:

1) 1(2)-RC-LT-0465	level 5% above program
2) 1(2)-RC-LT-0467	level 5% above program

References:

1) P&ID, 5R149F05003
2) PLS, HL&P Spec. 5Z010ZS1101
3) USQE 99-66-23
4) 0POP04-CV-0004, Loss Of Normal Letdown
5) 0POP04-RP-0002, Loss of Automatic Pressurizer Level Control
6) CREE 10-23973-3, Pressurizer Backup Heaters Group 2A tripped upon receiving a "Pressurizer Level Deviation High Backup Heaters On" signal.

NOTE

- This test should be performed over a two hour period unless steady conditions cannot be maintained. A test period of one hour is sufficient to satisfy the surveillance test. The test period may be specified at the discretion of the Shift Manager/Unit Supervisor.
- Although makeup to the VCT for T_{ave} control during the data collection period does not invalidate the test results, plant conditions should be established, when possible, such that this activity is not be required during the test period. IF a makeup is required, THEN sufficient time for plant stabilization should be allowed prior to obtaining stop time data.
- Steps 4.8 and 4.9 direct data recording requirements when Makeup Totalizer FQI-0111B is **NOT** functional.

5.5 WHEN determined test period has elapsed, THEN PERFORM the following:

5.5.1 ENSURE Stop time Reactor Power is within 1% of Start time Reactor Power. ______

5.5.2 ENSURE Stop time PRZR pressure is within 10 psig of Start time PRZR pressure. (IF <1700 psig PRZR pressure and utilizing stable RCS pressure, THEN ENSURE STOP time RCS pressure is within 10 psig of START time QDPS RCS pressure.) ______

5.5.3 WHEN Reactor Power and Pressure data are within above limits, THEN RECORD STOP data on Data Sheet 1. ______

5.6 DETERMINE PRT level in gallons from percent using the Plant Curve Book, Figure 10.8, PZR Relief Tank. ______

5.7 DETERMINE RCDT level in gallons from percent using the Plant Curve Book, Figure 10.9, RC Drain Tank. ______

5.8 PERFORM Pressurizer level correction factor calculation:

68.3 + 0.023 (2235 - __________ psig) = ________ gal/% ______
Stop Press

5.9 PERFORM temperature correction factor calculation:

5.9.1 IF pressure is approximately 400 psig AND Stop Temperature is between 240 and 400°F, THEN PERFORM the following:

32 + 0.147 (_________°F - 240) = _________ gal/°F
Stop Temp ______

5.9.2 IF pressure is approximately 2235 psig AND Stop Temperature is between 500 and 600°F, THEN USE Addendum 1. ______

Addendum 2	**Alternate Charging Path With FCV-0205 Inoperable**	Addendum 2 Page 1 of 1

STEP	**ACTIONS/EXPECTED RESPONSE**	**RESPONSE NOT OBTAINED**
_____ **1.0**	**DETERMINE If Local Manual Control Of Charging Is Desired**	PERFORM the following: **a.** CLOSE MOV-0025 to isolate charging. **b.** CLOSE letdown orifice header stop valve FV-0011 to isolate letdown. **c.** PLACE excess letdown in service per Addendum 3 to maintain Pressurizer level on Pressurizer Program Level. **d.** RETURN TO Procedure Step 4.0. ---
_____ **2.0**	**DISPATCH An Operator To 19 Ft MAB Room 079, CVCS High Energy Valve Room To Perform The Following:** • CLOSE “1(2)-CV-0254A FCV-205 ISOLATION” • THROTTLE “1(2)-CV-0255 FCV-205 BYPASS” to *MAINTAIN* Pressurizer Level – AT PROGRAM	---
_____ **3.0**	**RETURN TO Procedure Step 4.0**	---

Exam Bank No.: 3212 **Last used on an NRC exam:** Never

RO Sequence Number: 6

Unit 1 is in a refueling outage.

- RCS level is at mid-loop.

A Loss of Offsite Power occurs and the crew enters 0POP04-RH-0001, Loss of Residual Heat Removal.

- All Emergency Diesel Generators start and energize their respective 4160V ESF Buses.

The operator is FIRST directed to close the ______(1)______ Isolation Valves for the secured/tripped RHR pumps as soon as possible to prevent the RWST from ______(2)______ the RCS.

A. (1) Low Pressure Letdown
(2) draining to

B. (1) Low Pressure Letdown
(2) filling from

C. (1) Cold Leg Injection
(2) draining to

D. (1) Cold Leg Injection
(2) filling from

Answer: C (1) Cold Leg Injection (2) draining to

Exam Bank No.: 3212 **Source:** New **Modified from**

K/A Catalog Number: APE025 AK3.03 Knowledge of the reasons for the following responses as they apply to the Loss of Residual Heat Removal System: Immediate actions contained in EOP for loss of RHRS

RO Importance: 3.9 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 505.01 **Objective Number:** 32636

Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-RH-0001, Steps 2 and 3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as the Low Pressure Letdown line is used during RHR operations for cleanup and pressure control, and the RHR system and the CVCS system also contains connections to the RWST and prior to starting an RHR Pump it is required to be closed. Low pressure letdown is isolated in step 5 but that is after cold leg injection vlaves are isolated in step 3 and the stem asks which valves are isolated first. The RWST does drain to the RCS if the right valves are opened.

B: INCORRECT: Plausible as the Low Pressure Letdown line is used during RHR operations for cleanup and pressure control, and the RHR system and the CVCS system also contains connections to the RWSTand prior to starting an RHR Pump it is required to be closed. Low pressure letdown is isolated in step 5 but that is after cold leg injection vlaves are isolated in step 3 and the stem asks which valves are isolated first. Students may have confusion over the elevations and pressures in the RCS and RWST.

C: CORRECT: The operator is directed by 0POP04-RH-0001 for this situation to close the respective Cold Leg Injection Valves in step 3 to prevent gravity fill of thr RCS from the RWST.

D: INCORRECT: Plausible as these are the correct valves per 0POP04-RH-0001 and with confusion over the elevations and pressures in the RCS and RWST.

Question Level: F **Question Difficulty** 3

Justification:

The student must procedure actions and reason.

CAUTION

Changes in RCS pressure could result in inaccuracies in RCS level indication.

NOTE

Addendum 1 can be referenced for RCS/RHR Simplified Elevation Diagram.

NOTE

- **In Mode 5 or 6:** An unplanned rise in RCS temperatures to greater than 200°F (Tavg) represents a potential ALERT per 0ERP01-ZV-IN01, Emergency Classification.
- **In Mode 5 or 6:** IF the Core is OR will be uncovered, THEN a potential Site Area Emergency exists per 0ERP01-ZV-IN01, Emergency Classification.
- When a RHR pump is stopped during Reduced Inventory a gravity flow path from RWST to RCS exists. To minimize the RCS inventory transient, the cold leg injection isolation valve SHALL be closed as soon as possible when gravity fill is **NOT** directed by this procedure.
- Foldout CIP should be open.

______	2.0	**CHECK RCS Level In Reduced Inventory - LESS THAN 36 FEET 3 INCHES**	GO TO Step 24.0
______	3.0	**CLOSE Cold Leg Injection Valves For Secured / Tripped RHR Pump(s):** • "LOOP A Tc INJ MOV-0031A" • "LOOP B Tc INJ MOV-0031B" • "LOOP C Tc INJ MOV-0031C"	

Exam Bank No.: 2393 **Last used on an NRC exam:** 2015

RO Sequence Number: 7

Given the following conditions:

- Unit 1 is at 100% Power
- CCW Surge tank level 63% and slowly lowering
- Annunciator CCW SURGE TANK LEVEL LO in
- A train CCW pump in operation
- B train CCW pump in Standby

To mitigate this event the crew should first enter _____(1)______.

Automatic closure of ___(2)______ has occurred.

A. (1) 0POP04-CC-0001, Component Cooling Water System Leak
(2) "RCDT HX 1A INL 1-CC-MOV-0392" (CCW To RCDT Hx isol)

B. (1) 0POP04-CV-0004, Loss of Normal Letdown
(2) "RCDT HX 1A INL 1-CC-MOV-0392" (CCW To RCDT Hx isol)

C. (1) 0POP04-CC-0001, Component Cooling Water System Leak
(2) "SUPPLY ISOL 1-CC-MOV 0768" (CCW Train A isol to charging pump header)

D. (1) 0POP04-CV-0004, Loss of Normal Letdown
(2) "SUPPLY ISOL 1-CC-MOV 0768" (CCW Train A isol to charging pump header)

Answer: A (1) 0POP04-CC-0001, Component Cooling Water Leak (2) "RCDT HX 1A INL 1-CC-MOV-0392" (CCW to RCDT Hx isolation)

Exam Bank No.: 2393 **Source:** Bank **Modified from** N/A

K/A Catalog Number: APE 026 G2.4.47 Ability to recognize abnormal indications for systems operating parameters that are entry level conditions for emergency and abnormal operating procedures

RO Importance: 4.5 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 505.01 **Objective Number:** 92106

GIVEN plant conditions/systoms, EVALUATE the conditions/symptoms and STATE whether or not the referenced procedure is to be used.

Reference: 0POP04-CC-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: At 64.6% level on CCW Surge tank "RCDT HX 1A(2A) INL MOV-0392" (CCW to RCDT Hx isolation) will automatically close. 63% level indication on the CCW surge tank is below the required entry condition for 0POP04-CC-0001 (Component Cooling Water Leak)

B: INCORRECT:Plausible because letdown will have to be isolated due to due to CCW to letdown HX isolating at 64.6% in the CCW Surge Tank. Second part is correct.

C: INCORRECT: Plausible because AUTOMATIC closure of "SUPPLY ISOL MOV 0768 (CCW Train A isol to charging pump header) does happen on a lowering surge tak level but does not happen until CCW surge tank level is below 61%. First part is correct.

D: INCORRECT: Plausible because letdown will have to be isolated due to due to CCW to letdown HX isolating at 64.6% in the CCW Surge Tank and AUTOMATIC closure of "SUPPLY ISOL MOV 0768 (CCW Train A isol to charging pump header) does happen on a lowering surge tak level but does not happen until CCW surge tank level is below 61%. First part is correct.

Question Level: H **Question Difficulty** 3

Justification:

Must analyze the given conditions and then determine the procedure needed and the affected component.

_____ **3.0** ***MONITOR* For CCW Surge Tank Low Level Non-Vital Supply Valves Isolation:**

- CCW Surge Tank Level - LESS THAN 64.6%

OR

- CCW Surge Tank Level - HAS BEEN LESS THAN 64.6%

Plant Computer Points

- Train A – CCLE4504
- Train B – CCLE4506
- Train C – CCLE4508

GO TO Step 9.0.

_____ **a.** **CHECK the following valves are closed:**

- "NNS LOADS INL ISOL MOV-0235" (CCW to common supply header isolation)
- "NNS LOADS INL ISOL MOV-0236" (CCW to common supply header isolation)
- "BRANCH ISOL MOV-0297" (CCW to RCDT Hx and Excess LD Hx isolation)
- "RCDT HX 1A(2A) INL MOV-0392" (CCW to RCDT Hx isolation)
- "EXCESS LETDOWN HX 1A(2A) INL MOV-0393" (CCW to Excess LD isol)

a. Manually CLOSE the valves.

_____ **11.0** ***MONITOR*** **For CCW Surge Tank Low Level Common Header Valves Isolation:**

- CCW Surge Tank Level - LESS THAN 61.5%

OR

- CCW Surge Tank Level - HAS BEEN LESS THAN 61.5%

Plant Computer Points

- Train A – CCLE4504
- Train B – CCLE4506
- Train C – CCLE4508

GO TO Step 25.0.

Step 11.0 continued on next page

Step 11.0 continued from previous page

_____ **a. CHECK the following valves are closed:**

Manually CLOSE the valves.

Charging System Components

- "SUPPLY ISOL MOV-0768"(CCW Train A isol to charging pump header)
- "RET ISOL MOV-0772"(CCW return header isolation from charging pumps to Train A)
- "SUPPLY X-CONN FV-4656"(CCW to charging pump A crosstie)
- "RETURN X-CONN FV-4657"(CCW return from charging pump A crosstie)

CCW Header Isolations

Train A

- "CCW SPLY HDR ISOL" "MOV-0316"
- "CCW RET HDR ISOL" "MOV-0052"

Train B

- "CCW SPLY HDR ISOL" "MOV-0314"
- "CCW RET HDR ISOL" "MOV-0132"

Train C

- "CCW SPLY HDR ISOL" "MOV-0312"
- "CCW RET HDR ISOL" "MOV-0192"

Addendum 1

Automatic Actions for CCW Surge Tank Level Change

1. WHEN CCW Surge Tank level is less than 64.6%, THEN the following valves will automatically close:

   - "NNS LOADS INL ISOL MOV-0235" (CCW to common supply header isolation)
   - "NNS LOADS INL ISOL MOV-0236" (CCW to common supply header isolation)
   - "BRANCH ISOL MOV-0297" (CCW to RCDT Hx and Excess LD Hx isolation)
   - "RCDT HX 1A(2A) INL MOV-0392" (CCW to RCDT Hx isolation)
   - "EXCESS LETDOWN HX 1A(2A) INL MOV-0393" (CCW to Excess LD isol)

2. WHEN CCW Surge Tank level is less than 61.5%, THEN the following valves will automatically close:

   Charging System Components

   - "SUPPLY ISOL MOV-0768"(CCW Train A isol to charging pump header)
   - "RET ISOL MOV-0772"(CCW return header isolation from charging pumps to Train A)
   - "SUPPLY X-CONN FV-4656"(CCW to charging pump A crosstie)
   - "RETURN X-CONN FV-4657"(CCW return from charging pump A crosstie)

   CCW Header Isolations

   Train A

   - "CCW SPLY HDR ISOL" "MOV-0316"
   - "CCW RET HDR ISOL" "MOV-0052"

   Train B

   - "CCW SPLY HDR ISOL" "MOV-0314"
   - "CCW RET HDR ISOL" "MOV-0132"

   Train C

   - "CCW SPLY HDR ISOL" "MOV-0312"
   - "CCW RET HDR ISOL" "MOV-0192"

3. WHEN CCW Surge Tank Level less than 61.5%, AND the associated CCW Pump in that train is operating, THEN the following RHR HX "CCW OUTL" valve will automatically open:

Train A	Train B	Train C
"FV-4531"	"FV-4548"	"FV-4565"

Exam Bank No.: 2949 **Last used on an NRC exam:** 2020

RO Sequence Number: 8

The unit is at 100% power.

- The Pressurizer Pressure Selector Switch is in the 455/456 position.
- Pressurizer Pressure Transmitter PT-455 fails HIGH

Per 0POP04-RP-0001, Loss of Automatic Pressurizer Pressure Control, complete the following:

As an immediate action, the RO should first __________(1)__________.
The reason for this response is to __________(2)_________.

A. (1) place Pressurizer Pressure Controller PK-0655A in MANUAL
   (2) prevent Pressurizer Backup Heaters from energizing

B. (1) place Pressurizer Pressure Controller PK-0655A in MANUAL
   (2) ensure PORV PCV-0655A is closed

C. (1) deselect the failed channel from service
   (2) prevent Pressurizer Backup Heaters from energizing

D. (1) deselect the failed channel from service
   (2) ensure PORV PCV-0655A is closed

Answer: D (1) deslect the failed channel from service (2) ensure PORV PCV-655A is closed

Exam Bank No.: 2949 **Source:** Bank **Modified from** N/A

K/A Catalog Number: APE 027 AK3.03 Knowledge of the reasons for the following responses as they apply to the Pressurizer Pressure Control Malfunctions: Actions contained in EOP for Pzr PCS malfunction

RO Importance: 3.7 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 505.01 **Objective Number:** 92110

Given a precaution, note, or step(s) and the context in which it is used from the referenced procedure, describe its basis and any applicable limits.

Reference: 0POP04-RP-0001, Step 1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as this is the correct action to take for a failed low channel and this response occurs for a failed low channel.

B: INCORRECT: Plausible as this is the correct action to take for a failed low channel and this is the right reason for the given conditions.

C: INCORRECT: Plausible as this is the correct action to take for a channel failed high, and this response occurs for a failed low channel

D: CORRECT: With a failed high channel, it is deselected quickly to ensure the PORV and spray valves all close.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess plant conditions and determine a course of action.

NOTE

Step 1.0 is an IMMEDIATE ACTION Step.

Foldout CIP should be opened.

_____ **1.0 CHECK Pressurizer Pressure Channels - ALL OPERABLE**

PERFORM the following: {CP004}

a. IF any channel has failed **HIGH**, THEN POSITION Pressurizer Pressure Control Selector Switch to remove failed channel from control: {CP004}

Failed Channel	Select
RC-PI-0455	P457/456
RC-PI-0456	P455/458
RC-PI-0457	P455/456 OR P455/458
RC-PI-0458	P455/456 OR P457/456

b. IF any controlling channel has failed **LOW**, THEN:

1) PLACE Pressurizer Pressure Controller RC-PK-0655A in MANUAL. {CP004}

2) POSITION Pressurizer Pressure Control Selector Switch to remove failed channel from control: {CP004}

Failed Channel	Select
RC-PI-0455	P457/456
RC-PI-0456	P455/458
RC-PI-0457	P455/456 OR P455/458
RC-PI-0458	P455/456 OR P457/456

3) IF Pressurizer Pressure Controller RC-PK-0655A is in MANUAL, THEN ADJUST Pressurizer Pressure Controller RC-PK-0655A to control Pressurizer pressure between 2220 and 2250 psig

Step 1.0 continued on next page

Exam Bank No.: 2568 **Last used on an NRC exam:** 2018

RO Sequence Number: 9

The preferred emergency boration suction source in 0POP05-EO-FRS1, Response to Nuclear Power Generation ATWS, is the _____(1)______.

AND

According to 0POP05-EO-FRS1, the reason for using this suction source is to ____(2)____.

A. (1) Boric Acid Tanks
(2) limit RCS inventory concerns

B. (1) Boric Acid Tanks
(2) ensure less time is required to achieve adequate shutdown margin

C. (1) Refueling Water Storage Tank
(2) limit RCS inventory concerns

D. (1) Refueling Water Storage Tank
(2) ensure less time is required to achieve adequate shutdown margin

Answer: A (1) Boric Acid Tanks
(2) limit RCS inventory concerns

Exam Bank No.: 2568 **Source:** Bank **Modified from** N/A

K/A Catalog Number: EPE 029 EK1.03 Knowledge of the operational implications of the following concepts as they apply to the ATWS: Effects of boron on reactivity.

RO Importance: 3.6 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(6)

STP Lesson: LOT 504.28 **Objective Number:** 83555

Given a step, note, or caution from 0POP05-EO-FRS1, state its basis.

Reference: LOT 504.28 Powerpoint Presentation, slide 26, 0POP05-EO-FRS1, EOPT-03.26, FRS1 STPEGS EOP Technical Guideline pgs 3-5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: The boron concentration in the Boric Acid Tanks (BAT)is much greater than in the RWST making the BATs the preferred suction source. If the boration is from a lower concentration source it will take much more volume to insert the same amount of negative reactivity.

B: INCORRECT: Plausible as this is the correct suction source and if the student does not consider that charging flow could be raised to inject RWST required volume in the same time as from the BAT. The procedure takes this into account and requires a higher flow if using the RWST.

C: INCORRECT: Plausible as the RWST is a Class 1E suction source and is one of the options used in the procedure.

D: INCORRECT: Plausible as the RWST is a Class 1E suction source and if the student does not consider that charging flow could be raised to inject RWST required volume in the same time as from the BAT. The procedure takes this into account and requires a higher flow if using the RWST.

Question Level: F **Question Difficulty** 3

Justification:

Student must have fundamental knowledge of emergency boration during ATWS condition.

CAUTION

WHEN using RWST for emergency boration flow, THEN pressurizer level should be closely monitored to prevent water solid conditions in the RCS.

NOTE

BAT(s) are the preferred emergency boration suction source to limit RCS inventory concerns.

___4 INITIATE Emergency Boration Of RCS:

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___a. OPEN alternate boration isolation valve	a. PERFORM one of the following: 1) IF VCT outlet valves are open, THEN COMMENCE boration using the normal boration flowpath. 2) ALIGN charging pump suction to RWST: a) OPEN charging pump suction valves from RWST. b) CLOSE VCT outlet valves.

Step 4 continued on next page.

TITLE RESPONSE TO NUCLEAR POWER GENERATION—ATWS

EOP NO. 0POP05-EO-FRS1 REVISION 23

STP EOP STEP NO.	WOG ERG STEP NO.	BASIS FOR DEVIATION
		These changes do not alter the technical intent of the ERG (1) and therefore are allowable ERG (1) deviations.
2	2	In Substep 2a. changed stop valves to throttle valves due to plant specific design. Added a contingency to place EH pumps in pull to lock if the turbine will not trip. Also changed the contingency for "IF the turbine can NOT be runback" to "IF the turbine throttle valves can NOT be closed", THEN close the MSIVs and MSIBs. Also Substep 2b. was added to ensure the automatic opening of the main generator output breaker to prevent damage to the main turbine generator. Also added Substep 2c. to ensure main steam is automatically isolated to the deaerator. This step was added to prevent an uncontrolled cooldown of the RCS. These changes do not alter the technical intent of the ERG (1) and therefore are allowable ERG (1) deviations.
3	3	Changed "Check" to "VERIFY" per Writer's Guide (2). Changed "running" to "status" on the high level action statement to distinguish the high level step from the substeps. Removed "if necessary" from the turbine-driven pump due to the STP specific turbine-driven logic. STP design starts all four AFW pumps on the same signals therefore the turbine-driven pump should also be running. These changes do not alter the technical intent of the ERG (1).
4-Caution	N/A	Added caution prior to step four to alert the operator of pressurizer level concerns if the RWST is used for emergency boration. This does not change the technical intent of the ERG (1) and therefore is an acceptable addition.

TITLE RESPONSE TO NUCLEAR POWER GENERATION—ATWS

EOP NO. 0POP05-EO-FRS1 REVISION 23

STP EOP STEP NO.	WOG ERG STEP NO.	BASIS FOR DEVIATION
4-Note	N/A	Added note identifying that the Boric Acid Tanks are the preferred suction source for emergency boration. This note added since use of the RWST adds significant complexity to the procedure in the terms of RCS inventory control. Once alignment of the BAT(s) is made then these concerns are mitigated. This does not change the technical intent of the ERG (1) and therefore is an acceptable addition.
4.h - Note	N/A	Added note prior to step 4.h to identify the basis for establishing letdown flow in subsequent step. This does not change the technical intent of the ERG (1) and therefore is an acceptable addition.
4	4	Rewrote substeps to follow the ERG (1) intent for STP specific design. Added use of the RWST for Emergency Boration source per CR # 00-12429. The 190 gpm RWST flowrate added is per CR # 98-14181 (Calc 00-ZE-015)(9). Also added RNO Substep d. 1) which directs the operator to refer to Addendum 1. Addendum 1 gives alternative means of boration. Added gravity feed method of boration per the FSAR (4). To ensure the minimum requirement of TRM 4.1.2.2.d (7) is satisfied, RCP seal flow of 20 gpm is added to the TRM requirement of 30 gpm for a total of 50. This ensures at least 30 gpm is delivered to the RCS. STP has determined that 50 will be used as an operational parameter. Removed the additional 20 gpm flow during use of the RWST for emergency boration since the flow instrument used is downstream of the seal injection tap. The flow rate of 190 GPM will be used for this flow as an operational parameter.

TITLE RESPONSE TO NUCLEAR POWER GENERATION—ATWS

EOP NO. 0POP05-EO-FRS1 REVISION 23

STP EOP STEP NO.	WOG ERG STEP NO.	BASIS FOR DEVIATION

Added steps to delineate proper sequence of equipment operation for start of CCP. Added additional step d and RNO to account for operator actions in the event that RWST method of Emergency Boration is used. Added the expected BAT source flowpaths to substep d. to clarify the intent of the step. Added action to establish charging flow per Addendum 5 to step 4.c.3 RNO in the event that charging flow control valve does not operate correctly. This revision also removed some ambiquity from step 4.g as to the source of the emergency boration source.

Added substep and RNO to step 4 h. in order to accomodate inventory concerns that could develop when using the RWST as a source for emergency boration. The intent of these steps is to either control inventory to maintain pressure control by placing normal letdown, Addendum 6, (or excess letdown, Addendum 7) in service or establish boration from the BAT's to limit RCS makeup rate. Addendum 7 caution uses the design pressure, 150 psig, for when damage could occur to the excess letdown heat exchanger. This pressure, 150 psig, also corresponds to the relief setpoint for the system. Note: The normal operating procedure uses a maximum pressure of 145 psig to provide margin to the design pressure.

Changed the format of step 4 h., to place letdown in service and then control pressurizer level per CR # 02-3683. The use of the control band of 22-85% PZR level was chosen to accomodate inventory concerns during use of the RWST for emergency boration providing a high enough level to allow time to place letdown in service or place the BAT's in service as the suction source.

EOP Step 4 i. equates to ERG Step 4 d. with the RNO changed from ERG RNO which has multiple actions into EOP form of "PERFORM the following:" with two Substeps, one for each action per the Writer's Guide (2).

Obj #3

NOTE: BAT(s) are the preferred emergency boration suction source to limit RCS inventory concerns.

Basis: Using the RWST as the boration source adds more difficulty to the control of RCS inventory due to the additional flow required.

Step 4 INITIATE Emergency Boration Of RCS.

Basis: After control rod trip and insertion functions, boration is the next most direct method of adding negative reactivity.

(sub steps are self-explanatory except;)

- i CHECK Pressurizer Pressure - LESS THAN 2335 PSIG

Basis: Verify RCS pressure is not high to limit boron injection flow.

Exam Bank No.: 2562 **Last used on an NRC exam:** Never

RO Sequence Number: 10

Unit 1 experienced a Steam Generator Tube Rupture at 100 % power.

- Due to RCS leakage, Containment pressure is elevated at 2.0 psig and stable
- Subcooled margin is 46ºF and lowering slowly

Per 0POP05-EO-EO30, Steam Generator Tube Rupture, AFW should be isolated to the ruptured SG after NR S/G Level is GREATER THAN_____(1)_____% and SG isolation should be complete just prior to performing the RCS ______(2)______.

A. (1) 14
(2) depressurization

B. (1) 34
(2) depressurization

C. (1) 14
(2) cooldown

D. (1) 34
(2) cooldown

Answer: C (1) 14
(2) cooldown

Exam Bank No.: 2562 **Source:** Bank **Modified from** N/A

K/A Catalog Number: EPE 038 EA2.01 Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more SGs.

RO Importance: 4.1 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)()

STP Lesson: LOT 504.15 **Objective Number:** 92408

Given a copy of a step from 0POP05-EO-EO30 state/identify how the action is performed and the basis for the action to include the action itself, its purpose and the result.

Reference: 0POP05-EO-EO00 CIP, 0POP05-EO-EO30

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible if student believes that lifting SG Safeties on the Ruptured SG could occur if isolation were to occur before cooldown. The first part is correct.

B: INCORRECT: Plausible if student believes that containemnt pressure warrants using the 34% limitt and that lifting SG Safeties on the Ruptured SG could occur if isolation were to occur before cooldown.

C: CORRECT: Based on containment pressure AFW would be isolated to the Ruptured SG when level is > 14% in either EO00 or EO30. The SG isolation is to be completed prior to the RCS cooldown to prevent depressurization of the SG.

D: INCORRECT: Plausible if student believes that containemnt pressure warrants using the 34% limitt. The second part is complete.

Question Level: H **Question Difficulty** 2

Justification:

Student must analyze conditions and determine proper time for SG isolation.

CONDITIONAL INFORMATION PAGE

RUPTURED SG ISOLATION (all bulleted substeps below are part of this CIP action)

IF a ruptured SG(s) is NOT required to maintain at least one SG available for RCS cooldown, THEN the US or SM **SHALL** direct actions be taken to isolate the ruptured SG(s) as follows:

- SG 1D(2D) ISOLATION

  WHEN at least one motor driven AFW pump is available, THEN ISOLATE main steam to AFW pump 14(24):

  1) PERFORM the following:
     a. RESET SI
     b. IF SI does NOT reset, THEN locally RESET SI. REFER TO 0POP01-ZA-0018A, EMERGENCY OPERATING PROCEDURE GENERIC GUIDANCE.
  2) RESET SG LO-LO level AFW actuations.
  3) ENSURE Permissive Lampbox 5M24-D-4 “C-20 AMSAC BLOCKED” is illuminated.
  4) TRIP turbine-driven AFW pump.
  5) MAINTAIN turbine-driven AFW pump steam inlet valve (AF-MOV-0143) closed until Step 6) completed.
  6) DISPATCH operator to OPEN breaker for AF-MOV-0143, E1D11(E2D11)/5C.
  7) ENSURE turbine-driven AFW pump trip/throttle valve closed (AF-MOV-0514).
  8) CROSS-CONNECT AFW to 1D(2D) SG until SG NR level is GREATER THAN 14%[34%].

- AFW ISOLATION CRITERIA

  WHEN BOTH of the following conditions below are satisfied, THEN ISOLATE AFW flow to the ruptured SG(s):

  - After ruptured SG(s) NR level is GREATER THAN 14%[34%]
  - Before RCS and ruptured SG(s) pressures have been equalized

- SG PORV ISOLATION CRITERIA

  IF the affected SG(s) PORV(s) is open with SG pressure LESS THAN setpoint, THEN manually CLOSE the SG PORV OR dispatch an operator to isolate the SG PORV locally.

MSIV AND MSIB CLOSURE CRITERIA

IF a loss of secondary support systems occurs that impairs the ability of secondary systems to provide a heat sink for the Steam Generators, THEN CLOSE MSIVs and MSIBs. (for example loss of Condenser Availability, C-9) IF any MSIV or MSIB will **NOT** close, THEN dispatch an operator to perform Addendum 6, FAILING AIR TO MSIVS AND MSIBS

MSR ISOLATION CRITERIA

IF any MSIV OR MSIB is not closed AND one of the conditions below exist, THEN ISOLATE MSRs per ADDENDUM 7, ISOLATING MSR COOLDOWN PATH:

- Loss of Offsite power
- Loss of Instrument Air(IA) header pressure LESS THAN 95 PSIG.

CAUTION

At least one SG SHALL be maintained available for RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___3	ISOLATE Flow From Ruptured SG(s):	
___a.	ADJUST ruptured SG(s) PORV controller setpoint to BETWEEN 1260 PSIG AND 1265 PSIG (QDPS PRI/SEC)	
___b.	CHECK ruptured SG(s) PORV controller - IN AUTO	b. PLACE RUPTURED SG(s) PORV controller in AUTO.
___c.	CHECK ruptured SG(s) PORV - CLOSED	c. WHEN ruptured SG(s) pressure LESS THAN 1260 PSIG, THEN: 1) VERIFY SG PORV closed. 2) IF SG PORV is NOT closed, THEN PLACE SG PORV controller in MANUAL and CLOSE SG PORV. 3) IF SG PORV can NOT be closed, THEN DISPATCH operator to isolate PORV: (58 ft IVC) o "1(2)-MS-0021" "S/G 1A(2A) MAIN STEAM" "OUTLET PORV ISOLATION" "VALVE" o "1(2)-MS-0038" "S/G 1B(2B) MAIN STEAM" "OUTLET PORV ISOLATION" "VALVE" o "1(2)-MS-0055" "S/G 1C(2C) MAIN STEAM" "OUTLET PORV ISOLATION" "VALVE" o "1(2)-MS-0072" "S/G 1D(2D) MAIN STEAM" "OUTLET PORV ISOLATION" "VALVE"

Step 3 continued on next page.

CAUTION

IF any ruptured SG is faulted AND intact SGs are available for RCS cooldown, THEN AFW flow should remain isolated to ruptured SG to prevent an uncontrolled RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___4	MONITOR Ruptured SG(s) Level:	
	___a. NR level - GREATER THAN 14% [34%]	a. PERFORM the following: 1) IF SG 1D(2D) is ruptured, THEN ESTABLISH AFW flow to SG 1D(2D) using the AFW cross-connects. 2) MAINTAIN AFW flow to ruptured SG(s). 3) WHEN ruptured SG(s) NR level is GREATER THAN 14% [34%], THEN STOP AFW flow to ruptured SG(s) per Step 4.b. 4) GO TO Step 5. OBSERVE the CAUTION prior to Step 5.
	___b. STOP AFW flow to ruptured SG(s):	
	___1) RESET SI	1) Locally RESET SI. REFER TO 0POP01-ZA-0018A, EMERGENCY OPERATING PROCEDURE GENERIC GUIDANCE.
	___2) RESET SG LO-LO level AFW actuations	
	___3) CLOSE AFW OCIV	3) PLACE AFW pump in PULL TO LOCK.

STEP ACTION/EXPECTED RESPONSE RESPONSE NOT OBTAINED

CAUTION

IF the RCPs are NOT running, THEN the following steps may cause a false 0POP05-EO-F004, INTEGRITY CRITICAL SAFETY FUNCTION STATUS TREE indication in the ruptured loop. Disregard the ruptured loop Tcold indication until after performing Step 31.

NOTE

o IF any SG pressure is rapidly lowering as indicated by illumination of the pressure high rate bistables (2/3) AND low steamline pressure SI has been blocked, THEN a main steam isolation will occur.

o WHEN low steamline pressure SI has been blocked, THEN limiting continuous SG depressurization to LESS THAN 100 psi/min should prevent a main steam isolation.

o Steps 8 through 15 SHALL be continued after initiating the RCS cooldown.

o When the ruptured SG pressure is between two listed values on the table, use the lower value for the required core exit temperature.

____7 INITIATE RCS Cooldown:

Step 7 continued on next page.

NOTE

ADDENDUM 14, DEPRESSURIZATION TERMINATION CRITERIA may be given to the operator to facilitate recognition of the termination criteria during Step 18 depressurization.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___18	DEPRESSURIZE RCS To Minimize Break Flow And Refill Pressurizer:	
	___a. Normal pressurizer spray - AVAILABLE	a. PERFORM the following: 1) ENSURE normal spray valves closed. 2) ESTABLISH pressurizer AUX SPRAY (REFER TO 0POP01-ZA-0018A, EOP Generic Guidance): a) <u>IF</u> AUX SPRAY is unavailable, GO TO Step 19, OBSERVE CAUTIONS and NOTE prior to Step 19. b) GO TO Step 18.b 3) GO TO Step 19, OBSERVE CAUTIONS and NOTE prior to Step 19.
	___b. PLACE group "C" pressurizer heater control switch to PULL TO LOCK	
	___c. PLACE all other pressurizer heater group control switches to OFF	
	___d. INITIATE maximum pressurizer spray	

Step 18 continued on next page.

Exam Bank No.: 3235 **Last used on an NRC exam:** Never

RO Sequence Number: 11

Unit 1 is in MODE 3 and cooling down per 0POP03-ZG-0007, Plant Cooldown.

- RCS pressure is 1900 psig.

An “A” Main Steam Line Break OUTSIDE containment occurs.

- Main Steam pressure is 900 psig and lowering.
- A Main Steam Line Isolation signal is received.

The MSIVs closed because of a _______(1)_______ signal.

Per 0POP05-EO-EO20, Faulted Steam Generator Isolation, if secondary radiation is detected in step 5, CHECK Secondary Radiation, the crew should GO TO _____(1)_____.

A. (1) Steam Pressure High Rate
(2) 0POP05-EO-EO30, Steam Generator Tube Rupture

B. (1) Steam Pressure High Rate
(2) 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant

C. (1) Low Steam Pressure
(2) 0POP05-EO-EO30, Steam Generator Tube Rupture

D. (1) Low Steam Pressure
(2) 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant

Answer: A (1) Steam Pressure High Rate; (2) GO TO 0POP05-EO-EO30, Steam Generator Tube Rupture

Exam Bank No.: 3235 **Source:** New **Modified from** N/A

K/A Catalog Number: APE 040 G2.2.44 Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions

RO Importance: 4.2 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.20 **Objective Number:** 93123

Given a description of plant conditons, PREDICT how the Solid State Protection System will respond.

Reference: 0POP03-ZG-0007 pgs 38-40; 0POP05-EO-EO20 Step 5, LOT201.20 HO1 pg 48

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Below 1985 psig (P-11) 0POP03-ZG-0007 has a step to block Steamline Pressure Lo SI. This unblocks the High Steam Pressure Rate Isolation. Step 5 of 0POP05-EO-EO20 RNOs b and c both require a transition to 0POP05-EO-EO30, step 1 if secondary radiation is detected.

B: INCORRECT. Plausible because there are conditions that send you to 0POP05-EO-EO10 but this situation sends you to 0POP05-EO-EO30.. The first part is correct.

C: INCORRECT. Plausible because prior to blocking this actuation at 1985 this is the signal that would isolate the main steam lines. The second part is correct.

D: INCORRECT. Plausible because prior to blocking this actuation at 1985 this is the signal that would isolate the main steam lines and there are conditions that send you to 0POP05-EO-EO10 but this situation sends you to 0POP05-EO-EO30.

Question Level: H **Question Difficulty** 4

Justification:

Student must analyze cxonditions given in the stem to determine the correct isolation signal and determine the correct procedure.

Initials

5.25 WHEN Pressurizer pressure is less than 1985 psig (P-11), THEN PERFORM the following:

5.25.1 ENSURE RCS Shutdown Margin Concentration (Cb) requirements are met by performing the following:

5.25.1.1 ENSURE one of the following: _____

a. (Normal Cooldown or TS LCO directed Cooldown **with** time) ENSURE the RCS boration to RCS Shutdown Margin Concentration (Cb) requirements of Step 5.7.7 concentration completed. (Ref 2.86)

OR

b. (TS LCO directed Cooldown **AND** time does **NOT** permit obtaining the RCS SDM Boron Concentration of Step 5.25.1.1) ENSURE the RCS boration to RCS Shutdown Margin Concentration (Cb) requirements of Step 5.7.7 are completed, **as soon as possible** AND the RCS Boron Concentration (Cb) meets the requirements of Step 5.20.2.4.

OR

c. (Xenon Credit Cooldown per Step 5.7.8) ENSURE the RCS boration to RCS Shutdown Margin Concentration (Cb) requirements of Form 2, Xenon Credit Shutdown Boration Worksheet, are met.

5.25.1.2 IF **NOT** using Xenon Credit for cooldown, THEN PERFORM 0PSP10-ZG-0003, Shutdown Margin Verification - Modes 3, 4 and 5, for RCS 68°F, xenon free conditions. _____

Initials

NOTE

- The following Step 5.25.2 is performed to wipe the unblock contacts for the PRZR PRESS SI and LO STM LN PRESS SI Train R and Train S block switches three times before taking the switches to the block position. (Ref. 2.117)
- Step 5.25.2 SHALL be completed prior to step 5.25.3 that will block PRZR PRESS SI and LO STM LN PRESS SI.

5.25.2 PERFORM the following to wipe the unblock contacts of the Train R and Train S block switches: (Ref. 2.117)

5.25.2.1 VERIFY the following Status Lights are extinguished:

- "LO STM LN PRESS SI BLOCKED TRAIN R" _____
- "LO STM LN PRESS SI BLOCKED TRAIN S" _____
- "PRZR PRESS SI BLOCKED TRAIN R" ("PRZR SI BLOCKED TRAIN R") _____
- "PRZR PRESS SI BLOCKED TRAIN S" ("PRZR SI BLOCKED TRAIN S") _____

5.25.2.2 TURN the following switches to the UNBLOCK position and RETURN to Mid Position (CP005):

- "PRZR PRESS SI TRAIN R" _____
- "PRZR PRESS SI TRAIN S" _____
- "LO STM LN PRESS SI TRAIN R" _____
- "LO STM LN PRESS SI TRAIN S" _____

5.25.2.3 REPEAT Steps 5.25.2.1 and 5.25.2.2 two more times. _____

5.25.3 VERIFY the following Status Lights are extinguished:

- "LO STM LN PRESS SI BLOCKED TRAIN R" _____
- "LO STM LN PRESS SI BLOCKED TRAIN S" _____
- "PRZR PRESS SI BLOCKED TRAIN R" ("PRZR SI BLOCKED TRAIN R") _____
- "PRZR PRESS SI BLOCKED TRAIN S" ("PRZR SI BLOCKED TRAIN S") _____

Initials

5.25.4 BLOCK Pressurizer pressure safety injection signal by taking both Train R and Train S block switches to the BLOCK position. (CP005)

- "PRZR PRESS SI TRAIN R" _____
- "PRZR PRESS SI TRAIN S" _____

5.25.5 VERIFY both Train R and Train S Pressurizer pressure safety injection block lamp boxes illuminated. (CP005)

- "PRZR PRESS SI BLOCKED TRAIN R" ("PRZR SI BLOCKED TRAIN R") _____
- "PRZR PRESS SI BLOCKED TRAIN S" ("PRZR SI BLOCKED TRAIN S") _____

5.25.6 BLOCK low compensated steam line pressure safety injection signal by taking both Train R and Train S block switches to the BLOCK position. (CP005)

- "LO STM LN PRESS SI TRAIN R" _____
- "LO STM LN PRESS SI TRAIN S" _____

5.25.7 VERIFY both Train R and Train S low compensated steam line pressure injection block lamp boxes illuminated. (CP005)

- "LO STM LN PRESS SI BLOCKED TRAIN R" _____
- "LO STM LN PRESS SI BLOCKED TRAIN S" _____

D. Actuates Containment Isolation Phase B

E. Actuates Containment Spray (valves open, but ESF Sequencer controls pumps so an SI signal must also exist for the pumps to start)

3.8.4 High steam pressure rate steam line isolation

A. Provides some protection for steamline break at times when low steamline pressure safety injection is blocked (<P11)

B. > 100 psi drop with a 50 second time constant; 2/3 channels in 1/4 SGs and not blocked

C. Shuts all loop steamline stop valves

D. Automatically blocked above P-11 setpoint

E. Manually unblocked below P-11 setpoint

F. Alarms and annunciators (CP-005)

3.8.5 Low compensated steamline pressure

A. Indicative of a steam break accident

B. 735 psig (rate sensitive) 2/3 channels on any one steam line and not blocked (P-11)

C. Can be manually blocked when < P-11

1. 2 switches located on CP-005

2. Block automatically clears above P-11

3. Can be manually reset at anytime

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____5	CHECK Secondary Radiation:	
____a.	PERFORM the following:	
	____1) RESET SI	1) Locally RESET SI. REFER TO 0POP01-ZA-0018A, EMERGENCY OPERATING PROCEDURE GENERIC GUIDANCE.

	____2) RESET SG LO-LO level AFW actuations	
	____3) RESET SG blowdown and sample isolations	
	____4) NOTIFY Chemistry to sample all SGs hourly for activity	

NOTE

<u>IF</u> secondary systems listed below have been isolated, <u>THEN</u> verification for NORMAL radiation level is performed by checking levels prior to isolation.

____b.	CHECK the following: o Main steamline radiation - NORMAL o SG blowdown radiation - NORMAL o CARS pump radiation - NORMAL	b. GO TO 0POP05-EO-EO30, STEAM GENERATOR TUBE RUPTURE, Step 1.

____c.	<u>WHEN</u> SG sample results are received, <u>THEN</u> VERIFY SG sample activity - NORMAL	c. GO TO 0POP05-EO-EO30, STEAM GENERATOR TUBE RUPTURE, Step 1.

Exam Bank No.: 3236 **Last used on an NRC exam:** Never

RO Sequence Number: 12

The overall mitigative strategy of 0POP05-EO-EC21, Uncontrolled Depressurization of All Steam Generators, includes ….

A. establishing 100 gpm AFW flow to each SG to conserve AFWST inventory.

B. establishing 100 gpm AFW flow to each SG to minimize cooldown rate.

C. securing all AFW flow to the SGs to conserve AFWST inventory.

D. securing all AFW flow to the SGs to minimize cooldown rate.

Answer: B establishing 100 gpm AFW flow to each SG to minimize cooldown rate.

Exam Bank No.: 3236 **Source:** New **Modified from** N/A

K/A Catalog Number: W/E12 EK2.2 Knowledge of the interrelations between the (Uncontrolled Depressurization of All Steam Generators) 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

RO Importance: 3.6 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(4)

STP Lesson: LOT 504.14 **Objective Number:** 33221

Given a copy of a step from 0POP05-EO-EC21, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose and the result.

Reference: LOT 504.14 PPT slide 41 and 0POP05-EO-EC21, Step 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because all four SGs are dumpimg steam into the contaiment and lowering flow would reduce the amount of inventory being released into containment over a given time period.

B: CORRECT. 0POP05-EO-EC21 limits AFW flow to a maximum of 100 gpm to each SG to minimize cooldown rate and not stopping flow prevents SG dryout.

C: INCORRECT. Plausiible because because stopping flow would further reduce the cooldown rate but it is not what is done by procedure and it would conserve AFWST inventory but it is not the reason for lowering flow.

D: INCORRECT. Plausiible because stopping flow would further lower the cooldown rate but is not what is done per procedure.

Question Level: F **Question Difficulty** 2

Justification:

Student must know overall strategy and procedure requirements of EC21.

EC-21

What is the basis for minimizing AFW flow? (e.g., 100 gpm / SG)

- Limit RCS cooldown
- Prevent SG tube dryout
- Limit mass release through the fault

CAUTION

IF total AFW flow is LESS THAN 576 GPM due to operator action, THEN 0POP05-EO-FRH1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, SHALL NOT be performed.

NOTE

Shutdown margin should be monitored during RCS cooldown.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___2	MAINTAIN AFW Flow To Minimize RCS Cooldown And Prevent SG Dryout:	
	___a. CHECK cooldown rate in RCS cold legs - LESS THAN 100°F/HR	a. PERFORM the following: 1) LOWER AFW flow to 100 gpm to each SG. 2) GO TO Step 2.c.
	___b. CHECK NR level in all SGs:	
	___1) LESS THAN 50%	1) CONTROL AFW flow to maintain NR level LESS THAN 50% in all SGs.
	___2) GREATER THAN 14% [34%]	2) CONTROL AFW flow to 100 gpm to all SGs with NR levels LESS THAN 14% [34%].
	___c. CHECK RCS hot leg temperatures - STABLE OR LOWERING	c. ESTABLISH stable RCS hot leg temperatures by: o Controlling AFW flow. OR o Dumping steam.

Exam Bank No.: 3205 **Last used on an NRC exam:** Never

RO Sequence Number: 13

Unit 1 was at 100% power when:

- A reactor trip due to a complete loss of Main Feedwater occurs.
- Only AFW Pump 14 could be started following the SG Lo-Lo Level actuation.
- After the RO performs initial Immediate Operator Actions, the Unit Supervisor directs cross-connecting AFW to the remaining S/Gs per 0POP01-ZA-0018A, Emergency Operating Procedure Generic Guidance.

The MAXIMUM total AFW Pump 14 flow should be limited to less than ______ gpm.

A. 576

B. 605

C. 640

D. 675

Answer: D 675

Exam Bank No.: 3205 **Source:** New **Modified from** N/A

K/A Catalog Number: APE 054 AA1.01 Ability to operate and/or monitor the following as they apply to the Loss of Main Feedwater (MFW): AFW controls, including the use of alternate AFW sources

RO Importance: 4.5 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(8)

STP Lesson: LOT 202.28 **Objective Number:** 43808

STATE the function and design bases of the AFWS including major components, instrumentation, and sources of water.

Reference: LOT202.28 PPT slides 19,32 0POP01-ZA-0018A Add. 5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because 576 gpm is the minimum acceptable flow in EOPs for heat removal.

B: INCORRECT. Plausible because 605 gpm is the nominal flow QDPS throttles to after an AFW actuation.

C: INCORRECT. Plausible because 640 gpm is the upper flow limit that QDPS allows on the AFW actuation.

D: CORRECT. 675 gpm is the flow referenced in Normal, Off-normal and Emegency Operating Procedures and is the runout flow for a single AFW pump.

Question Level: F **Question Difficulty** 2

Justification:

Student must know procedural guidance and flow limits for the AFW Pumps

Addendum 5	Cross Connecting AFW	Page 1 of 1

1. ENSURE SI is reset. _____

2. ENSURE SG LO-LO level AFW actuations are reset. _____

> NOTE
>
> 5M24 D-4, "C20 AMSAC BLOCKED" illuminates when the 360 second time delay for the AMSAC disarming relay times out. Computer point AMJR0511A would indicate "BLKD"

3. VERIFY 5M24 D-4, "C20 AMSAC BLOCKED" illuminated. _____

4. CLOSE AFW REG valve for the SG(s) that will be fed via the cross-connect(s). _____

5. ENSURE AFW OCIV OPEN for required SG(s). _____

6. ESTABLISH AFW flowpath to the required SG(s):

   - OPEN "X CONN" on the SG with the running AFW pump. _____

   - OPEN "X CONN" to the SG(s) to be supplied. _____

7. THROTTLE OPEN AFW REG valve(s) while maintaining total AFW flow from the feeding AFW pump to LESS THAN 675 gpm. _____

Obj. 1

Design basis

- Designed to operate automatically within 1 minute following an actuation signal to:
  - Deliver 500 gpm to at least 1 S/G on FW rupture or Steam Line Break.
  - Deliver 500 gpm to each of at least 2 S/G's on a loss of FW or a LOOP.
  - Able to provide this flow from a pressure range of 1378 psia to 100 psia.
  - AFW Pump runout is 675 gpm

Obj. 2

AFW Trains

- Four Independent AFW Trains, one for each S/G
  - Trains A, B, & C are each powered from their respective 480 & 4160 ESF Buses
  - Train D is DC & Main Steam Powered
  - Pumps are rated at 600 gpm at 1394 psig
  - Single speed (3600) 11 stage counter flow
  - Arranged in C B A D sequence
  - Closed Doors are required for Operability

Exam Bank No.: 3132 **Last used on an NRC exam:** Never

RO Sequence Number: 14

Given the following:

- A Station Blackout has occurred in Unit 2.
- Operators have entered 0POP05-EO-EC00, Loss Of All AC Power and are directed to perform Addendum 4, VITAL BUS DC MONITORING.

Which of the following plant loads will continue to be supplied power from 125VDC **E2B11** Switchboard?

A. Train ‘A’ Load Sequencer

B. Pressurizer PORV PCV-0656A

C. Field flashing to Standby Diesel Generator 21

D. Turbine Driven Auxiliary Feedwater Pump 24

Answer: B Pressurizer PORV PCV-0656A

Exam Bank No.: 3132 **Source:** Modified **Modified from** 3014

K/A Catalog Number: 055 EA2.04 Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available

RO Importance: 3.7 **Tier:** **Group/Category:** **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.37 **Objective Number:** 92049

List the typical loads on the Class 1E 125 VDC System.

Reference: LOT 201.37 Power Point slide 49 and 0POP04-DJ-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as ESF Load Sequencer ‘A’ is powered from DC Switchboard E2A11.

B: CORRECT. Pressurizer PORV PCV-656A is powered from Switchboard E2B11 but has the 'A' designator.

C: INCORRECT. Plausible as field flashing to ESF Diesel Generator 21 is powered from DC Switchboard E2A11.

D: INCORRECT. Plausible as TDAFW Pump 14 is powered from DC Switchboard E2D11.

Question Level: F **Question Difficulty** 2

Justification:

The student must recall specific loads on 125VDC switchboards

SOURCE FOR RO #14

Exam Bank No.: 3014 **Last used on an NRC exam:** 2021

RO Sequence Number: 48

A loss of 125 VDC power to the E1D11 switchboard will result in which of the following?

A. Loss of field flash to the "B" ESF Diesel Generator.

B. Loss of control power to one train of reactor trip breakers.

C. Loss of power to pressurizer PORV (PCV-655A).

D. Loss of the Turbine Driven Auxiliary Feedwater Pump.

Answer: D Loss of the Turbine Driven Auxiliary Feedwater Pump

Exam Bank No.: 3014 **Source:** New **Modified from**

K/A Catalog Number: 063 K.2.01 Knowledge of bus power supplies to the following: Major DC loads.

RO Importance: 2.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.37 **Objective Number:** 11

List the typical loads on the Class 1E 125 VDC System.

Reference: LOT 201.37

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as this is powered from E1B11.
B: INCORRECT: Plausible as either train of reactor trip breakers is from E1B11 or E1A11.
C: INCORRECT: Plausible as this is powered from E1A11.
D: CORRECT: The E1D11 switchboard provides power to the MOVs necessary to start and control steam flow to the TDAFWP.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall loads on DC switchboards.

125V DC Bus E1B11(E2B11)			
CUBICLE NUMBER	DESCRIPTION	REFERENCE DRAWINGS	FAILURE MODE
1B	"125V BATT E1B11(E2B11) TO 125V DC SWBD E1B11(E2B11)"	9-E-DJAC-01	N/A
2A	"BATT CHGR E1B11(E2B11)-1 TO 125V DC SWBD E1B11(E2B11)"	9-E-DJAC-01	N/A
3A	"BATT CHGR E1B11(E2B11)-2 TO 125V DC SWBD E1B11(E2B11)"	9-E-DJAC-01	N/A
3B	"TO 125V DC DIST PNL PL039B EAB 35' EL"	9-E-DJAC-01 9-E-DJAE-01	LOSS OF POWER TO PL039B
3C	"STBY DIESEL GEN 12(22) CONT PNL ZLP103"	9-E-DJAC-01 9-E-DG01-01	LOSS OF FLASHING POWER TO DIESEL
4B	"480V ESF LC E1B(E2B) DC CONTROL PWR"	9-E-DJAC-01 9-E-PL05-01	LOSS OF CONTROL PWR TO BKRs ON 480V LC E1B(E2B)
4C	"TO 125V DC DIST PNL PL139B STBY D/G 12(22) RM 35' EL"	9-E-DJAC-01 9-E-DJAF-01	LOSS OF PWR DG CONTROL PNLS
4D	"4.16KV BUS E1B(E2B) DC CONTROL PWR"	9-E-DJAC-01 9-E-PK04-02	LOSS OF CONTROL PWR TO BKRs ON 4.16KV ESF BUS E1B(E2B)
5B	"ESF LOAD SEQUENCER CABINET B-ZLP802"	9-E-DJAC-01 4176-01020	DISABLES SEQUENCER
5C	"TRAIN B RX TRIP SWGR CONTROL PWR"	9-E-DJAC-01 0386-00189 9-E-SP18-03	DISABLES RTS 1C & BYS 2C SHUNT TRIP CKTs REDUNDANT RTR 1B/BYR 2B NO IMMEDIATE EFFECT
5D	"CH III INST/CONT PWR NSSS INVERTER DC PWR"	9-E-DJAC-01 9-E-VAAB-01	ALTERNATE SOURCE SPLY PWR
7B	"B/U BKR-PRZR PORV 0656A 1(2)-RC-PCV-0656A"	9-E-DJAC-01	DISABLES PORV (TRAIN A REDUNDANT)
7C	PRZR PORV 0656A 1(2)-RC-PCV-0656A"	9-E-DJAC-01 9-E-RC13-01	DISABLES PORV (TRAIN A REDUNDANT)

B-Train Loads

- Reactor Trip Breaker Control Power
- ESF DG “B” Field Flash
- PL-139B (DG Control Panels)
- PL-39B (EAB)
- 4160 & 480 VAC (ESF) breaker control power
- Load Sequencer B
- Inverter 1203
- Pzr PORV 656A

Obj. 11

Exam Bank No.: 3133 **Last used on an NRC exam:** Never

RO Sequence Number: 15

Unit 1 is at 100% power.

- PR NI-41 failed four days ago.
- All actions of 0POP04-NI-0001, Nuclear Instrumentation Malfunction and all required Technical Specification actions have been taken.

A loss of power to Channel II (ESF Class 1E Vital 120V Instrumentation) occurs.

The affected panel is ___(1)____.

The Reactor Operator should ___(2)___.

A. (1) DP-1202
   (2) verify reactor trip in accordance with 0POP05-EO-EO00, Reactor Trip Or Safety Injection

B. (1) DP-1202
   (2) bypass the affected channel in accordance with 0POP04-NI-0001, Nuclear Instrumentation Malfunction

C. (1) DP-1203
   (2) verify reactor trip in accordance with 0POP05-EO-EO00, Reactor Trip Or Safety Injection

D. (1) DP-1203
   (2) bypass the affected channel in accordance with 0POP04-NI-0001, Nuclear Instrumentation Malfunction

Answer: A (1) DP-1202
(2) verify reactor trip in accordance with 0POP05-EO-EO00, Reactor Trip or Safety Injection

Exam Bank No.: 3133 **Source:** New **Modified from** N/A

K/A Catalog Number: 057 G2.4.1 Loss of Vital AC Instrument Bus: Knowledge of EOP entry conditions and immediate action steps

RO Importance: 4.6 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 504.05 **Objective Number:** 30309

Given a list of conditions, STATE/IDENTIFY which ones would require a reactor trip or safety injection.

Reference: 0POP04-NI-0001, Addendum 3, 0POP04-VA-0001, Addendum 12, TS 3.3.1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Channel II is for ‘D’ train equipment and is powered from DP-1202. PR NI-42 is powered from DP-1202. NI-41 has been out of services for 4 days as stated in the stem so per the off-normal and Tech Specs the channel would be in the tripped condition. The loss of power to NI-42 would cause it to go to a tripped condition and the reactor will trip on 2/4 PR channels in the tripped condition. The required action is to verify reactor trip IAW 0POP05-EO-E0OO, Reactor Trip Or Safety Injection

B: INCORRECT. Plausible because NI-41 initially would have been placed in a bypass condition when performing 0POP04-NI-0001 but Tech Specs and the offnormal require that the channel be placed in a tripped conditon within 72 hours. DP-1202 is the correct power supply which powers NI-42.

C: INCORRECT. Plausible if the student confuses power supplies and channel numbers.

D: INCORRECT. Plausible if the student confuses power supplies and channel numbers and that the reactor will not trip.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions and determine required actions.

STEP		ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
______	15.0	**PERFORM the following within 72 hours of the Power Range channel failure:**	
______	a.	**CHECK the OTDT Reactor Trip bistable for the failed channel - BYPASSED**	**a.** GO TO Addendum 3 Step 15.0c. ---
______	b.	**NOTIFY I&C to remove the OTDT Reactor Trip bistable for the failed channel from bypass. REFER TO Addendum 7**	
______	c.	**ENSURE all individual function bypass switches in NORMAL {CP011}**	
______	d.	**ENSURE “BYPASS ENABLE” keyswitch in NORMAL {CP011}**	
______	e.	**ENSURE key from the “BYPASS ENABLE” keyswitch is REMOVED {CP011}**	
______	f.	**TRIP the protective bistables for the failed channel per Addendum 5**	
______	g.	**REMOVE the control power fuses on the affected power range drawer {CP011}**	
______	h.	**VERIFY the bistable monitoring lights listed in Addendum 6 for the failed channel are illuminated**	---

Channel II NI's NI-0042, NI-0032, and NI-0036 fail

Loss of Power to QDPS APC-D2, which results in loss of the following control functions:

- Steam Generator 1D(2D) PORV, PV-7441 (Manual control available using Addendum 9)
- AFW Pump 14(24) Flow Controller, FT-7526
- **(UNIT 1 ONLY)** With Control Rods in Auto will result in Control Rod stepping in (insertion)

Loss of Steam Pressure Steam Flow Compensation for Loops 1, 2, 3, and 4.

Loss of Feedwater Flow Signal for Loop 1, 2, 3, and 4.

OPEN Permissive for RHR Pump Suction Valves is satisfied for RH-MOV-0061C and RH-MOV-0060A.

Loss of one Auctioneered Power Supply to SSPS Logic S (other from DP1204).

Disables Pressurizer PORV PCV-0655A actuation on COMS.

Permissive P-13 Train D disabled.

Failure of Pressurizer Level LT-0466 on Loss of Power to QDPS APC-D2

The controllers for the following valves may be affected:

- Feedwater Control Valve, FW-FCV-0551
- Feedwater Control Valve, FW-FCV-0552
- Feedwater Control Valve, FW-FCV-0553
- Feedwater Control Valve, FW-FCV-0554
- SG A Feedwater Bypass Flow Control Valve, FW-FV-7151
- SG B Feedwater Bypass Flow Control Valve, FW-FV-7152
- SG C Feedwater Bypass Flow Control Valve, FW-FV-7153
- SG D Feedwater Bypass Flow Control Valve, FW-FV-7154
- Pressurizer PORV, PCV-0656A, if PT-0456 selected
- Pressurizer PORV, PCV-0655A (Auctioneered low WR Thot for COMS. If COMS is armed, PORV-0655A will **NOT** open because bistable energizes to actuate)

The following annunciators alarm:

•	Lampbox 1M02	Window C-1	"RWST LEVEL HI/LO"
•	Lampbox 1M02	Window D-1	"RWST LO-LO/EMPTY"
•	Lampbox 4M08	Window A-6	"PRZR LEVEL HI RX TRIP ALERT"
•	Lampbox 4M08	Window A-7	"PRZR PRESS HI RX TRIP ALERT"
•	Lampbox 4M08	Window B-6	"PRZR LEVEL HI"

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Chapter 16 in the Updated Final Safety Analysis Report (UFSAR).

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit at a frequency in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train such that both trains are verified at a frequency in accordance with the Surveillance Frequency Control Program and one channel per function such that all channels are verified at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
7. Extended Range, Neutron Flux	2	0	2	3, 4, 5	5
8. Overtemperature ΔT	4	2	3	1, 2	6
9. Overpower ΔT	4	2	3	1, 2	6
10. Pressurizer Pressure -- Low (Interlocked with P-7)	4	2	3	1	6
11. Pressurizer Pressure-- High	4	2	3	1, 2	6
12. Pressurizer Water Level--High (Interlocked with P-7)	4	2	3	1	6

TABLE NOTATIONS

*When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

##Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

###Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. For Functional Units with installed bypass test capability,

Note: A channel may be bypassed for up to 12 hours for surveillance testing per Specification 4.3.1.1, provided no more than one channel is in bypass at any time.

1. The inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours, and

2. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

b. For Functional Units with no installed bypass test capability,

1. The inoperable channel is placed in the tripped condition within 72 hours, and

2. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 12 hours for surveillance testing of other channels per Specification 4.3.1.1, and

3. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

Exam Bank No.: 3135 **Last used on an NRC exam:** Never

RO Sequence Number: 16

Unit 1 is at 100% power with Battery Charger E1A11-1 in service when:

- '480V LC E1A1 TRBL' alarm occurs.
- 480V Load Center E1A1 Bus Volts is 0 VAC.
- Annunciator '125 VDC SYSTEM E1A11 TRBL' alarms.
- E1A11 Bus Voltage is 125 VDC.

With these conditions, Bus E1A11 is being powered from the ___(1)___.

Subsequently, Bus E1A11 DE-ENERGIZES. Per 0POP04-DJ-0001, Loss of Class 1E 125VDC Power, the breaker supplying RCFC 11A _______(2)_______.

A. (1) Standby Battery Charger
(2) must be operated locally

B. (1) Standby Battery Charger
(2) can still be operated from the control room

C. (1) Class 1E Battery
(2) must be operated locally

D. (1) Class 1E Battery
(2) can still be operated from the control room

Answer: C (1) Class 1E battery
(2) must be operated locally

Exam Bank No.: 3135 **Source:** New **Modified from** N/A

K/A Catalog Number: 058 AA2.01 Ability to determine and interpret the following as they apply to the Loss of DC Power: that a loss of dc power has occurred; verification that substitute power sources have come on line

RO Importance: 3.7 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.37 **Objective Number:** 92050

DRAW and LABEL a basic one-line diagram of the Class 1E 125VDC System showing normal breaker alignment and its interconnections with other electrical systems.

Reference: 0POP04-DJ-0001 Pg 51, LOT202.33 SL 19, LOT201.37 SL 11

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as Non-Class 1E DC systems at the plant with both battery chargers in service and parallelled. The 2nd half is correct.

B: INCORRECT: Plausible as Non-Class 1E DC systems at the plant with both battery chargers in service and parallelled and if the student does not recall that the control power for the RCFC breaker is from Class 1# 125VDC.

C: CORRECT: The Class 1E battery will provide power upon the loss of the in-service battery charger. The standby battery charger must be manually placed in service. With a loss of Class 1E 125VDC power, the associated 480V LC breakers cannot be operated from the control room as their control power comes from Class 1E 125VDC.

D: INCORRECT: Plausible as the Class 1E battery will provide power and if the student and if the student does not recall that the control power for the RCFC breaker is from Class 1# 125VDC.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to determine what has occurred in the 125 VDC electrical system based on the given conditions.

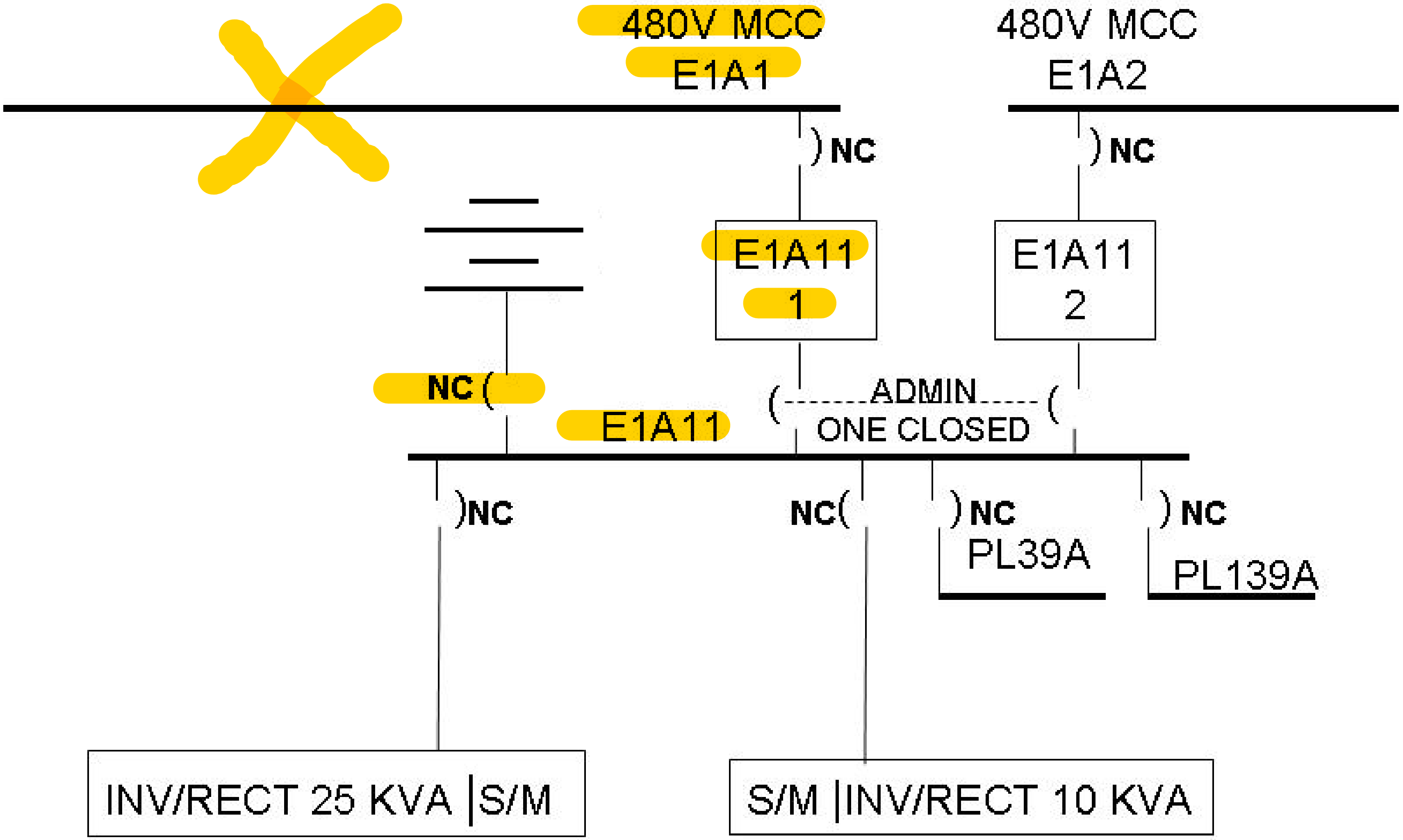
Addendum 5	Train A Class 1E 125 VDC Load/Failure Mode List	Addendum 5 Page 1 of 20

125V DC Bus E1A11(E2A11)			
CUBICLE NUMBER	DESCRIPTION	REFERENCE DRAWINGS	FAILURE MODE
1B	"125V BATT E1A11(E2A11) TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
2A	"BATT CHGR E1A11(E2A11)-2 TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
3A	"BATT CHGR E1A11(E2A11)-1 TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
3B	"TO 125V DC DIST PNL PL039A EAB 10' EL"	9-E-DJAA-01 9-E-DJAE-01	LOSS OF POWER TO PL039A
3C	"STBY DIESEL GEN 11(21) CONT PNL ZLP101"	9-E-DJAA-01 9-E-DG01-01	LOSS OF FLASHING POWER TO DIESEL
4A	"TRAIN A RX TRIP SWGR CONTROL PWR"	9-E-DJAA-01 0386-00189 9-E-SP18-03	DISABLES RTR 1B & BYR 2B SHUNT TRIP CKTS – REDUNDANT RTS 1C OR BYS 2C. NO IMMEDIATE EFFECT
4B	"TO 125V DC DIST PNL PL139A STBY D/G 11(21) RM 35' EL"	9-E-DJAA-01 9-E-DJAF-01	LOSS OF POWER TO DG CONTROL PANELS
4D	"4.16KV BUS E1A(E2A) DC CONTROL PWR"	9-E-DJAA-01 9-E-PK04-02	LOSS OF CONTROL PWR TO BKRs ON 4.16KV ESF BUS E1A(E2A)
5B	"480V ESF LC E1A(E2A) DC CONTROL PWR"	9-E-DJAA-01 9-E-PL01-01	LOSS OF CONTROL POWER TO BREAKERS ON 480V LC E1A(E2A)
5C	"ESF LOAD SEQUENCER CABINET A-ZLP801"	9-E-DJAA-01 4176-01020	DISABLES SEQUENCER
5D	"CH I INST/CONT PWR TMI INVERTER DC PWR"	9-E-DJAA-01 9-E-VAAA-01	ALTERNATE SOURCE SUPPLIES POWER
5E	"CH I INST/CONT PWR NSSS INVERTER DC PWR"	9-E-DJAA-01 9-E-VAAA-01	ALTERNATE SOURCE SUPPLIES POWER
7B	"B/U BKR-PRZR PORV 0655A 1(2)-RC-PCV-0655A"	9-E-DJAA-01	DISABLES PORV (TRAIN B REDUNDANT)
7C	"PRZR PORV 0655A 1(2)-RC-PCV-0655A"	9-E-DJAA-01 9-E-RC13-01	DISABLES PORV (TRAIN B REDUNDANT)

Reactor Containment Fan Coolers

- Flow path
  - Suction on ring duct
  - Discharge at RCB lower level
- With containment spray system, RCFCs makeup ESF heat removal system for the RCB.
- RCFC fan motors - powered from 480V Class 1E load centers.
  - Fan 11A – LC E1A, Fan 12A – LC E1A
  - Fan 11B – LC E1B, Fan 12B – LC E1B
  - Fan 11C – LC E1C, Fan 12C – LC E1C

125 VDC Train A Channel I



Obj. 2

LOT201.37 SL.11

Exam Bank No.: 3136 **Last used on an NRC exam:** Never

RO Sequence Number: 17

Unit 1 is at 100% power.

- ‘A’ Train ECW/CCW Mode Select Switch is selected to RUN.
- ‘B’ Train ECW/CCW Mode Select Switch is selected for STANDBY, with ECW Pump 1B NOT running.
- ‘C’ ECW/CCW Mode Select Switch is selected to OFF with ECW Pump 1C running and CCW Pump 1C secured.

Subsequently, CCW Pump 1A trips on overcurrent.

With no operator action, one minute later, _____(1)_____ **ECW** Pumps are running because _____(2)______.

A. (1) only 1A and 1C
   (2) loss of the CCW pump only provides an Auto start of the Standby CCW Pump

B. (1) only 1A and 1C
   (2) pressure on both ECW headers remained greater than 30 psig

C. (1) ALL
   (2) CCW common header pressure dropped below 76 psig

D. (1) ALL
   (2) ECW Pump 1B Auto started because CCW Pump 1B breaker closed

Answer: C (1)ALL
(2)CCW common header pressure dropped below 76 psig

Exam Bank No.: 3136 **Source:** New **Modified from** N/A

K/A Catalog Number: 062 AK3.04 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Effect on the nuclear service water discharge flow header of a loss of CCW

RO Importance: **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(4)

STP Lesson: LOT 201.13 **Objective Number:** 95303

GIVEN a plant or system condition, PREDICT the operation of the Essential Cooling Water System.

Reference: LOT 201.13 PPT slide 40; LOT 201.12 PPT slide 22

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because it is reasonable to think that the B ECW Pump would not be affected by a CCW pump trip but the ECW Pump is started to provide a heat sink for the running CCW train.

B: INCORRECT. Plausible because the ECW Pump Auto starts on 2 ECW discharge headers < 30 psig and ECW pressures are still normal.

C: CORRECT. A low CCW header pressure (76 psig) on the train selected to RUN automatically starts the standby Train ‘B’ ECW and CCW pumps. The trip of CCW pump A would result in the CCW A discharge header pressure lowering to essentially suction header pressure which would only be the head provided by the surge tank.

D: INCORRECT. Plausible because another feasible way to auto start the ECW pump at the same time as auto starting the CCW pump would be to interlock it with the associated CCW pump breaker. This concept is used with many ventilation fans and associated pumps.

Question Level: H **Question Difficulty** 3

Justification:

The student must analyze plant conditions and determine the correct system response.

Mode Selector Switches (3)

- **Three position switches (RUN/STANDBY/OFF) on CP-001 and -002**

  A. **Run- Starts CCW pump and that train ECW pump if an SI or LOOP signal is not present**

  B. **Standby- Will automatically start CCW pump and that train's ECW pump after 15 second time delay if one of the following conditions occur and a SI signal or loss of offsite power does not exist**

    a. **Low ECW pressure in the other two trains (30 PSIG)**

    b. **Low CCW common discharge header pressure (76 PSIG)**

- **If two of three switches are in the "OFF" position, annunciator "standby train not selected" is actuated on CP-002**

- **SI and loop signals bypass these switches to directly control the CCW pumps**

ECW PUMPS

- Transfer switches located in the EAB Switchgear Rooms on ZLP-653, 654, 655; Trains A,B,C
- Remote: allows auto operation from the ECW/CCW Train selector switches and sequencer
- Auto start:
  - SI starts the pump after 25 sec. sequencer delay
  - Loss of power (LOOP) or SI coincident with LOOP starts the pump 25 sec. after the DG output is connected to ESF bus
  - Low CCW header pressure, at 76 psig dec.
  - Low pressure on the other two trains of ECW, at 30 psig dec. (15 sec TD added for CC/ECW Lo pressure auto start)

Exam Bank No.: 3237 **Last used on an NRC exam:** Never

RO Sequence Number: 18

A Large Break LOCA has occurred in the unit.

- The crew is currently performing 0POP05-EO-EC11, Loss of Emergency Coolant Recirculation.

If RWST level drops below ____(1)____ gallons, the crew should **STOP** _______(2)_______ taking suction from the RWST.

A. (1) 32,500
(2) ALL pumps

B. (1) 32,500
(2) ONLY Centrifugal Charging Pumps

C. (1) 75,000
(2) ALL pumps

D. (1) 75,000
(2) ONLY Centrifugal Charging Pumps

Answer: A (1) 32,500 (2) ALL pumps

Exam Bank No.: 3237 **Source:** New **Modified from**

K/A Catalog Number: W/E11 EK1.1 Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation): Components, capacity, and function of emergency systems

RO Importance: 3.7 **Tier:** 1 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(8)

STP Lesson: LOT 504.27 **Objective Number:** 30127

DESCRIBE the readings which confirm that a Safety Injection or Containment Spray pump should be stopped.

Reference: 0POP05-EO-EC11, Steps 4 and 5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: Per 0POP05-EO-EC11, if RWST level drops below 32,500 gallons, ALL pumps taking a suction from the RWST will be stopped.

B: INCORRECT: Plausible as this is the right level value, and if the student believes that reducing flow significantly rather than terminating all flow is more conservative.

C: INCORRECT: Plausible as this is the RWST value for entering 0POP05-EO-ES13, Transfer to Cold Leg Recirculation and ALL pumps are secured.

D: INCORRECT: Plausible as this is the RWST value for entering 0POP05-EO-ES13, Transfer to Cold Leg Recirculation and if the student believes that reducing flow significantly rather than terminating all flow is more conservative.

Question Level: F **Question Difficulty** 3

Justification:

The student must assess plant conditions and make a determination of proper pump operation.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____3	VERIFY RCFC Status:	
	____a. RCFCs - RUNNING	a. Manually START RCFCs
	____b. Cooling water - TRANSFERRED TO CCW	b. PERFORM the following: 1) IF RCFC inlet temperatures are GREATER THAN 116°F OR Containment Pressure was GREATER THAN OR EQUAL TO 3 psig, THEN CONTACT the TSC prior to transferring cooling. 2) IF RCFC inlet temperatures are LESS THAN OR EQUAL TO 116°F AND Containment Pressure has remained LESS THAN 3 psig, THEN manually TRANSFER cooling to CCW.
____4	MONITOR RWST Level - GREATER THAN 32,500 GALLONS (6%)	GO TO Step 29

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___26	**MONITOR If RCPs Should Be Stopped:**	
___a.	CHECK the following: o Number 1 seal DP - LESS THAN 220 PSID OR o Number 1 seal leakoff flow - LESS THAN required flow per ADDENDUM 9, REACTOR COOLANT PUMP SEAL LEAKOFF VALUES	a. GO TO Step 27.
___b.	STOP affected RCP(s)	
___27	**CHECK RCS Temperature - GREATER THAN 200°F**	GO TO Step 37.
___28	**CHECK RWST Level - LESS THAN 32,500 GALLONS (6%)**	RETURN TO Step 1.
___29	**STOP Pumps Taking Suction From RWST And PLACE In PULL TO LOCK:** o LHSI pumps o HHSI pumps o Charging pumps o Containment spray pumps	

Step 4 MONITOR RWST Level - GREATER THAN 32,500 GALLONS (6%)

BASIS:

Determines actions necessary based on water inventory in RWST. With adequate RWST inventory the operators continue on with steps to minimize RWST outflow by stopping CS and reducing SI flowrate. Without adequate RWST inventory, the operators must secure (protect) the pumps taking suction from it.

If < 32,500 gallons,

- Jumps to Step 29 to secure all pumps taking suction on the RWST and attempt Alternate makeup via CCP.

STEP 29

Exam Bank No.: 3138 **Last used on an NRC exam:** Never

RO Sequence Number: 19

Unit 1 is at 70% power.

- Rod Control is in AUTO.
- S/G “A” PORV opens due to the setpoint failing low.
- Rods begin stepping out.
- The crew manually closes S/G “A” PORV and power is stable.

Subsequently,

- The Tavg-Tref deviation is +1ºF.
- Control rods are stepping out at 6 STEPS PER MIN.
- NO operator actions have been taken.

The RO should ___(1)___.

In accordance with 0POP04-RS-0001, Control Rod Malfunction. if rod motion does NOT stop when Rod Control is placed in MANUAL, then the RO should ____(2)____.

A. (1) place Rod Control in MANUAL
   (2) place Rod Control in CB-A. IF rods continue to move, THEN trip the reactor

B. (1) place Rod Control in MANUAL
   (2) trip the reactor

C. (1) place Rod Control in MANUAL when the Tavg-Tref deviation is +1.5°F
   (2) place Rod Control in CB-A. IF rods continue to move, THEN trip the reactor

D. (1) place Rod Control in MANUAL when the Tavg-Tref deviation is +1.5°F
   (2) trip the reactor

Answer: B (1) place Rod Control in MANUAL (2) trip the reactor and perform the immediate actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Exam Bank No.: 3138 **Source:** Modified **Modified from** 2913

K/A Catalog Number: 001 AK2.06 Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: T-ave./ref. deviation meter

RO Importance: 3.0 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(6)

STP Lesson: LOT 201.18 **Objective Number:** 86061

Describe the instrumentation and controls available to monitor and operate the Rod Control System.
LOT 505.01 (92108) - Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: LOT 201.18 PPT slide 72, 0POP04-RS-0001 Step 2 RNO

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as this is the correct action and placing Rod Control in CB-A would stop automatic rod motion but not an allowed procedural action.

B: CORRECT. The rods should have stopped stepping out at a Tavg-Tref deviation of -1.0ºF so the RO should place the rods in manual for the given conditon in the stem. An Immediate Operator Action of 0POP04-RS-0001 is to trip the reactor and GO TO 0POP05-EO-EO00 if rods continue to move after they are placed in manual.

C: INCORRECT: Plausible because there is a 0.5ºF lockup between the start and reset of the automatic motion signal but for outward demand this would be -1.5ºF to start and -1.0ºF to stop rod motiion and placing Rod Control in CB-A would stop automatic rod motion but not an allowed procedural action.

D: INCORRECT. Plausible because there is a 0.5ºF lockup between the start and reset of the automatic motion signal but for outward demand this would be -1.5ºF to start and -1.0ºF to stop rod motiion. This is the correct action.

Question Level: H **Question Difficulty** 2

Justification:

The student must determine rod motion parameters for a given set of plant conditions.

SOURCE FOR RO #19

Exam Bank No.: 2913 Last used on an NRC exam: 2020

The Unit is at 80% power with the following conditions:

- Rod Control is in AUTO.
- Due to a plant transient, Tave is reading 591º.

Control rods will step in at ___(1)___ steps per minute and will stop stepping when the Tave-Tref deviation is ___(2)___.

A. (1) 6
(2) 0°F

B. (1) 6
(2) +1°F

C. (1) 39
(2) 0°F

D. (1) 39
(2) +1°F

Answer: D (1) 39 (2) +1F

Exam Bank No.: 2913 **Source:** New **Modified from:**

K/A Catalog Number: 001 K5.97 Control Rod Drive System: Knowledge of the following operational implications as they apply to the CRDS: Relationship of Tave to Tref

RO/SRO Importance: 3.3 / **Tier:** 2 **Group/Category:** 2

RO-10CFR55.41 # 7 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 201.18 **Objective Number:** 86061

Describe the instrumentation and controls available to monitor and operate the Rod Control System.

Reference: LOT 201.18, Powerpoint Presentation, slide 60

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible with confusion over rod speeds or calculating Tave wrong and with a misunderstanding of the dead band in the circuit.

B: INCORRECT: Plausible with confusion over rod speeds or calculating Tave wrong. This is the correct deadband.

C: INCORRECT: Plausible as this rod speed is correct and with a misunderstanding of the dead band in the circuit.

D: CORRECT: Tref is linear from 567 to 592F, 0 to 100% power. At 80% then, Tref is 587F and the given Tave is 4F hot. Automatic rod control is 6 spm at 3F deviation, 39 spm at 4F deviation, and 72 spm at 5F deviation. Rod motion stops at +1F deviation.

Level H **Difficulty** 3

Justification:

The student must determine rod motion based on given plant parameters.

NOTE

Steps 1.0 through 3.0 are IMMEDIATE ACTION Steps.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ **1.0**	**ENSURE “ROD BANK SEL” Switch In “MAN” {CP005}**	
_____ **2.0**	**VERIFY All Rods – NO ROD MOTION**	PERFORM the following: **a.** TRIP the Reactor. **b.** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
_____ **3.0**	**CHECK For Dropped Rods:**	
_____	**a. CHECK All Rods – ANY RODS DROPPED**	**a.** GO TO Step 4.0.
_____	**b. CHECK All Rods – ONLY ONE ROD DROPPED**	**b.** <u>IF</u> in Modes 1 OR 2, <u>THEN</u> PERFORM the following: **1)** TRIP the Reactor. **2)** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
_____	**c. GO TO Addendum 1, Recovery of a Dropped Rod**	

Rod Speed Programmer

OBJ 1

(key component of the Reactor Control Unit)

- +/- 1.5°F error rod motion starts at 6 spm
  - 1.5°F Deadband prevents excessive motion
  - 0.5°F Lockup (within the deadband outer band) prevents cycling.
- +/- 3°F to 5°F error rod speed ramps
  - starts at 6 spm (+/- 3 °F error)
  - ramps at 33 spm/°F up to 72 spm (linear)
  - reaches 72 spm (+/- 5 °F error).
- > +/- 5°F error rod speed is 72 spm (max).

Exam Bank No.: 3134 **Last used on an NRC exam:** Never

RO Sequence Number: 20

Unit 1 is in MODE 3.

- Operators are responding to a steam generator (SG) tube leak and are implementing 0POP04-RC-0004, Steam Generator Tube Leak.

In accordance with the procedure, the setpoint of the SG PORV on the leaking SG is set to between _____(1)_____ psig to _____(2)_____.

A. (1) 1225 and 1230
   (2) minimize the release of radioactivity

B. (1) 1225 and 1230
   (2) prevent an uncontrolled cooldown

C. (1) 1260 and 1265
   (2) minimize the release of radioactivity

D. (1) 1260 and 1265
   (2) prevent an uncontrolled cooldown

Answer: C (1) 1260 and 1265
(2) minimize the release of radioactivity

Exam Bank No.: 3134 **Source:** Modified **Modified from** 2396

K/A Catalog Number: 037 AK3.07 Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: Actions contained in EOP for S/G tube leak

RO Importance: 4.2 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 504.15 **Objective Number:** 92408

Given a copy of a step from 0POP05-EO-EO30, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose and the result.

Reference: 0POP04-RC-0004 step 17 and basis

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the number is one of the setpoints used at STP and would not result in an uncontrolled cooldown..

B: INCORRECT. Plausible because the number is one of the setpoints used at STP and would not result in an uncotroleed cooldown. which makes the basis plausible also. Depressurizing the S/G would cause leak rate to rise which is undesirable but not the basis for the setpoint.

C: CORRECT. Adjusting the setpoint to between 1260 and 1265 psig minimizes release of radioactivity from the affected SG to atmosphere by minimizing cycling of the SG PORV but also allows for not challenging the Main Steam Safety valve with the lowest setpoint (1285 psig).

D: INCORRECT. Plausible because the required setpoint will not cause a cooldown of the S/G which is undesirable.

Question Level: F **Question Difficulty** 3

Justification:

Must have knowledge of the ruptured S/G PORV setpoints and reason for the setpoint during a SGTR.

SOURCE FOR RO #20

Exam Bank No.: 2396 Last used on an NRC exam: 2015

The control room operators are responding to the symptom of a SGTR and Control Board indications show that the 1D SG is the affected SG.

Step 3 of 0POP05-EO-EO30, Steam Generator Tube Rupture, states: ADJUST ruptured Steam Generator (SG) PORV controller setpoint to BETWEEN 1260 PSIG AND 1265 PSIG.

1D SG PORV setpoint is now reading 1260 psig

The SG PORV setpoint is raised to between 1260 and 1265 psig on the affected SG to …

A. stabilize the ruptured SG pressure and level to prevent an uncontrolled cooldown of the RCS.

B. prevent steam generator overpressure due to overfilling of the ruptured steam generator.

C. prevent an unmonitored release by keeping the SG PORV from lifting.

D. minimize atmospheric releases.

Answer: D minimize atmospheric releases

Exam Bank No.: 2396 **Source:** New **Modified from:**

K/A Catalog Number: EPE 038 EK3.0 Knowledge of the reasons for the following responses as the apply to the SGTR:
EK3.02 Prevention of secondary PORV cycling

RO/SRO Importance: 4.4 / 4.5 **Tier:** 1 **Group/Category:** 1

RO-10CFR55.41 # 5 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 504.15 **Objective Number:** 92199

GIVEN a copy of a caution or note from 0POP05-EO-EO30, STATE/IDENTIFY the basis for the caution or note to include its purpose and the adverse impact of failure to comply with the caution or note.

Reference: 0POP05-EO-EO30, Steam Generator Tube Rupture; LP 504.15
0POP05-EO-EO30, Steam Generator Tube Rupture; LP 504.15.8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Lowering the set point would not prevent an uncontrolled RCS cooldown of the RCS, Plausible because if the there were no other SG's available feeding a intact SG with a tube rupture is a possible procedure flow path.

B: INCORRECT: There is no indication in the stem that the affected SG is overfilled, Plasuable because feeding a SG with a tube rupture that is faulted can lead to an uncontrolled cooldown.

C: INCORRECT: If the PORV did lift the release would be monitored

D: CORRECT:Isolation of the ruptured steam generator(s) effectively minimizes release of radioactivity from this generator. In addition, isolation is necessary to establish a pressure differential between the ruptured and non-ruptured steam generators in order to cool the RCS and stop primary to secondary leakage.

Level F **Difficulty** 3

Justification:

Must have knowledge of the reason SG PORV setpoints are raised during a SGTR.

_____ **16.0 PERFORM The Following:**

_____ **a. ADJUST boric acid flow controller to 8.0 [2800 to 3000 ppm]**

_____ **b. SET VCT makeup control to AUTO**

_____ **c. COMMENCE boration to minimum concentration required for cold shutdown conditions**

_____ **17.0 PERFORM The Following To Isolate Steam Flow From The Affected SG(s):**

_____ **a. ADJUST affected SG PORV controllers to BETWEEN 1260 AND 1265 PSIG {QDPS PRI/SEC}**

_____ **b. CHECK affected SG(s) PORV controllers in – AUTO**

b. PLACE Affected SG(s) PORV controller in AUTO.

_____ **c. CHECK affected SG(s) PORV – CLOSED**

c. <u>WHEN</u> affected SG(s) pressure LESS THAN 1260 PSIG, <u>THEN:</u>

1) PLACE affected SG PORV controllers in MANUAL and CLOSE affected SG(s) PORV.

2) PLACE Affected SG(s) PORV controller in AUTO.

Step 17.0 continued on next page

STEP DESCRIPTION FOR 0POP04-RC-0004 STEP 17.0

STEP: PERFORM The Following To Isolate Steam Flow From The Affected SG(s):

PURPOSE: To isolate flow from the affected SG(s) to minimize radiological releases. To maintain pressure in the affected SG(s) greater than the pressure in at least one intact SG following cooldown of the RCS in subsequent steps.

BASIS: Isolation of the affected steam generator(s) effectively minimizes the release of radioactivity. In addition, isolation is necessary to establish a pressure differential between the affected and intact steam generators in order to cool the RCS and stop primary-to-secondary leakage. In order to remove heat from the RCS, the affected steam generator pressure and RCS pressure must be maintained greater than the intact steam generator pressures. As this differential pressure is increased, so is the subcooling in the RCS. If sufficient differential pressure cannot be maintained, leakage from the RCS will continue since RCS pressure will remain greater than the affected steam generator pressure in order to remove decay heat.

ACTIONS:

- Adjust affected SG PORV(s) controller setpoint – 1260 to 1265 psig
- Check affected SG PORV(s) controller in AUTO and closed
- Place affected SG PORV(s) controller in MANUAL and close
- Locally isolate affected SG PORV(s)
- Close affected SG(s) MSIV(s) & MSIB(s)
- Locally close affected SG(s) MSIV(s) & MSIB(s)
- Close all remaining MSIV(s) & MSIB(s)
- Close main steam header isolation valves between MSIV(s) and main turbine throttle valves and use intact SG(s) PORV for steam dump
- Isolate affected SG(s) from at least one intact SG(s), if not, determine the SG with the smallest leakage to use for cooldown of the RCS.
- Close steam valves from affected SG to TDAFW pump
- Locally isolate TDAFW pump

STEP DESCRIPTION FOR 0POP04-RC-0004 STEP 17.0 (continued)

INSTRUMENTATION:

- SG pressure
- Motor driven AFW pump status indication
- SG PORV position indication
- MSIV(s) & MSIB(s) position indication
- Steam supply valve(s) position indication for TDAFW pump
- Position indication for Addendum 4 valves

CONTROL/EQUIPMENT:

- SG PORV controller
- MSIV(s) & MSIB(s)
- Steam supply valves to TDAFW pump
- Addendum 4 valves

KNOWLEDGE: The PORV on the affected SG(s) should remain available to limit SG pressure unless it fails open. This will minimize any challenges to the code safety valve(s). The main steam header isolation valves on Addendum 4 provide a backup means of isolating the affected SG(s) if the associated MSIV(s) and MSIB(s) cannot be closed. These valves may be closed in parallel with subsequent recovery steps after the affected SG(s) are isolated from the intact SG(s) by closing the appropriate intact SG(s) MSIV(s) and MSIB(s).

In the event that all SGs are leaking the only remaining option for the operator is to use the SG with the least leakage to perform the cooldown and notify Health Physics to estimate exposures to personnel on site and at the site boundry if required.

Exam Bank No.: 2167 **Last used on an NRC exam:** 2013

RO Sequence Number: 21

A Safety Injection occurred in Unit 2.

- Operators are performing the actions of Addendum 5, Verification of SI Equipment Operation of 0POP05-EO-EO00, Reactor Trip or Safety Injection.

At step 6, VERIFY Containment Isolation Phase ‘A’, the following indications are noted on the CONTAINMENT ISOLATION PHASE A/B status monitoring panel:

Train	PHASE A ISOL red light	BYP INOP white light	F/ACT white light
A	ON	OFF	ON
B	ON	ON	OFF
C	ON	OFF	OFF

Which of the following (1) describes the status of Phase ‘A’ Isolation and (2) any actions that may be required in accordance with 0POP05-EO-EO00, Reactor Trip or Safety Injection?

A. (1) At least one Train ‘A’ valve is open.
(2) Manually close the valve(s).

B. (1) At least one Train ‘B’ valve is open.
(2) Manually close the valve(s).

C. (1) At least one Train ‘A’ valve is open.
(2) Manually actuate Phase ‘A’ isolation.

D. (1) At least one Train ‘B’ valve is open.
(2) Manually actuate Phase ‘A’ isolation.

Answer: A (1) At least one Train ‘A’ valve is open. (2) manually close the valve(s).

Exam Bank No.: 2167 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 069 AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices

RO Importance: 3.5 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 504.05 **Objective Number:** 80483

Given a copy of a subsequent step or from memory an immediate action step from POP05-EO-EO00, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose and result.

Reference: LOT 201.21 PPT slides 56; 0POP05-EO-EO00, Addendum 5, Step 6; LOT 201.21 HO1 pg. 11

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT - For the given ESF Status Panel indications, a Phase ‘A’ CIV is not in the closed position in Train A. Per Addendum 5 of 0POP05-EO-EO00 the required operator action is to manually close the valve.

B: INCORRECT - Plausible because Train B also has an abnormal which could be an indication of a motor operated valves inability to reposition on Phase A (loss of power).

C: INCORRECT - Plausible beccause manually actuating Containment Isolation Phase A would be appropriate if the actuation had not occurred

D: INCORRECT - Plausible because Train B also has an abnormal indication lit and manually actuating Containment Isolation Phase A would be appropriate if the actuation had not occurred.

Question Level: H **Question Difficulty** 3

Justification:

The student must correctly analyze given plant indications and apply the correct EOP step action.

ADDENDUM 5
VERIFICATION OF SI EQUIPMENT OPERATION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 6	VERIFY containment isolation phase A:	
	___ a. Phase A - ACTUATED	a. Manually ACTUATE phase A.
	___ b. Phase A valves - CLOSED, REFER TO ADDENDUM 1, PHASE A ISOLATION VERIFICATION	b. Manually CLOSE valves.
___ 7	VERIFY ECW status: o ECW pumps - RUNNING o ECW pump discharge isolation valves - OPEN	WHEN the respective ESF Load Sequencer has completed its automatic sequence OR it is determined that the respective ESF Load Sequencer has failed, THEN PERFORM the following: a. Manually START pump(s). b. Manually OPEN discharge isolation valve(s). c. IF any ECW pump can NOT be started OR its discharge isolation valve can NOT be opened, THEN: 1) STOP associated STBY DG by PLACING in PULL-TO-STOP. 2) ENSURE associated Essential Chiller(s) TRIP.
___ 8	VERIFY CCW pumps - RUNNING	WHEN the respective ESF Load Sequencer has completed its automatic sequence OR it is determined that the respective ESF Load Sequencer has failed, THEN manually START pump(s).

condition clears (i.e., back to normal), the operator will hear a single chime at which time he can use the reset pushbutton to turn the light off.

4.3.3 ESF Actuated Status Indication

A. Monitors all three ESF trains through status of field contacts.

B. Upon ESF actuation, this system automatically monitors for proper actuation and positions of ESF equipment to the safe position

C. If components fail to actuate/position properly, a component lamp is lit in the respective status panel and a flashing annunciator for the affected systems will alarm visually and audibly (F/ACT).

D. Can be manually tested from the control room.

4.3.4 Actuation Signal Indicator Lights

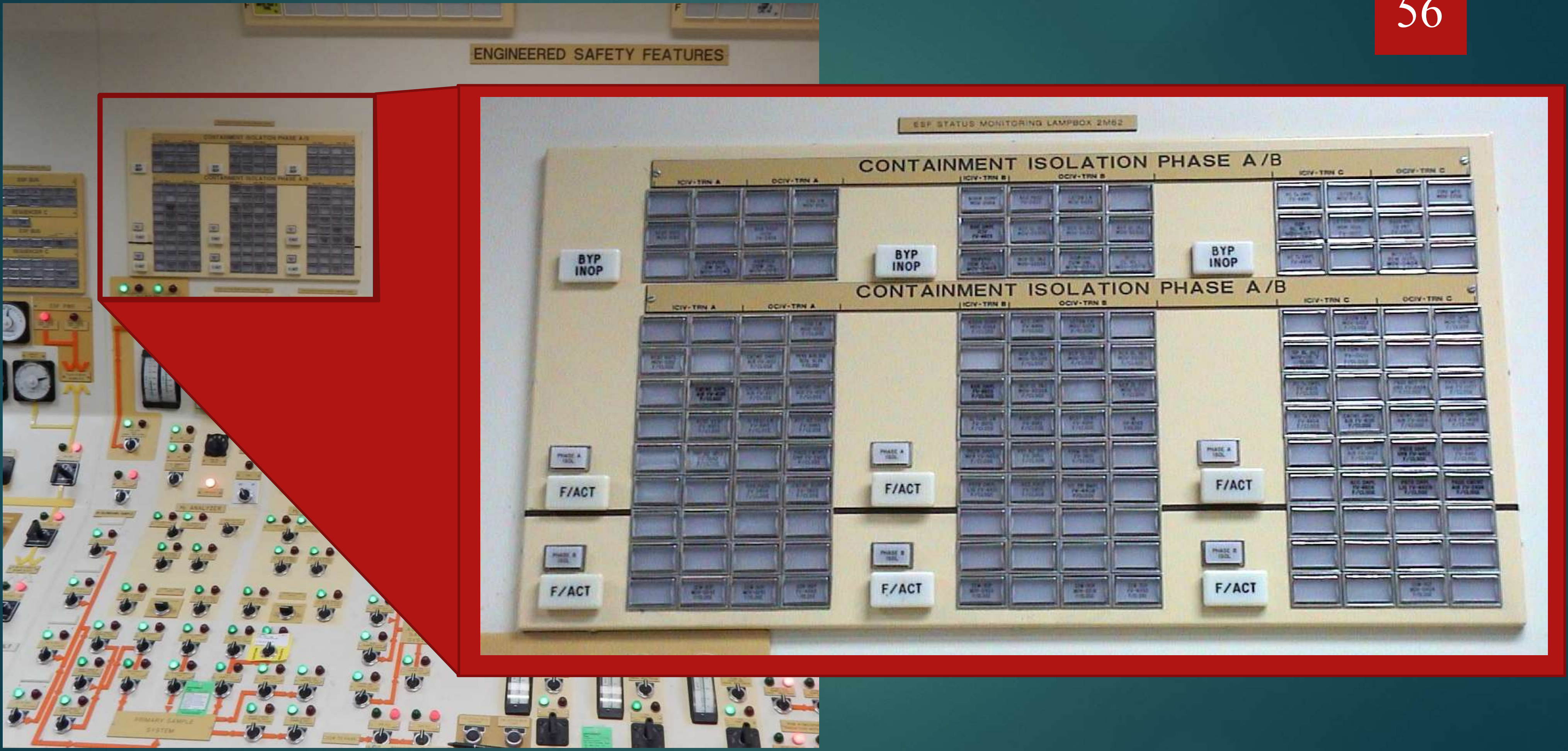
A. Each ESF Monitoring Window group is provided with a red lamp which lights upon the activation of their ESF signals

B. Red lights extinguish upon RESET/BLOCK of respective ESF signal

4.4 Power Supplies

Power supplies for the individual containment isolation valves are given in the associated system descriptions and are beyond the scope of this system description

CP002 Engineered Safety Features



Exam Bank No.: 1593 **Last used on an NRC exam:** Never

RO Sequence Number: 22

Unit 1 has entered 0POP05-EO-FRC2, Response to Degraded Core Cooling.

- All intact Steam Generators are depressurized to 285 psig and at least two (2) RCS hot leg temperatures are less than 435 degrees F.

The basis for stopping the SG depressurization at this point is to prevent _________________.

A. excessive RCS cooldown causing Pressurized Thermal Shock

B. losing the ability to maintain RCP support conditions

C. impeding Natural Circulation through overcooling

D. injection of Accumulator nitrogen into the RCS

Answer: D Injection of Accumulator nitrogen into the RCS.

Exam Bank No.: 1593 **Source:** Bank **Modified from** N/A

K/A Catalog Number: EPE W/E06 EK3.3 Knowledge of the reasons for the following responses as they apply to the (Degraded Core Cooling): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations

RO Importance: 4.0 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 504.31 **Objective Number:** 82960

Given a step, note or caution from 0POP05-EO-FRC2, STATE its basis

Reference: FRC2 background document page 28 & 29, LOT 504.31 PPT slides 20 & 21

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because PTS is a concern, but it is expected to have an excessive cooldown and a Red Path on Integriity may occur.

B: INCORRECT: Plausible because performance of the cooldown could cause the RCP's normal conditons to degrade but RCPs will not be tripped at this time..

C: INCORRECT: Plausible because excess cooldown can impede NC, however procedure requirement limits the cooldown rate (which is what could impede NC)

D: CORRECT: Per the background document the SG depressurization is stopped at 305 psig and 2 hot leg temperatures < 435 degrees F is to prevent injection nitrogen into the RCS from the accumulators.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a knowledge of procedural bases.

CAUTION

- The following step will cause accumulator injection which may cause an Integrity Red Path condition.
- This procedure SHALL be completed before transition to 0POP05-EO-FRP1, Response To Imminent Pressurized Thermal Shock Condition.

Basis – Once the RCS is cooled/depressurized to the point that the accumulators inject, the RCS cold leg temperatures could be reduced to the point that a red path on Integrity is created. Transition to FRP1 is deferred until after completion of FRC2 actions to preclude a deterioration of core cooling during the required soak time.

NOTE

IF any SG pressure lowers at a rate greater than 100 psi/min and low steam line pressure has been blocked, then a Main Steam Isolation will occur.

Basis – In the following step the operator will be dumping steam from the intact SGs, which will reduce their pressure below the low steam line pressure SI setpoint. The operator is also reminded that, after blocking low steamline press SI, limiting the depress to <100 psi/min will prevent MS ISOL.

Step 9 DEPRESSURIZE All intact SGs To 305 PSIG

BASIS: The controlled secondary depressurization has been shown to be an effective way to reduce RCS Pressure. RCS Pressure must be reduced in order for the SI accumulator and LHSI pumps to inject.

STEP DESCRIPTION TABLE FOR FR-C.2 Step 10

STEP: Depressurize All Intact SGs To (O.07) PSIG

PURPOSE: To recover the core via SI accumulator injection

BASIS:

The controlled secondary depressurization, similar to the one in ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, has been shown to be an effective way to reduce RCS pressure. RCS pressure must be reduced in order for the SI accumulator and low-head SI pumps to inject.

To prevent accumulator nitrogen injection, the operator should stop the secondary depressurization when the SG pressure reaches (O.07) psig and when at least two RCS hot leg temperatures fall below (F.05)°F. A steam generator pressure limit is set to preclude significant nitrogen injection into the RCS. To determine the steam generator pressure limit, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tank pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume. The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1V_1^{\gamma} = P_2V_2^{\gamma}$$

where $\gamma = 1.0$ for ideal gas expansion. The steam generator pressure limit is then determined by subtracting the RCS to SG delta P from P_2. The RCS to SG delta P should be calculated as described in the RCP TRIP/RESTART section in the Generic Issues of the Executive Volume. Instrument uncertainties are not included in the determination of the Steam generator pressure limit to ensure the amount of water injected into the RCS is not unnecessarily limited.

The hot leg temperature of (F.05) °F should be determined so that the RCS saturation pressure exceeds the accumulator pressure after the accumulator water has been discharged. This precludes nitrogen injection into the RCS. To determine the hot leg temperature, an ideal gas expansion calculation should be performed based on nominal plant specific values for initial accumulator tank pressure (P_1), initial nitrogen gas volume (V_1), and final nitrogen gas volume (V_2). The final nitrogen gas volume should be equivalent to the total accumulator tank volume.

STEP DESCRIPTION TABLE FOR FR-C.2 Step 10

The RCS pressure at empty tank conditions (P_2) is determined from:

$$P_1V_1^{\gamma} = P_2V_2^{\gamma}$$

where $\gamma = 1.0$ for ideal gas expansion. The setpoint temperature of (F.05)°F is the saturation temperature corresponding to P_2. Instrument uncertainties are not included in the determination of the RCS hot leg temperature limit to ensure the amount of water injected into the RCS is not unnecessarily limited.

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

○	(O.07)	Minimum SG pressure which prevents accumulator nitrogen injection. Refer to background document for FR-C.2.
○	(F.05)	RCS hot leg temperature to prevent accumulator nitrogen injection. Refer to background document for FR-C.2.

Exam Bank No.: 3141 **Last used on an NRC exam:** Never

RO Sequence Number: 23

Unit 1 is at 100% power.

- RT-8039, Failed Fuel Monitor indication is rising and in ALERT.
- The crew has entered 0POP04-RC-0001 High Reactor Coolant System Activity.
- Chemistry reports that Dose Equivalent Iodine DEQ I-131 is 0.4 µci/gm

For the given conditions, which of the following is a required action per 0POP04-RC-0001, High Reactor Coolant System Activity?

A. Perform an orderly plant shutdown.

B. Start an additional Charging Pump.

C. Raise letdown flow using additional orifices.

D. Isolate the VCT purge to the Gaseous Waste Processing System.

Answer: C Raise letdown flow using additional orifices.

Exam Bank No.: 3141 **Source:** Modified **Modified from** 958

K/A Catalog Number: 076 G2.4.11 High Reactor Coolant Activity: Knowledge of abnormal condition procedures.

RO Importance: 4.0 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-RC-0001, pgs. 4, 5, 22

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the procedure requires an orderly shutdown on DEQ- I-131 levels but only if it exceeds 0.5 uci/gm for 48 continuous hours.

B: INCORRECT. Plausible because this would allow raing letdown flow more than a single pump but the procedure requires limiting letdown flow to the capacity of one pump.

C: CORRECT. 0POP04-RC-0001 step 4 raises letdown flow by aligning additional letdown orifices.

D: INCORRECT. Plausible because 0POP04-RC-0001, Step 9 requires that VCT purge flow rate be lowered to 0.7 scfm initially but does not isolate the purge.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to evaluate the conditions given and determine the correct action to take for high RCS activity

Source For RO #23

Exam Bank No.: 958 **Last used on an NRC exam:** 2003

RO Sequence Number: 24

Plant conditions:

- Operating in Mode 1 at 100% power.
- CVCS Letdown Failed Fuel Monitor is in High Alarm.
- Chemistry has determined that the inservice demineralizer is working properly.

Which ONE of the following operator actions is required per 0POP04-RC-0001, High Reactor Coolant System Activity?

A. Reduce power

B. Isolate letdown

C. Increase letdown to 220 gpm and 250 gpm

D. Initiate hourly sampling of the RCS

Answer: C Increase letdown to 220 gpm

Exam Bank No.: 958 **Source:** Bank **Modified from**

K/A Catalog Number: 076.G2.4.10 Knowledge of annunciator response procedures.

RO Importance: 3.0 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)()

STP Lesson: LOT 504.02 **Objective Number:**

Reference: 0POP04-RC-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A:
B:
C:
D:

Question Level: **Question Difficulty**

Justification:

CAUTION

Placing a cation bed in service will reduce the lithium concentration in the RCS; thus, reducing the primary coolant pH. Significant reduction in RCS pH, for example from 7.2 to 6.9, may cause a crud burst.

NOTE

Raising Letdown flow SHALL be limited to only the flowrate allowed with one CCP in-service. (CR 11-13341)

_____ **4.0 PERFORM Any Of The Following Per 0POP02-CV-0004, Chemical And Volume Control System As Recommended By Chemistry:**

_____ **a. RAISE letdown flow by placing additional letdown orifice(s) in service**

_____ **b. PLACE cation demineralizers in service**

_____ **c. (Mode 5) PLACE Reactor Coolant Purification Pump (RCPP) In Service Observing All Notes And Cautions**

_____ **5.0 CHECK RT-8039, Failed Fuel Monitor –**

- GREATER THAN ALERT SETPOINT

OR

- RISING INDICATION

Response Not Obtained: GO TO Step 8.0.

NOTE

RCS activity sample requires at least 70 minutes to obtain results.

_____	6.0	**DETERMINE RT-8039, Failed Fuel Monitor Indication – VALID** • PERFORM a source check on RT-8039 • PERFORM a channel check on RT-8039 • DIRECT Health Physics to survey Letdown area piping • DIRECT Chemistry to sample RCS to verify activity rise	GO TO Step 8.0.

CAUTION

Placing a cation bed in service will reduce the lithium concentration in the RCS; thus, reducing the primary coolant pH. Significant reduction in RCS pH, for example from 7.2 to 6.9, may cause a crud burst.

_____	7.0	**CHECK Cation Demineralizers In Service**	IF Recommended by Chemistry, THEN PLACE cation demineralizers in service per 0POP02-CV-0004, Chemical and Volume Control System Control Subsystem
_____	8.0	**CONTACT Nuclear Fuel & Analysis Department To Implement Actions Of 0PGP04-ZE-0004, Fuel Integrity Program**	
_____	9.0	**ADJUST "1(2)-WG-FIC-4653 VOLUME CONTROL TANK FLOW" to LOWER purge flow rate to GWPS to 0.7 scfm {ZLP-116}**	

NOTE

- RCS Dose Equivalent Iodine sample analysis may take up to two hours to complete.
- RCS Dose Equivalent Iodine LESS THAN 2.80E-4 µCi/gm allows maximum VCT purge flow rate to GWPS of 2.2 scfm per Plant Curve Book Figure 10.80.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ **10.0**	***MONITOR*** **RCS Dose Equivalent Iodine <u>AND</u> ADJUST "1(2)-WG-FIC-4653 VOLUME CONTROL TANK FLOW" to control purge flow rate to GWPS in accordance with Plant Curve Book Figure 10.80. {ZLP-116}**	
_____ **11.0**	**INITIATE Actions To Change VCT GWPS Flowrate High Alarm Setpoint To No More Than 20% Above Actual Flowrate {1(2)-WG-FSH-4653}**	
_____ **12.0**	**CHECK RCS Dose Equivalent I-131 –** • LESS THAN 50% OF LIMITS SPECIFIED IN TECHNICAL SPECIFICATION 3.4.8 • STABLE OR LOWERING	NOTIFY the following for evaluation of continued operation with the current and projected RCS Dose Equivalent I-131 levels. • Station Management • Nuclear Fuel & Analysis Department
_____ **13.0**	**CHECK RCS Dose Equivalent I-131 LESS THAN 0.5 µCi/gm**	<u>IF</u> RCS Dose Equivalent I-131Operational Steady State limit of **0.5 µCi/gm** is exceeded for more than 48 hours during one continuous time interval, <u>THEN</u> PERFORM an orderly plant shut down to be in at least HOT STANDBY with Tavg LESS THAN 500°F within 6 hours.

Exam Bank No.: 3142 **Last used on an NRC exam:** Never

RO Sequence Number: 24

A small break LOCA has occurred in Unit 1.

- Containment pressure is 5.5 psig and slowly lowering.
- A transition has been made to 0POP05-EO-ES11, SI Termination.
- All SI Pumps have been secured.

Subcooling is noted to be 39ºF.

In accordance with 0POP05-EO-ES11, subcooling ___(1)____ satisfactory and the operators are directed to ___(2)___.

A. (1) is NOT
   (2) control charging flow

B. (1) is NOT
   (2) start SI pumps manually

C. (1) is
   (2) control charging flow

D. (1) is
   (2) start SI pumps manually

Answer: B (1) is NOT
(2) start SI pumps manually

Exam Bank No.: 3142 **Source:** New **Modified from** N/A

K/A Catalog Number: W/E02 EK1.1 Knowledge of the operational implications of the following concepts as they apply to the (SI Termination): Components, capacity, and function of emergency systems

RO Importance: 3.2 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 504.07 **Objective Number:** 30398

LIST the conditions which would require manually restarting SI pumps.

Reference: 0POP05-EO-ES11 CIP and step 9

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as the action is directed first in 0POP05-EO-ES11 if PZR level is low vice subcooling..

B: CORRECT. Per 0POP05-EO-ES11 CIP, SI re-initiation is required if RCS subcooling is less than 45ºF for the containment pressure given in the conditions and the operator is required to manually restart SI pumps.

C: INCORRECT. Plausible as 35 degrees F is the subcooling criteria to be met if containment pressure were < 5 psig and this action is directed first in 0POP05-EO-ES11 if PZR level is low vice subcooling.

D: INCORRECT. Plausible as 35 degrees F is the subcooling criteria to be met if containment pressure were < 5 psig.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to determine action and correct Subcooling criteria based on plant conditions.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___9	MONITOR SI Reinitiation Criteria:	
	___a. RCS subcooling based on core exit T/Cs - GREATER THAN 35°F [45°F]	a. PERFORM the following: 1) START SI pump(s). 2) GO TO 0POP05-EO-EO10, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.
	___b. Pressurizer level - GREATER THAN 8% [44%]	b. PERFORM the following: 1) CONTROL charging flow to restore pressurizer level GREATER THAN 8% [44%]. 2) IF pressurizer level can NOT be restored, THEN: a) START SI pump(s). b) GO TO 0POP05-EO-EO10, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.

CONDITIONAL INFORMATION PAGE

SI REINITIATION CRITERIA

IF EITHER condition listed below occurs, THEN manually START SI pump(s) and GO TO 0POP05-EO-EO10, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1.

- o RCS subcooling based on core exit T/Cs - LESS THAN 35°F [45°F]
- o Pressurizer level cannot be maintained GREATER THAN 8% [44%] using charging pumps

SECONDARY INTEGRITY CRITERIA

IF any UNISOLATED SG pressure is lowering in an uncontrolled manner AND NOT needed for RCS cooldown, THEN GO TO 0POP05-EO-EO20, FAULTED STEAM GENERATOR ISOLATION, Step 1.

EO30 TRANSITION CRITERIA

IF ANY SG level rises in an uncontrolled manner OR ANY SG has abnormal radiation, THEN GO TO 0POP05-EO-EO30, STEAM GENERATOR TUBE RUPTURE, Step 1.

COLD LEG RECIRCULATION SWITCHOVER CRITERIA

IF RWST level lowers to LESS THAN 75,000 GALLONS (14%), THEN GO TO 0POP05-EO-ES13, TRANSFER TO COLD LEG RECIRCULATION, Step 1.

AFWST MAKEUP CRITERIA

IF AFWST level lowers to LESS THAN 138,000 GALLONS (26%), THEN INITIATE makeup to the AFWST per 0POP02-AF-0001, AUXILIARY FEEDWATER, to prevent inventory problems during cooldown.

SEQUENCER LOADING VERIFICATION

IF a LOOP occurs, THEN PERFORM Addendum 9, Sequencer Loading Verification - Mode III.

LOSS OF EMERGENCY COOLANT RECIRCULATION

IF emergency coolant recirculation can not be established in at least one train OR is established and subsequently lost, THEN GO TO 0POP05-EO-EC11, LOSS OF EMERGENCY COOLANT RECIRCULATION, Step 1.

Exam Bank No.: 3149 **Last used on an NRC exam:** Never

RO Sequence Number: 25

- A Unit 1 reactor trip occurred due to MSIV 1C closing at 100% power.
- On the trip the 1B Main Feed Regulating Valve had to be manually closed
- The crew has transitioned to 0POP05-EO-ES01, Reactor Trip Response.
- Total AFW flow - 550 gpm
- Containment pressure is 0.5 psig
- SG pressures and narrow range levels are as follows:

SG	Pressure	NR Level
‘A’	1220 psig	19%
‘B’	1200 psig	81%
‘C’	1340 psig	15%
‘D’	1210 psig	16%

If Yellow Path Functional Recovery Procedures are implemented, which procedure should be implemented first? (Reference Provided)

A. 0POP05-EO-FRH2, Response To Steam Generator Overpressure

B. 0POP05-EO-FRH3, Response To Steam Generator High Level

C. 0POP05-EO-FRH4, Response To Loss of Normal Steam Release Capabilities

D. 0POP05-EO-FRH5, Response to Steam Generator Low Level

Answer: A 0POP05-EO-FRH2, Response To Steam Generator Overpressure

Exam Bank No.: 3149 **Source:** New **Modified from** N/A

K/A Catalog Number: W/E13 EA2.1 Ability to determine and interpret the following as they apply to the (Steam Generator Overpressure): Facility conditions and selection of appropriate procedures during abnormal and emergency operations

RO Importance: 2.9 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 504.34 **Objective Number:** 92223

STATE/IDENTIFY the conditions under which 0POP05EO-FRH2 is entered.

Reference: 0POP05- EO-FO03

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☑ **Attachment:** 0POP05- EO-FO03 Page 2 of 2

Distractor Justification

A: CORRECT. For the given conditions, SG ‘D’ pressure exceeds the Yellow Path criteria (1325 psig) of the Heat Sink CSF Status Tree for entry into FRH2.

B: INCORRECT. Plausible because SG B meets adverse containment criteria given for entry into FRH3 however the conditions given in the stem do not support that.

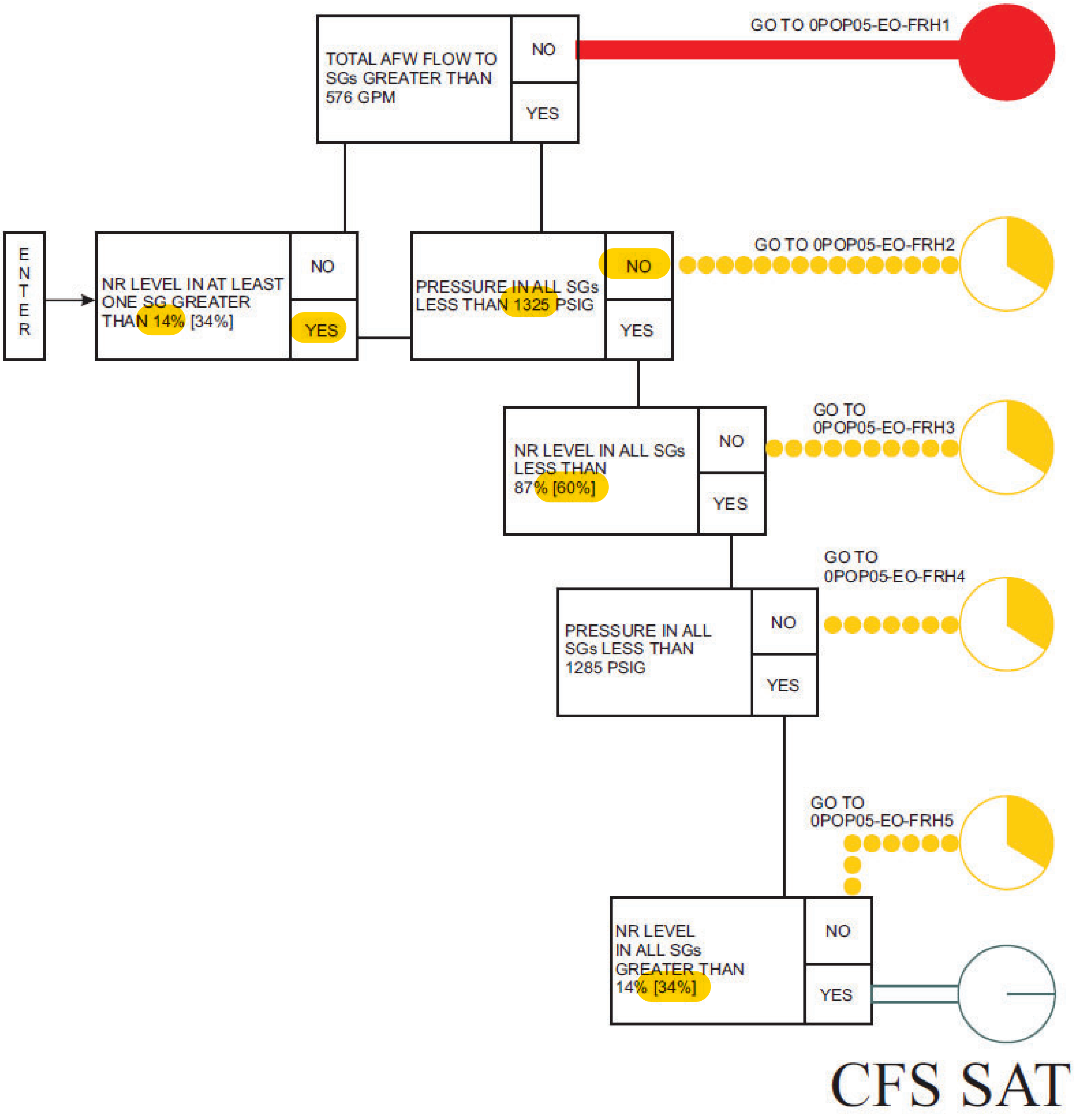
C: INCORRECT. Plausible because entry conditions are met for entry into FRH4 but a more severe conditions exists challenging the design pressure of the SG and steam line and the Critical Safety Function Status Tree would send you to FRH2 first.

D: INCORRECT. Plausible because 3 of 4 SGs meet the adverse containment criteria for entry into FRH5 however the conditions given in the stem do not support that.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the conditions and determine the correct procedure to implement.



CFL0723(10/29/07)

Exam Bank No.: 3234 **Last used on an NRC exam:** Never

RO Sequence Number: 26

Following a Large Break LOCA, the crew enters 0POP05-EO-FRZ2, Response to Containment Flooding, due to an unexpected rise in containment water level following transfer to cold leg recirculation.

According to 0POP05-EO-FRZ2, the cause of the flooding could be a break in the ______(1)______ system.

This system would normally isolate on a Containment Isolation _____(2)_____ signal.

A. (1) AFW
(2) Phase A

B. (1) AFW
(2) Phase B

C. (1) CCW
(2) Phase A

D. (1) CCW
(2) Phase B

Answer: D (1) CCW (2) Phase B

Exam Bank No.: 3234 **Source:** New **Modified from**

K/A Catalog Number: W/E15 EA1.1 Ability to operate and/or monitor the following as they apply to the (Containment Flooding): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

RO Importance: 2.9 **Tier:** 1 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 504.41 **Objective Number:** 83786

State/Identifythe indications that are available for identifying unexpected sources of water to the emergency sump.

Reference: 0POP05-EO-FRZ2, LOT201.12 Slide 11

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as there are AFW lines in containment, an the AFW system has an OCIV, but the valve does not close on a Copntainment Isolation signal.

B: INCORRECT: Plausible as there are AFW lines in containment, an the AFW system has an OCIV, but the valve does not close on a Copntainment Isolation signal.

C: INCORRECT: Plausible CCW is one of 4 systems listed in 0POP05-EO-FRZ2 as having a potential to flood containment and if the student confuses Phase A with Phase B.

D: CORRECT: CCW is one of 4 systems listed in 0POP05-EO-FRZ2 as having a potential to flood containment. The CCW valves isolate on a Containment Isolation Phase B signal.

Question Level: F **Question Difficulty** 3

Justification:

The student recall the contents of 0POP05-EO-FRZ2, and the automatic feature on the CCW lines entering containment.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout CIP page should be open.

___1 TRY TO Identify Unexpected Source Of Water To Emergency Sump:

- o CCW System
- o RCB Chilled Water System
- o Reactor M/U Water System
- o Fire Protection System

___2 CHECK Containment Emergency Sump Activity Level:

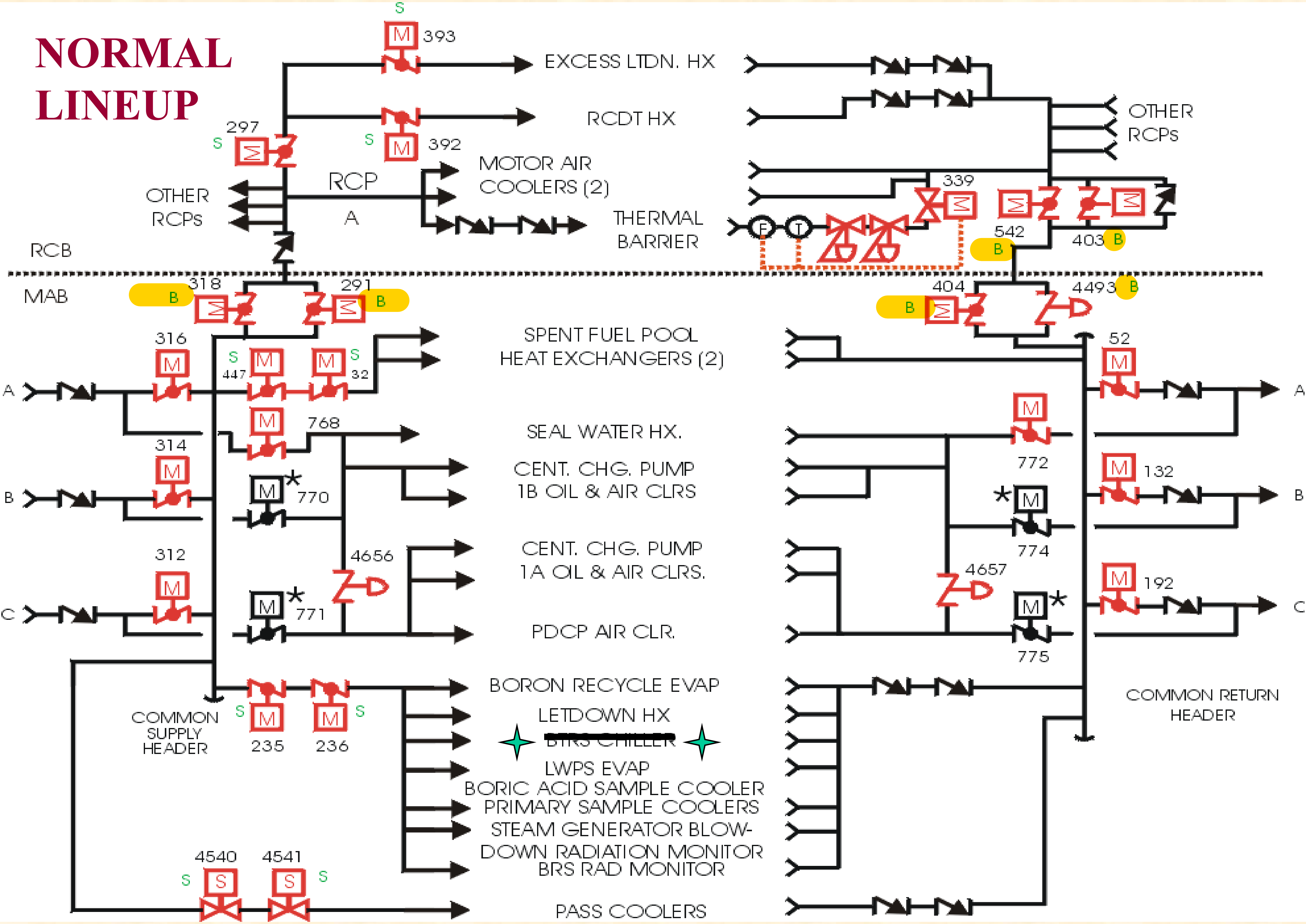
___a. NOTIFY Chemistry to sample containment emergency sump for activity

___3 NOTIFY TSC Staff Of Emergency Sump Water Level And Emergency Sump Activity Level To Obtain Recommended Action

___4 RETURN TO Procedure And Step In Effect

-END-

NORMAL LINEUP



Exam Bank No.: 3144 **Last used on an NRC exam:** Never

RO Sequence Number: 27

Unit 1 is at 100% power.

- The following is noted for RCP 1A motor bearing temperatures.
    - Motor Upper Radial Bearing - 180ºF
    - Motor Lower Radial Bearing - 205ºF

Per 0POP02-RC-0004, Operation of Reactor Coolant Pump _____(1)_____ temperature(s) would require going to 0POP04-RC-0002, Reactor Coolant Pump Off Normal.

These parameters can be monitored using ______(2)______.

A. (1) both the Upper and Lower Radial Bearings
(2) the ICS only

B. (1) both the Upper and Lower Radial Bearings
(2) a CP-005 control board indicator and the ICS

C. (1) only the Lower Radial Bearing
(2) the ICS only

D. (1) only the Lower Radial Bearing
(2) a CP-005 control board indicator and the ICS

Answer: C (1) Only the Lower Radial Bearing
(2) the ICS only

Exam Bank No.: 3144 **Source:** New **Modified from** N/A

K/A Catalog Number: 003 A4.02 Ability to manually operate and/or monitor in the control room: RCP motor parameters

RO Importance: 2.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.15 **Objective Number:** 4829

4829 - DESCRIBE instrumentation available for the RCPs in the control room and locally.
50672 - In regards to 0POP02-RC-0004, DESCRIBE the following: Precautions and Limitations.

Reference: 0POP02-RC-0004 Precaution 4.18, 0POP04-RC-0002, pgs 3 & 104

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as both indications would be abnormal but only the Motor Lower Radial Bearing Temperature exceeds the criteria to go to 0POP04-RC-0002 per 0POP02-RC-0004.

B: INCORRECT. Plausible as both indications would be abnormal but only the Motor Lower Radial Bearing Temperature exceeds the criteria to go to 0POP04-RC-0002 per 0POP02-RC-0004 and other RCP parameters such as loop flow and RCP amps have board indications on CP-005 but the motor radial bearing temperatures are only available on ICS in the control room.

C: CORRECT. In accordance with 0POP02-RC-0004, Operation of Reactor Coolant Pump precaution 4.18 only the lower motor radial bearing exceeds the criteria to go to 0POP04-RC-0002. The Motor radial bearing temperatures can only be monitored using an ICS computer point or trend.

D: INCORRECT. Plausible because other RCP parameters such as loop flow and RCP amps have board indications on CP-005 but the motor radial bearing temperatures are only available on ICS in the control room.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions and compare to the limits for RCP trip in accordance with the Off Normal procedure.

4.18 IF any of the conditions listed below occur, THEN GO TO 0POP04-RC-0002, Reactor Coolant Pump Off Normal:

- Motor Upper or Lower Radial Bearing Temperature - GREATER THAN OR EQUAL TO 195°F
- Motor Upper or Lower Thrust Bearing Temperature - GREATER THAN OR EQUAL TO 195°F
- Motor Stator Winding Temperature - GREATER THAN OR EQUAL TO 310°F
- Seal 1 Water Inlet Temperature - GREATER THAN OR EQUAL TO 230°F
- Lower Seal Water Bearing Temperature - GREATER THAN OR EQUAL TO 230°F
- SHAFT Vibration (Brg2-Vert OR Brg2-Horiz) -
  a. GREATER THAN OR EQUAL TO 20 MILS
  OR
  b. GREATER THAN OR EQUAL TO 15 MILS AND THE RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 1.0 MIL PER HOUR
- CASE Vibration (Mtr_Accel-Vert OR Mtr_Accel-Horiz) -
  a. GREATER THAN OR EQUAL TO 5 MILS
  OR
  b. GREATER THAN OR EQUAL TO 3 MILS AND THE RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 0.2 MIL PER HOUR
- Number 1 Seal DP - LESS THAN 220 PSID

4.19 WHEN a RCP is to be stopped, THEN the desired RCP handswitch SHALL be turned to STOP AND maintain it in the STOP position until the RCP motor breaker opens. The Pull-to-Lock (PTL) position does **NOT** have a STOP contact and rapidly placing the handswitch to PTL may **NOT** allow time for the relays to position. (CR 05-1339)

4.20 IF an RCP experiences a simultaneous loss of seal water injection flow AND loss of CCW flow to the thermal barrier, THEN GO TO 0POP04-RC-0002, Reactor Coolant Pump Off Normal.

4.21 Four RCPs operations are **NOT** permitted WHEN RCS average temperature is LESS THAN 140°F. Do **NOT** run all 4 RCPs UNTIL RCS average temperature reaches GREATER THAN 140°F. (3 pumps or less restriction RCS < 140°F, This limitation is required to demonstrate acceptable fuel assembly top nozzle hold down spring forces in the Cold Zero Power lift force calculation.) (CR 03-1149)

4.22 Experience has shown that starting RCP(s) MAY cause spikes in Nuclear Instruments indications due to induced electrical noise in the instrument cables.

4. RCP Pump Bearings, Motor Bearings or Stator Temperature High:

RCP Plant Computer Points
(Plant Computer Display RC-010 or RC-011
Reactor Coolant Pumps)

	1A(2A)	1B(2B)	1C(2C)	1D(2D)
• RCP Motor Stator Winding Temp	T0412	T0432	T0452	T0472
• RCP Motor Upper Radial Bearing Temp	T0413	T0433	T0453	T0473
• RCP Motor Upper Thrust Bearing Temp	T0414	T0434	T0454	T0474
• RCP Motor Lower Radial Bearing Temp	T0415	T0435	T0455	T0475
• RCP Motor Lower Thrust Bearing Temp	T0416	T0436	T0456	T0476
• RCP Lower Seal Water Bearing Temp	T0417	T0437	T0457	T0477
• RCP Seal 1 Water Inlet Temp	T0181	T0182	T0183	T0184

5. RCP High Vibration via "VIBR MNTR SYS TRBL": Lampbox 9M01, Window A-8

• RCP Case Vibration	Vibration Monitoring Panel CP014
• Mtr_Accel-Vert	
• Mtr_Accel-Horiz	
• RCP Shaft Vibration	Vibration Monitoring Panel CP014
• Brg2-Vert	
• Brg2-Horiz	

6. Abnormal RCP Number 1 Seal Indication:

• "RCP 1A(2A) NO 1 SEAL DP LO"	Lampbox 4M07, Window A-2
• "RCP 1B(2B) NO 1 SEAL DP LO"	Lampbox 4M07, Window A-4
• "RCP 1C(2C) NO 1 SEAL DP LO"	Lampbox 4M07, Window A-6
• "RCP 1D(2D) NO 1 SEAL DP LO"	Lampbox 4M07, Window A-8
• "RCP 1A(2A) NO 1 SEAL LKF FLOW HI/LO"	Lampbox 4M07, Window B-1
• "RCP 1B(2B) NO 1 SEAL LKF FLOW HI/LO"	Lampbox 4M07, Window B-3
• "RCP 1C(2C) NO 1 SEAL LKF FLOW HI/LO"	Lampbox 4M07, Window B-5
• "RCP 1D(2D) NO 1 SEAL LKF FLOW HI/LO"	Lampbox 4M07, Window B-7

Conditional Information Page

RCP TRIP CRITERIA

IF ANY VALID condition listed below occurs, THEN PERFORM the following:

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP affected RCP(s).
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection
4. *CONTINUE* at Step 1.0 of procedure as resources permit

RCP TRIP conditions:

Motor Upper or Lower Radial Bearing Temp - GREATER THAN OR EQUAL TO 195°F	Lower Seal Water Bearing Temp - GREATER THAN OR EQUAL TO 230°F
Case Vibration (Mtr_Accel-Vert OR Mtr_Accel-Horiz) – a. GREATER THAN OR EQUAL TO 5 MILS b. GREATER THAN OR EQUAL TO 3 MILS AND RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 0.2 MIL PER HOUR	Shaft Vibration (Brg2-Vert OR Brg2-Horiz) – a. GREATER THAN OR EQUAL TO 20 MILS b. GREATER THAN OR EQUAL TO 15 MILS AND RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 1.0 MIL PER HOUR
Seal 1 Water Inlet Temp - GREATER THAN OR EQUAL TO 230°F	Motor Stator Winding Temp - GREATER THAN OR EQUAL TO 310°F
Number 1 Seal DP - LESS THAN 220 PSID	

LOSS OF ALL RCP SEAL COOLING

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP affected RCP(s) within 1 minute.
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection **WHILE CONTINUING WITH** the following actions below:
   IF all RCP seal 1 inlet temperatures are LESS THAN 230°F, THEN PERFORM the following:
   a. ENSURE "RECIRC HCV-0285" PDP recirculation valve is 100% OPEN
   b. START the PDP.
   c. Slowly CLOSE "RECIRC HCV-0285", to establish seal injection flow to lower seal water inlet and lower seal water bearing temperatures at LESS THAN 1°F PER MINUTE.

RCP TRIP CRITERIA FOR HIGH NUMBER 1 SEAL LEAKOFF FLOW

IF RCP Number 1 Seal leakoff flow increases to GREATER THAN 6 gpm **OR** pegged high**,** THEN PERFORM the following:

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP the affected RCP.
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection.
4. *CONTINUE* actions of this procedure as resources permit.
5. CLOSE affected RCP Number 1 Seal leakoff isolation valve between 3 to 5 minutes after stopping RCP.

• RCP 1A(2A) "SEAL NO 1 LKF ISOL FV-3154"	• RCP 1B(2B) "SEAL NO 1 LKF ISOL FV-3155"
• RCP 1C(2C) "SEAL NO 1 LKF ISOL FV-3156"	• RCP 1D(2D) "SEAL NO 1 LKF ISOL FV-3157"

6. *MONITOR* CCW flow - ADEQUATE.

RCP MOTOR THRUST BEARING TEMPERATURE HIGH

IF Motor Upper or Lower Thrust Bearing Temp - GREATER THAN OR EQUAL TO 195°F, THEN, PERFORM Step 2 of the procedure body (page 7).

This Procedure is Applicable in ALL Modes

Exam Bank No.: 3145 **Last used on an NRC exam:** Never

RO Sequence Number: 28

Unit 1 is performing a Xe free reactor startup.

- Source Range NI Count Rate is stable at 5 x 10^3 cps.
- A malfunction of Letdown Heat Exchanger CCW TCV (TV-4494) reduces cooling water flow to the letdown heat exchanger.
- LET DN HX OUTLET TEMP rises to 135ºF.
- Rod withdrawal has been suspended while the malfunction is investigated.

Over time, Source Range NI Count Rate will ___(1)___ as a result of the boron concentration at the outlet of the CVCS demineralizers being ___(2)___.

A. (1) lower
(2) lower

B. (1) lower
(2) higher

C. (1) rise
(2) lower

D. (1) rise
(2) higher

Answer: B (1) lower
(2) higher

Exam Bank No.: 3145 **Source:** New **Modified from** N/A

K/A Catalog Number: 004 K5.37 Knowledge of the operational implications of the following concepts as they apply to the CVCS: Effects of boron saturation on ion exchanger behavior

RO Importance: 2.6 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.06 **Objective Number:** 93124

93124 - Given a description of plant conditions, ANALYZE the conditions and PREDICT how the Chemical and Volume Control System will respond.
LOT 105.03 N100008 - DESCRIBE the Demineralizer characteristics that can cause a change in boron concentration.

Reference: LOT105.08 Student Handout pg. 15

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student misinterprets how boron concentration will change due to a temperature change in letdown flow.

B: CORRECT. Reduced cooling water flow to the letdown heat exchanger will cause letdown temperature to rise and release more boron from the letdown demineralizers. The higher boron concentration will cause reactor power to lower.

C: INCORRECT. Plausible if the student misinterprets how boron concentration will change due to a temperature change in letdown flow and the resulting effect on reactor power.

D: INCORRECT. Plausible if the student misinterprets the resulting effect on rx power

Question Level: H **Question Difficulty** 3

Justification:

The student must assess plant conditions and determine the effect on the CVCS demineralizer boron concentration and reactor power.

TEMPERATURE

Ion exchange resins are very sensitive to elevated temperatures. Although the inert resin bead structure is stable up to fairly high temperatures (300°F), the active exchange sites are not. The anion resin begins to decompose slowly at about 140°F and the decomposition becomes rapid above 180°F. The cation resin is stable up to 250°F. Therefore, with purification systems that handle high temperature water, the influent to the demineralizers must be cooled to acceptable temperatures before it reaches the ion exchange resins.

The anion resin (OH^- form) decomposes by either of two mechanisms with approximately equal probability as shown in Equation 4-8 and Equation 4-9.

$$RCH_2N(CH_3)_3 + OH^- \rightarrow \overset{\text{Alcohol Resin}}{RCH_2OH} + \underset{\substack{\text{Trimethylamine} \\ \text{(TMA)}}}{N(CH_3)_3}$$

Equation 4-8

$$RCH_2N(CH_3)_3 + OH^- \rightarrow \underset{\text{Amine Resin}}{RCH_2N(CH_3)_2} + \underset{\text{Methyl Alcohol}}{CH_3OH}$$

Equation 4-9

The alcohol form of the resin has no exchange ability. Trimethylamine (TMA) is a weak base. The amine resin formed in the second reaction is an inferior ion exchange resin with exchange capabilities much less than the original form of the resin. Thus, both reactions lead to partial or complete loss of exchange ability and if the temperature is sufficiently high, or if a lower temperature (180°F) is sustained for a long period of time, the resin will be unfit for use.

Cation exchange resins begin to undergo thermal decomposition at temperatures above 250°F by the reaction:

$$RSO_3H + H_2O \rightarrow RH + H_2SO_4$$

Equation 4-10

This reaction destroys all exchange capacity of the cation bed and produces sulfuric acid.

In systems where it is possible to subject the demineralizer resin to high temperatures, the demineralizers have automatic features to protect against temperature damage. This is usually accomplished by automatic closure of the demineralizer inlet valves to isolate the demineralizer from the high temperature liquid, when a high temperature at the inlet to the demineralizer is sensed. These systems are typically equipped with a bypass valve that can divert flow around the demineralizer until normal system temperature is restored.

The boron affinity of a resin bed is affected by the temperature of the coolant passed through the bed. At lower temperatures, the borate ion bonding to the exchange site contains three boron atoms. At higher temperatures, the borate ion contains only one boron atom. The result of this characteristic is that at lower temperatures the resins are more efficient at removing boron from the coolant than at higher temperatures. A saturated resin bed will actually release boron as temperature is increased.

The temperature of a deborating demineralizer increases 75°F. What effect will this have on demineralizer operation?

Example 4-5

Exam Bank No.: 3146 **Last used on an NRC exam:** Never

RO Sequence Number: 29

Unit 1 is in MODE 5.

- RHR Pumps 1A and 1C are in service.
- Annunciator Lampbox 1M02 F-8, RHR Pump 1C DISCH FLOW LO alarms and locks in.
- RHR Pump 1C is running and flow is steady at 900 gpm

This alarm actuates at ___(1)____ gpm.

The action required for this alarm is to _________(2)________.

A. (1) 1200
   (2) ensure that RHR Pump 1C trips

B. (1) 1200
   (2) attempt to restore RHR Pump 1C flow

C. (1) 925
   (2) ensure that RHR Pump 1C trips

D. (1) 925
   (2) attempt to restore RHR Pump 1C flow

Answer: C (1)925
(2)ensure that RHR Pump 1C trips

Exam Bank No.: 3146 **Source:** New **Modified from** N/A

K/A Catalog Number: 005 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure modes for pressure, flow, pump motor amps, motor temperature, and tank level instrumentation

RO Importance: 2.7 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 201.09 **Objective Number:** 4245

GIVEN a plant or system condition, PREDICT the operation of the Residual Heat Removal System.

Reference: POP09-AN-01M2 Pgs. 13, 73, and 74

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because 1200 gpm is the setpoint for the Pretrip alarm. The action is correct.

B: INCORRECT. Plausible because 1200 gpm is the setpoint for the Pretrip alarm and the action is the action that would be taken if the Pretrip alarm is in alone.

C: CORRECT. The discharge flow lo alarm setpoint is 925 gpm per P0POP09-AN-01M2 Window F-8. The pump should trip if this alarm comes in, so the action per the POP09 is to ensure the pump is tripped.

D: INCORRECT. Plausible because this is the action that would be taken for the Pretrip alarm. The setpoint is correct.

Question Level: F **Question Difficulty** 3

Justification:

The student must know the setpoint and significance of the alarm and determine the correct procedural action.

RHR PUMP 1C(2C) LOW FLOW PRE-TRIP

Automatic Actions: 1) RHR Pump 1C(2C) trips at 925 gpm (after 5 sec time delay).

Immediate Actions: None

Subsequent Actions:

______ 1) MONITOR RHR Flow at FI-0869, RHR PUMP 1C(2C) FLOW.

______ 2) ATTEMPT to restore flow to 3,000 gpm or as directed by the Shift Manager/Unit Supervisor.

______ 3) IF flow continues to lower, THEN RHR pump will trip at 925 gpm.

______ 4) IF the RHR Pump does NOT trip at 925 gpm lowering, THEN TRIP RHR Pump 1C(2C).

______ 5) REFER TO Window F-8, RHR PUMP 1C(2C) DISCH FLOW LO.

______ 6) IF all RHR flow is lost, THEN GO TO 0POP04-RH-0001, Loss of Residual Heat Removal

Probable Causes:

1) Low RHR pump 1C flow (less than 1200 gpm for greater than 5 seconds)
2) Low RHR pump 2C flow (less than 1200 gpm for greater than 10 seconds)
3) RHR Suction valve closed or throttled
4) Low RCS hot leg level
5) RHR vortexing during mid-loop
6) Instrument failure

RHR PUMP 1C(2C) DISCH FLOW LO

Automatic Actions: RHR Pump 1C(2C) trips

Immediate Actions: None

Subsequent Actions: ______ 1) ENSURE RHR Pump 1C(2C) trips.

______ 2) IF all RHR flow is lost, THEN GO TO 0POP04-RH-0001, Loss of Residual Heat Removal.

______ 3) IF alarm occurs during mid-loop operations, THEN perform the following:

______ a) ENSURE RCS narrow range hot leg level stable and greater than +8 inches.

______ b) CLOSE Cold Leg Injection Valve “LOOP C Tc INJ MOV-0031C”

______ c) IF RCS NR hot leg level is less than +6 inches, THEN GO TO 0POP04-RH-0001, Loss of Residual Heat Removal.

______ 4) IF any RHR train remains in service, THEN PERFORM the following:

______ a) CONTROL RHR HX flow to maintain RCS temperature stable or controlled.

______ b) IF additional RCS cooling is required, THEN START the standby RHR train per 0POP02-RH-0001, Residual Heat Removal System Operation.

______ 5) PLACE RHR Pump 1C(2C) Control Switch in PULL TO LOCK.

______ 6) IF RHR Pump 1C(2C) was being used for Pumping the Refueling Cavity to the RWST, THEN SECURE Refueling Cavity draining by closing the RWST isolation valve being used to throttle flow.

______ 7) INVESTIGATE cause of RHR Pump 1C(2C) trip.

Page 1 of 2

1M02-F-8

RHR PUMP 1C(2C)
DISCH FLOW
LO

RHR PUMP 1C(2C) DISCH FLOW LO

Subsequent Actions: (Continued)

______ 8) IF the RHR pump was running following post maintenance activities, THEN REFER TO 0POP04-GI-0001, Gas Intrusion.

______ 9) TAKE appropriate action per Technical Specifications 3.5.6, 3.9.8.1, 3.9.8.2, 3.4.1.4.1, 3.4.1.3 and 3.4.1.4.2.

______ 10) WHEN the cause of the Residual Heat Removal Pump trip is identified AND corrected, THEN ENSURE 1C(2C) Residual Heat Removal Pump aligned in accordance with 0POP02-RH-0001, Residual Heat Removal System Operation, as directed by the Shift Manager/Unit Supervisor.

Probable Causes:

1) Flow controller malfunction.
2) RHR suction valve(s) closed.
3) Instrument failure.
4) Low RCS hot leg level.
5) RHR vortexing during mid-loop.

Origin and Setpoint:

1) 1(2)-RH-FT-0869 Less than or equal to 925 gpm

References:

1) 0POP04-RH-0001, Loss of Residual Heat Removal
2) Logic Diagram 5R169Z42180
3) Technical Specifications 3.5.6, 3.9.8.1, 3.9.8.2, 3.4.1.4.1, 3.4.1.3 and 3.4.1.4.2.
4) DBD RHR System 5R169MB1021
5) 0POP02-RH-0001, Residual Heat Removal System Operation
6) 0POP04-GI-0001, Gas Intrusion

Exam Bank No.: 3147 **Last used on an NRC exam:** Never

RO Sequence Number: 30

- RCS pressure is 950 psig.
- 0POP03-ZG-0001, Plant Heatup is in progress.

In accordance with Technical Specifications, a Safety Injection (SI) accumulator boron concentration is required to be between ___(1)___ ppm to be considered operable.

When placing the SI accumulators in service, after the discharge valve is opened the Power Lockout Switch on CP-001 is placed in the ___(2)____ position.

A. (1) 2300 and 2700
   (2) POWER ON

B. (1) 2300 and 2700
   (2) POWER OFF

C. (1) 2700 and 3000
   (2) POWER ON

D. (1) 2700 and 3000
   (2) POWER OFF

Answer: D (1) 2700 and 3000 (2) POWER OFF

Exam Bank No.: 3147 **Source:** New **Modified from** N/A

K/A Catalog Number: 006 A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Boron concentration in accumulator, boron storage tanks

RO Importance: 3.0 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(8)

STP Lesson: LOT 503.01 **Objective Number:** 80056

80056 - DETERMINE the applicable Technical Specification and/or Technical Requirements Manual (TRM) Limiting Conditions for Operations and the required action(s) to be taken.
LOT 506.01 92161 - DESCRIBE the general sequence of operation of components in the referenced procedure.

Reference: Technical Specifications 3.5.1; 0POP03-ZG-0001, Step 7.26, 0POP01-ZQ-0022, Form 10 pg. 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the Accumulator isolation valve power lockout switch must be turned on to operate the valve and 2300-2700 satisfies the minimum shutdown margin curve figure 5.5 of the Unit 1 Plant Curve Book.

B: INCORRECT. Plausible because 2300-2700 satisfies the minimum shutdown margin curve, figure 5.5 of the Unit 1 Plant Curve Book.

C: INCORRECT. Plausible because the Accumulator isolation valve power lockout switch is first placed in the POWER ON position prior to opening the valve. The first part is correct.

D: CORRECT. Technical Specifications requires SI accumulator boron concentration to be between 2700 and 3000 ppm. In accordance with 0POP03-ZG-0001, Plant Heatup, the SI accumulators are placed in service by energizing the Accumulator isolation valve, opening the valve, then placing the Accumulator isolation valve power lockout switch in the POWER OFF position.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall Technical Specification limits and correct procedural actions.

Initials

CAUTION

50 to 100°F RCS subcooling SHALL be maintained during heatup. This will help ensure that Annunciator Lampbox 5M02 Windows B-6 “RCS COLD OVERPRESS ALERT – TRN A” and B-7 “RCS COLD OVERPRESS ALERT – TRN B” will not come in even though COMS has been blocked.

7.25 INITIATE RCS heatup to greater than 540°F while continuing with Steps 7.26 through 7.28. ______

7.26 WHEN RCS pressure is between 900 and 1000 psig, THEN PLACE all safety injection accumulators in service as follows (References 2.1.6 and 2.1.7):

7.26.1 VERIFY accumulator chemistry within the specifications of 0PCP01-ZA-0038, Plant Chemistry Specifications. ______

7.26.2 ENSURE accumulator levels between 8822.8 and 9076.0 gallons AND RECORD indicated level from the Plant Computer.

- ACC 1A(2A) SILA0950 ________ / SILA0951 ________ ______
- ACC 1B(2B) SILA0952 ________ / SILA0953 ________ ______
- ACC 1C(2C) SILA0954 ________ / SILA0955 ________ ______

7.26.3 ENSURE accumulator pressures between 616.3 and 643.7 psig AND RECORD indicated pressure from the Plant Computer.

- ACC 1A(2A) SIPA0960 ________ / SIPA0961 ________ ______
- ACC 1B(2B) SIPA0962 ________ / SIPA0963 ________ ______
- ACC 1C(2C) SIPA0964 ________ / SIPA0965 ________ ______

7.26.4 VERIFY accumulator level and pressure annunciators clear.

- “ACC TK 1A(2A) PRESS HI/LO” (1M02-A-3) ______
- “ACC TK 1B(2B) PRESS HI/LO” (1M02-A-5) ______
- “ACC TK 1C(2C) PRESS HI/LO” (1M02-A-7) ______
- “ACC TK 1A(2A) LEVEL HI/LO” (1M02-B-3) ______
- “ACC TK 1B(2B) LEVEL HI/LO” (1M02-B-5) ______
- “ACC TK 1C(2C) LEVEL HI/LO” (1M02-B-7) ______

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Safety Injection System accumulator shall be OPERABLE

APPLICABILITY: MODES 1 and 2
MODE 3 with pressurizer pressure > 1000 psig

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration outside the required limits, within 24 hours restore the inoperable accumulator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With more than one accumulator inoperable, except as a result boron concentration outside the required limits, within 1 hour restore at least two accumulators to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With the boron concentration of one accumulator outside the required limit, within 72 hours restore the boron concentration to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

d. With the boron concentrations of more than one accumulator outside the required limit, within 1 hour restore the boron concentration of at least two accumulators to within the required limits or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

1) Verifying the contained borated water volume is ≥ 8800 gallons and ≤ 9100 gallons and nitrogen cover-pressure is ≥ 590 psig[1] and ≤ 670 psig, and

2) Verifying that each accumulator isolation valve is open.

b. At a frequency in accordance with the Surveillance Frequency Control Program and within 6 hours* after each solution volume increase of greater than or equal to 1% of tank volume that is not the result of addition from the RWST by verifying the boron concentration of the accumulator solution is ≥ 2700 ppm and ≤ 3000 ppm, and

c. At a frequency in accordance with the Surveillance Frequency Control Program when the RCS pressure is above 1000 psig by verifying that power to the isolation valve operator is removed.

* The 6 hr. SR is only required to be performed for affected accumulators.

[1] For Unit 1 only, the nitrogen cover-pressure is verified to be ≥ 500 psig for the remainder of Unit 1 Cycle 23.

EQUIPMENT DESCRIPTION	NOTES	TOTAL # REQ	TOTAL # OPERABLE		OAS NUMBER/REMARKS, IF APPLICABLE	REFERENCES
			0600-1800	1800-0600		
ECW TRAINS	Includes self-clean strainers, traveling screens & screen wash booster pumps.	3				T. S. 3.7.4
CCW TRAINS		3				T. S. 3.7.3
SI ACCUMULATORS	8822.8-9076.0 gal, 2700-3000 ppm, 616.3-643.7 psig, isol vlvs open w/pwr removed.	3				T. S. 3.5.1
RHR TRAINS		3				T. S. 3.5.6 T. S. 3.5.2
HHSI TRAINS		3				T. S. 3.5.2
LHSI TRAINS	An operable LHSI Train must have its respective RHR HX Temperature Control Valve fully OPEN AND RHR HX Bypass Valve fully CLOSED. RHR HX Temperature Control Valve and RHR HX Bypass Valve in each train also required to be deenergized.	3				T. S. 3.5.2
CONTAINMENT VENTILATION SYSTEM ISOLATION DAMPERS	All containment ventilation inside and outside isolation dampers operable w/ 48 inch isol. dampers locked closed.	SAT/ UNSAT				T. S. 3.6.1.7
ESSENTIAL CHILL WATER SYSTEM LOOPS		3				T. S. 3.7.14
FHB EXHAUST AIR SYSTEM	1) System requires 1 exhaust air filter train, 1 exhaust booster fan, 1 main exhaust fan, and associated dampers. 2) Applicable during movement of fuel within the spent fuel pool <u>OR</u> when conducting crane operation with loads over the spent fuel pool.	SAT/ UNSAT				TRM 3.9.12
SPENT FUEL POOL COOLING SYSTEM	System requires 2 cooling pumps and 2 Spent Fuel Pool HX's.	SAT/ UNSAT				0POP02-FC-0001
CONTROL RM MAKEUP & FILTRATION TRAINS		3				T. S. 3.7.7

Exam Bank No.: 3148 **Last used on an NRC exam:** Never

RO Sequence Number: 31

Unit 1 is at 50% power.

- PRT level and pressure are rising due to inleakage.
- Annunciator LAMPBOX 4M07 D-1 'PRT PRESS HI' is in alarm.

If PRT pressure continues to rise, the PRT ________(1)_______ will limit PRT pressure.

The condensed steam and water from the PRT released to the RCB will be seen first on _________(2)________.

A. (1) relief valves
   (2) LI-7812, Containment Normal Sump Level

B. (1) relief valves
   (2) SI-LT-3925, Containment Water Level Wide Range

C. (1) rupture discs
   (2) LI-7812, Containment Normal Sump Level

D. (1) rupture discs
   (2) SI-LT-3925, Containment Water Level Wide Range

Answer: C (1) rupture discs
(2) LI-7812, Containment Normal Sump Level

Exam Bank No.: 3148 **Source:** Modified **Modified from** 1661

K/A Catalog Number: 007 K3.01 Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

RO Importance: 3.3 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.04 **Objective Number:** 91014

91014 - DESCRIBE the overpressure protection scheme for the PRT.
LOT201.01 91026 - DESCRIBE the purpose of the following controls and instrumentation and their location(s) for monitoring and indications: Containment sump

Reference: LOT 201.04 Power Point slide 62-63; Instruction Manual FCI Level Probe

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because many vessels/tanks have a relief valve for overpressure protection, however, the PRT utilizes two rupture discs. The second part is correct.

B: INCORRECT. Plausible because many vessels/tanks have a relief valve for overpressure protection, however, the PRT utilizes two rupture discs. It is reasonable to believe that a PRT discharge into containment would first be indicated by SI-LT-3925, Containment Water Level Wide Range but a PRT rupture will collect in the containment normal sump and will indicate first on LI-7812, not on containment wide range water level.

C: CORRECT. The PRT is protected by two rupture discs which relieve to the RCB atmosphere. A discharge from the PRT would collect in the containment normal sump and indicate on LI-7812. Containment wide range water level indication is from discrete sensors mounted on the RCB wall and would take several thousand gallons to rise to this level.

D: INCORRECT. Plausible because it is reasonable to believe that a PRT discharge into containment would first be indicated by SI-LT-3925, Containment Water Level Wide Range but a PRT rupture will collect in the containment normal sump and will indicate first on LI-7812, not on containment wide range water level. The first part is correct.

Question Level: F **Question Difficulty** 2

Justification:

The student must have a knowledge of PRT design and RCB level indications.

Source for RO #31

STP LOT-20 Audit-1 EXAM Print Date 7/28/2022

Exam Bank No.: 1661 Last used on an NRC exam: 2014

Given the following conditions:

- Unit 1 is operating at full power
- PRT level is rising due to Reactor Makeup Water inleakage
- The PRT PRESS HI alarm annunciates
- The operator verifies the alarm is valid

If PRT pressure continues to rise, which of the below correctly describes the NEXT system action that will occur and the prescribed action for the operator to take that will prevent this occurrence from happening?

A. The PRT relief valve will lift and discharge to the RCB Normal Sump. The operator should lower PRT pressure by venting the PRT.

B. The PRT relief valve will lift and discharge to the RCB Normal Sump. The operator should lower PRT pressure by cooling the PRT.

C. The PRT Rupture Disc will relieve to the RCB atmosphere. The operator should lower PRT pressure by cooling the PRT.

D. The PRT Rupture Disc will relieve to the RCB atmosphere. The operator should lower PRT pressure by venting the PRT.

Answer: D The PRT Rupture Disc will relieve to the RCB atmosphere. The operator should lower PRT pressure by venting the PRT.

Exam Bank No.: 1661 **Source:** Bank **Modified from:**

K/A Catalog Number: 007 A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Exceeding PRT high-pressure limits

RO/SRO Importance: 3.2 / 3.6 **Tier:** 2 **Group/Category:** 1

RO-10CFR55.41 # 10 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 201.04 **Objective Number:** 91014

DESCRIBE the overpressure protection scheme for the PRT.

Reference: POP09-AN-04M7, window D1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT - Plausible because many vessels/tanks utilize a relief valve for overpressure protection, however the PRT does not have a relief valve.

B: INCORRECT - Plausible because many vessels/tanks utilize a relief valve for overpressure protection, however the PRT does not have a relief valve. Depending on the source of the influent, cooling the PRT could lower pressure (but in this case the influent is subcooled).

C: INCORRECT - Plausible because there is procedural direction for cooling and venting the PRT, but since the influent is subcooled, cooling will not give the desired results.

D: CORRECT: Overpressure protection for the PRT is provided by a rupture disc. Since RMW is entering the PRT, it must be vented to lower pressure.

Level F **Difficulty** 3

Justification:

The applicant requires an knowledge of PRT design and procedural requirements for a high pressure condition.

OBJ #11

Pressurizer Relief Tank (PRT)

SPARGER LINE

- Relief discharge line in PRT maintained under water to condense steam discharge and minimize pressure excursions in PRT

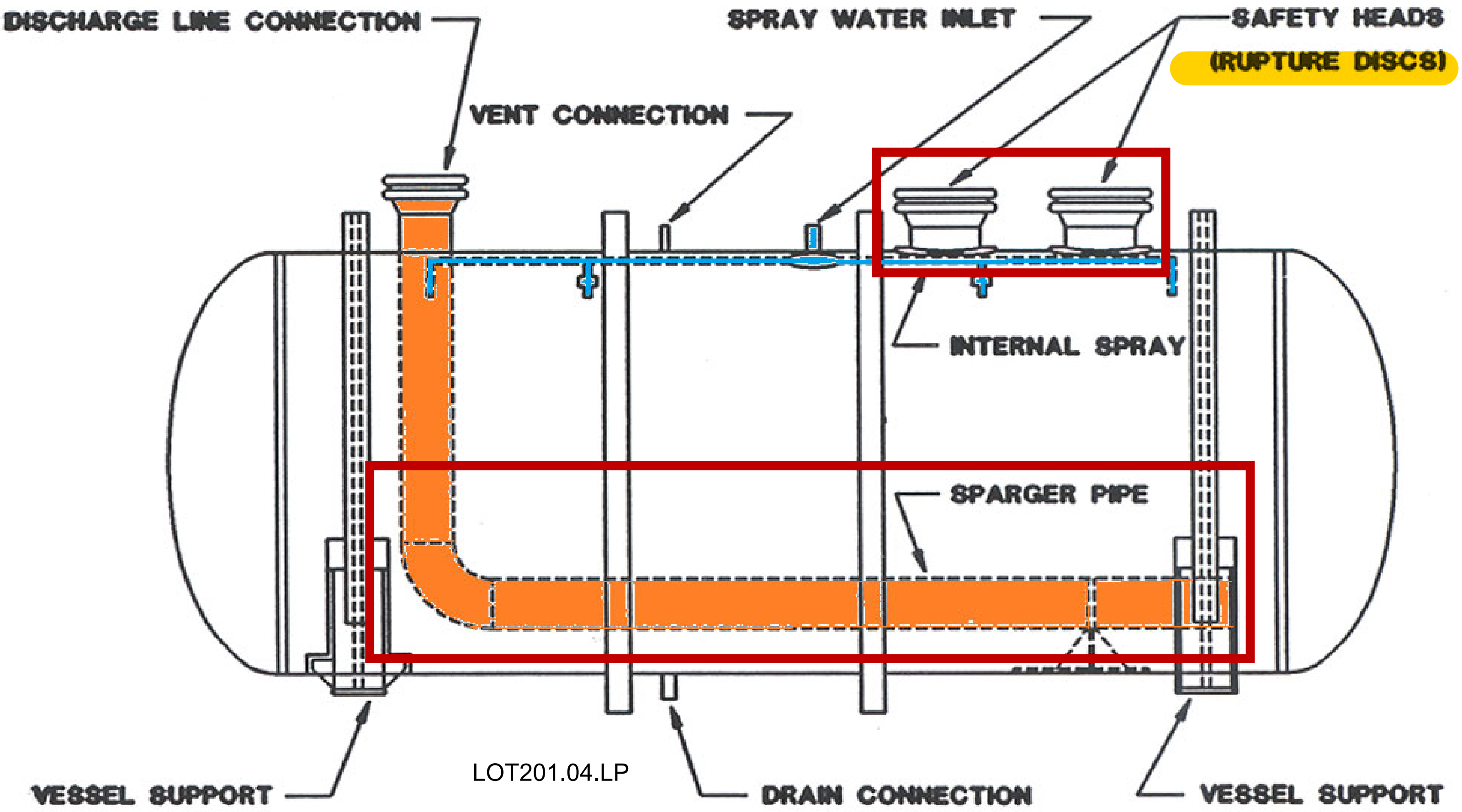
RUPTURE DISCS

- Provided on the PRT to ensure tank integrity with full discharge from all PZR safeties or break vacuum if PRT is cooled with no nitrogen addition
- 100 psig release pressure
- capacity 8E+05 lbm/hr each
  - exceeds capacity of safeties and PORVs

PRESSURIZER RELIEF TANK

OBJ #11

Protection from over-pressurization by 2 rupture discs



	0PGP04-ZA-0328	Rev. 18	
Engineering and Vendor Document Processing			
Form 1	Controlled Document/Record Cover Sheet	Page 1	

0PGP04-ZA-0328 Rev 18	Controlled Document Cover Sheet	Form 1

VENDOR TECHNICAL INFORMATION

Instruction Manual Model 8-66 MA
FCI Level Probe

SOUTH TEXAS PROJECT NUCLEAR OPERATING COMPANY	**STP UNIQUE DOC. NUMBER** 4374--00037KF STI#34676326 8374--00037KF STI#34676327			REV/ SUB C
	PRIORITY	FSUG NO.	DCP/EC/NDCC NO.	
	5	D070907	DCP 18-3094-3 SUPP. 0	

REV./ SUB. NO.	REVISION DESCRIPTION	PREP	CKR	DV (JPR 9543)	RS	APPROVAL DATE
C	Implement DCNs JM00246, JM00316, JM00665, JM00241	RW	JM	GW	JRR	5/23/18

1. SCOPE

1.1 This Manual describes installation, operation, maintenance, and trouble shooting procedures relevant to Model 8-66MA 2, 9, and 12 Point Level Sensors, as supplied to South Texas Project. The use of this Manual provides all of the information required to operate and maintain this equipment.

2. APPLICABLE DOCUMENTS

2.1 Drawing Number 84-261301 Final Assembly and Installation, is needed to install this equipment.

3. DESCRIPTION

3.1 General:

The Model 8-66MA consists of a 2, 9, and 12 point level sensing array, and an associated remote electronics assembly. It is designed to supply a 4-20mA signal which is a function of the number of points on the array which are wet. Signal currents are tabulated on Drawing Number 84-261301.

3.1.1 Circuit Description:

Each level sensing element consists of two platinum resistance temperature detectors (RTD's) and a low power heating element, which is thermally connected to one of the RTD's. When powered, a temperature difference (and therefore a resistance difference) exists between the RTD's; the magnitude of which depends on whether the sensors are dry or wet. The RTD's form the two legs of a half-bridge, which converts the resistance change to a voltage change. This voltage is inputted to an isolation amplifier/filter located on a signal pro-

05

cessor board in the electronics assembly. The output of this amplifier is then compared to an adjustable reference voltage, which has been set to a value between the wet and dry signal voltages.

When the sensor is dry, the comparator output is driven high and a green LED is lit. When the sensor is wet, the comparator is driven low, the LED is extinguished, and an output voltage generated. The output voltages for all of the channels on one signal processor board are summed on a signal output line. The output voltages of both of the signal processor boards are summed on the transmitter board, and a proportional 4-20mA signal is generated.

In addition, the electronics assembly includes a power supply which generates a regulated +28 volts, which powers the electronics, and a regulated current to excite the array.

3.2 Operating Characteristics - Electrical/Electronic:

3.2.1 AC Power Input: 108-132 VAC, 60 Hz., 80 VA Max.

3.2.2 Recommended AC Line Fuse Protection: 1 Amp Slo-Blo (2 points); 1½ Amp Slo-Blo (9, 12 points).

3.2.3 Electronic Assembly Operating Ambient Temperature Range: 0°F to 130°F.

3.2.4 Maximum Remote Cable Allowable: 1000 ft. of 12-gauge conductor cable.

4. INSTALLATION

4.1 Mounting and Wiring:

Refer to Drawing Number 84-261301.

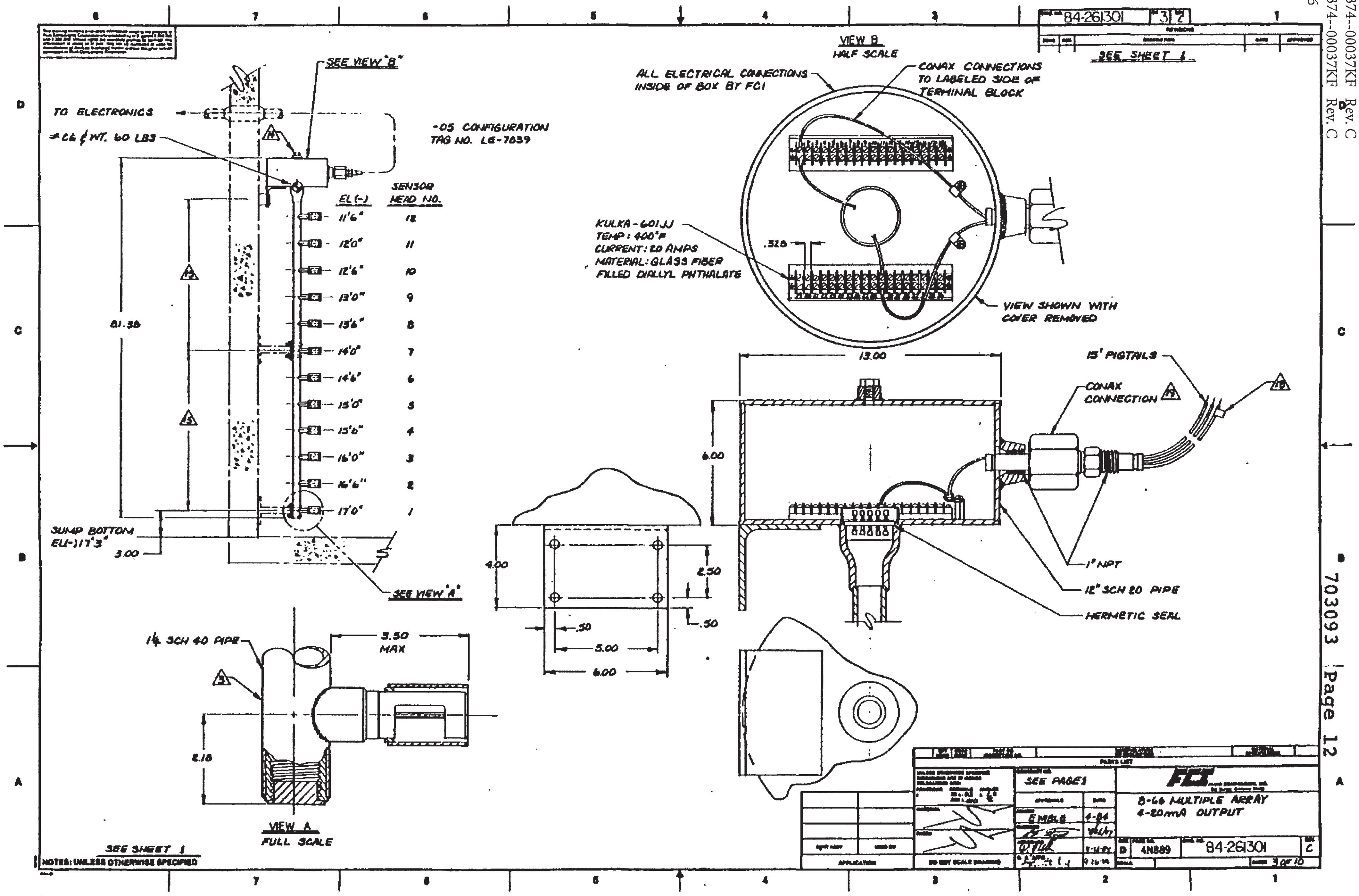

84-261301
3
2
SEE SHEET 1
SEE VIEW "B"
TO ELECTRONICS
⌀ CG & WT. 60 LBS
-05 CONFIGURATION
TAG NO. LE-7639
EL(-)
SENSOR HEAD NO.
11'6" 12
12'0" 11
12'6" 10
13'0" 9
13'6" 8
14'0" 7
14'6" 6
15'0" 5
15'6" 4
16'0" 3
16'6" 2
17'0" 1
81.38
SUMP BOTTOM
EL(-)17'3"
3.00
SEE VIEW "A"
1¼ SCH 40 PIPE
3.50
MAX
2.18
VIEW A
FULL SCALE
SEE SHEET 1
NOTES: UNLESS OTHERWISE SPECIFIED
VIEW B
HALF SCALE
ALL ELECTRICAL CONNECTIONS
INSIDE OF BOX BY FCI
CONAX CONNECTIONS
TO LABELED SIDE OF
TERMINAL BLOCK
KULKA-601JJ
TEMP: 400°F
CURRENT: 20 AMPS
MATERIAL: GLASS FIBER
FILLED DIALLYL PHTHALATE
.328
VIEW SHOWN WITH
COVER REMOVED
13.00
6.00
15' PIGTAILS
CONAX
CONNECTION
1" NPT
12" SCH 20 PIPE
HERMETIC SEAL
4.00
2.50
.50
.50
5.00
6.00
SEE PAGE 1
FCI
8-66 MULTIPLE ARRAY
4-20mA OUTPUT
E MABLE
4-84
D
4N889
84-261301
C
3 OF 10
APPLICATION
DO NOT SCALE DRAWING

FIGURE 3

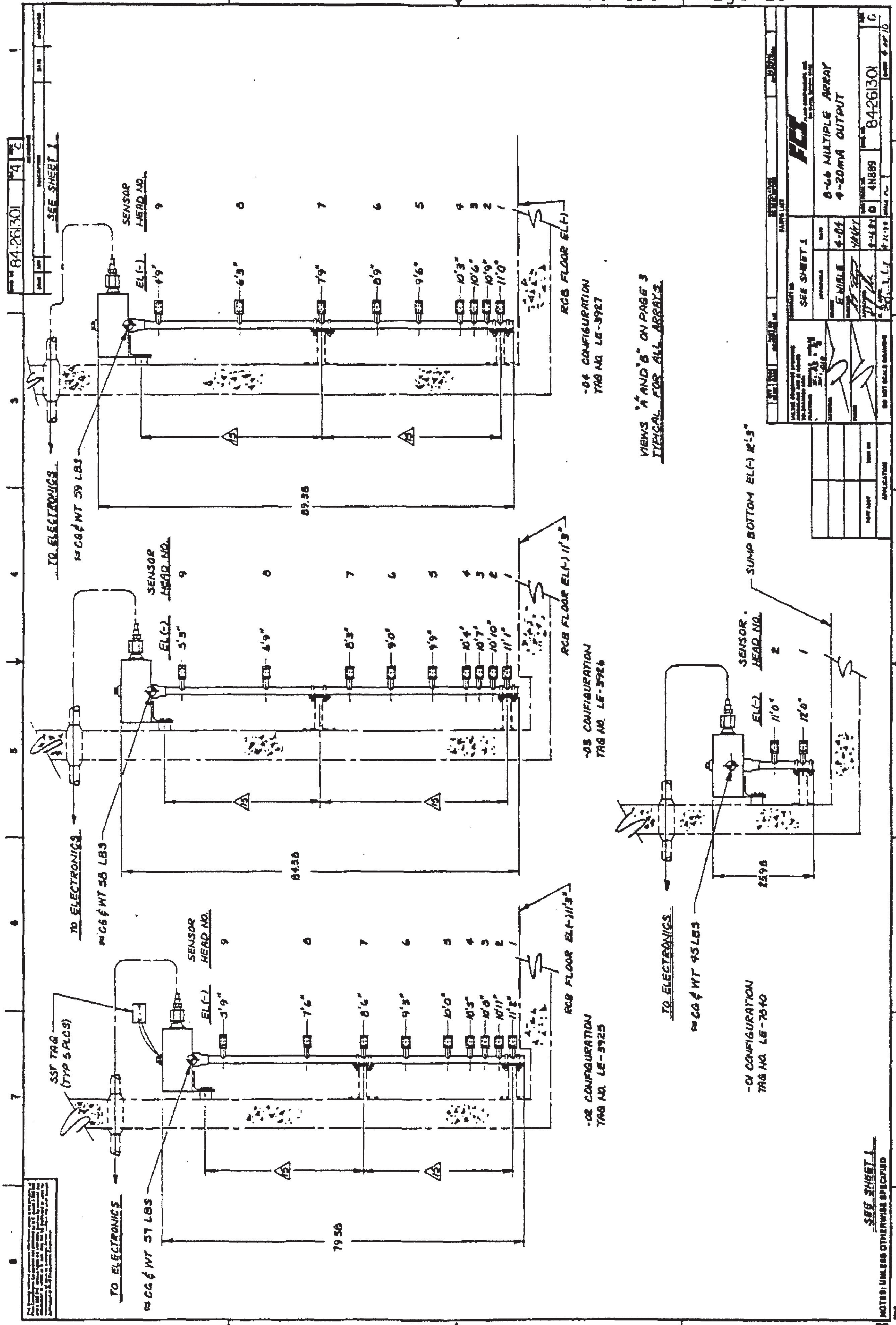

84261301
4
C
SEE SHEET 1
SENSOR HEAD NO.
EL(-)
4'9"
6'3"
7'9"
8'9"
9'6"
10'3"
10'6"
10'9"
11'0"
-04 CONFIGURATION TAG NO. LE-3927
RCB FLOOR EL(-)
89.38
TO ELECTRONICS
≈CG & WT 59 LBS
VIEWS "A" AND "B" ON PAGE 3 TYPICAL FOR ALL ARRAYS
5'3"
6'9"
8'3"
9'0"
9'9"
10'4"
10'7"
10'10"
11'1"
-03 CONFIGURATION TAG NO. LE-3926
RCB FLOOR EL(-) 11'3"
84.38
≈CG & WT 58 LBS
SUMP BOTTOM EL(-) 12'-3"
11'0"
12'0"
-01 CONFIGURATION TAG NO. LE-7340
25.98
≈CG & WT 45 LBS
SST TAG (TYP 5 PLCS)
3'9"
7'6"
8'6"
9'3"
10'0"
10'5"
10'8"
10'11"
11'2"
-02 CONFIGURATION TAG NO. LE-3925
RCB FLOOR EL(-) 11'3"
79.38
≈CG & WT 57 LBS
B-66 MULTIPLE ARRAY 4-20mA OUTPUT
84261301
D
4N889
E. WISLE
4 of 10
NOTES: UNLESS OTHERWISE SPECIFIED
SEE SHEET 1

FIGURE 4

Exam Bank No.: 3150 **Last used on an NRC exam:** Never

RO Sequence Number: 32

A Reactor Trip and Safety Injection occurred on Unit 1.

- RCS pressure is 1845 psig and lowering.
- RCS temperature is 490ºF and lowering.
- Containment pressure is 13 psig and rising.

Which of the following describes the loads that are being cooled by the Component Cooling Water System?

1. Reactor Containment Fan Coolers
2. Residual Heat Removal Heat Exchangers
3. Spent Fuel Pool Heat Exchangers
4. RCP Thermal Barrier Heat Exchangers

A. 1 and 2

B. 2 and 3

C. 1 and 4

D. 3 and 4

Answer: A 1 and 2

Exam Bank No.: 3150 **Source:** New **Modified from** N/A

K/A Catalog Number: 008 A3.04 Ability to monitor automatic operation of the CCWS, including: Requirements on and for the CCWS for different conditions of the power plant

RO Importance: 2.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.12 **Objective Number:** 80107

Given a plant or system condition, PREDICT the operation of the Component Cooling Water System

Reference: LOT 201.12 Power Point slides 39 and 40

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. On a Safety Injection, cooling to the Reactor Fan Cooler Units swaps to CCW and is established to the RHR Heat Exchangers while flow is isolated to the SFP Cooling HXs. Containment Phase B Isolation occurs at ≥ 9.5 psig and isolates CCW to the RCP Thermal Barrier HXs.

B: INCORRECT. Plausible because it is desirable to maintain cooling of the irradiated fuel assemblies in the spent fuel pool.

C: INCORRECT. Plausible because it is desirable to maintain cooling to the RCP seals even if they are secured.

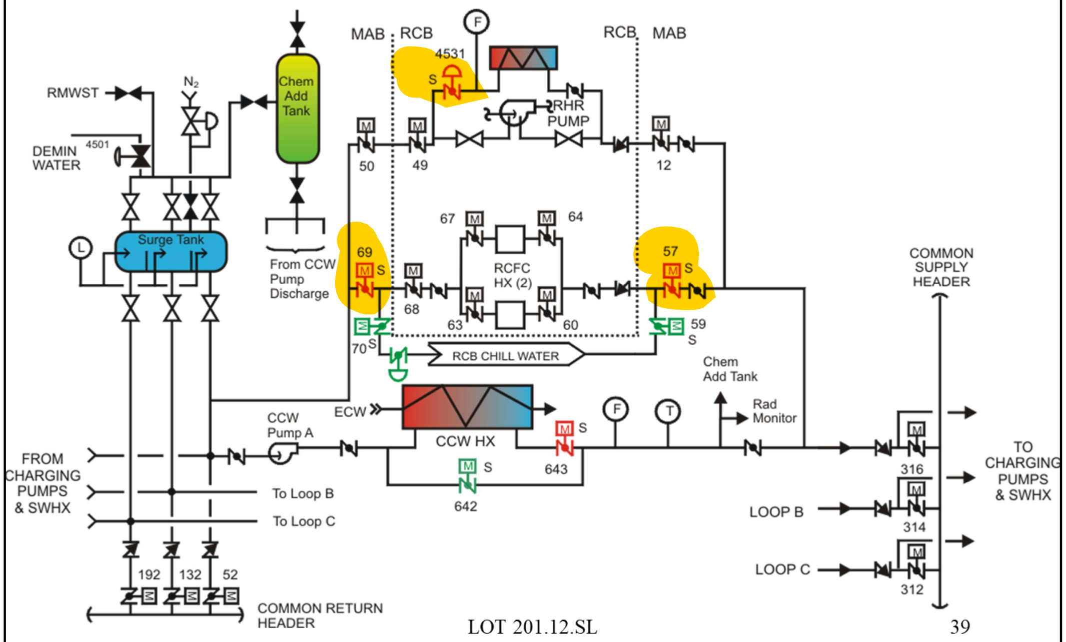
D: INCORRECT. Plausible because it is desirable to maintain cooling to the RCP seals even if they are secured and it is desirable to maintain cooling of the irradiated fuel assemblies in the spent fuel poo.

Question Level: H **Question Difficulty** 3

Justification:

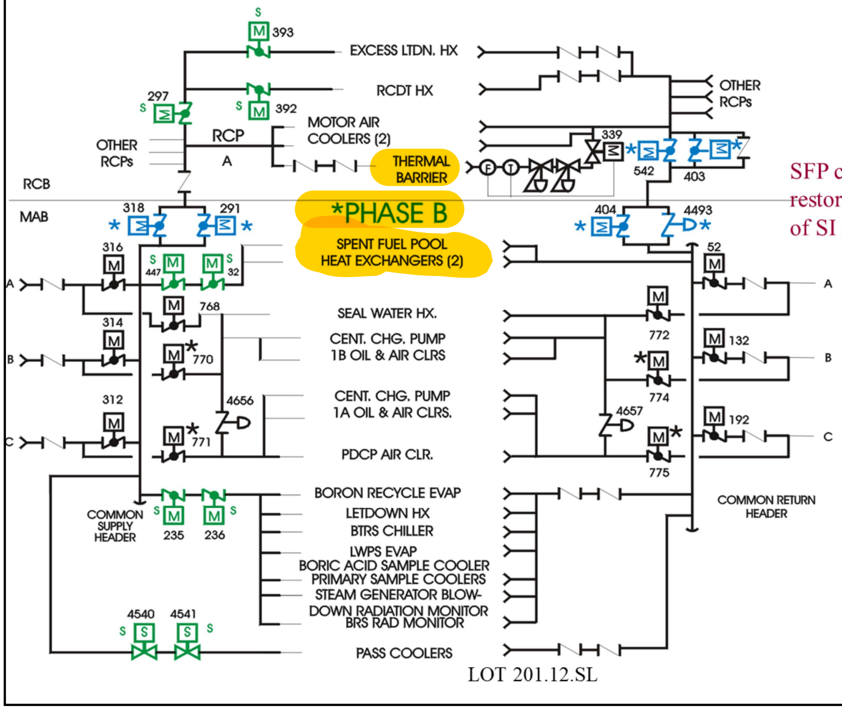
The student must assess plant conditions and the effect on CCW components.

ESF LOOP - SI



- Flow established through RHR Hx
- RCFC cooling swaps to CCW
- Hx outlet goes full open, bypass full closed

COMMON HEADER - SI



SFP cooling is manually restored within 2.5 hours of SI actuation

Exam Bank No.: 2755 **Last used on an NRC exam:** 2018

RO Sequence Number: 33

To control RCS pressure using Auxiliary Spray, the RO should ______(1)_____.

Auxiliary Spray is used ________(2)_________.

A. (1) start an additional charging pump
(2) to assist Normal Pressurizer Spray

B. (1) start an additional charging pump
(2) when Reactor Coolant Pumps are not running

C. (1) close both the Normal and Alternate Charging Loop MOVs
(2) to assist Normal Pressurizer Spray

D. (1) close both the Normal and Alternate Charging Loop MOVs
(2) when Reactor Coolant Pumps are not running

Answer: D (1) close both the Normal and Alternate Charging Loop MOVs (2) when Reactor Coolant Pumps are not running

Exam Bank No.: 2755 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 010 K1.06 Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following: CVCS

RO Importance: 2.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.06 **Objective Number:** 48241

Describe the CVCS system flowpaths to include: 2. Charging (Normal, Alternate, and Auxiliary Spray)

Reference: LOT 201.04 PPT slide 45, 46, 0POP01-ZA-0018A Addendum 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible if the student believes that since the normal spray valves are capable of 525 gpm, and the charging pump can only provdie 200 gpm flow, then starting an additional charging pump would be effective and if the student believes there is a need to augment normal spray if < 4 RCPs are running.

B: INCORRECT: Plausible if the student believes that since the normal spray valves are capable of 525 gpm, and the charging pump can only provide 200 gpm flow, then starting an additional charging pump would be effective. The second part is correct.

C: INCORRECT: Plausible as closing off charging to the RCS loops is required and if the student believes there is a need to augment normal spray if < 4 RCPs are running.

D: CORRECT: To utilize Auxiliary Spray, the RO must close all charging to the RCS loops to force charging flow through the auxiliary spray line. When RCPs are not running, normal spray is not available.

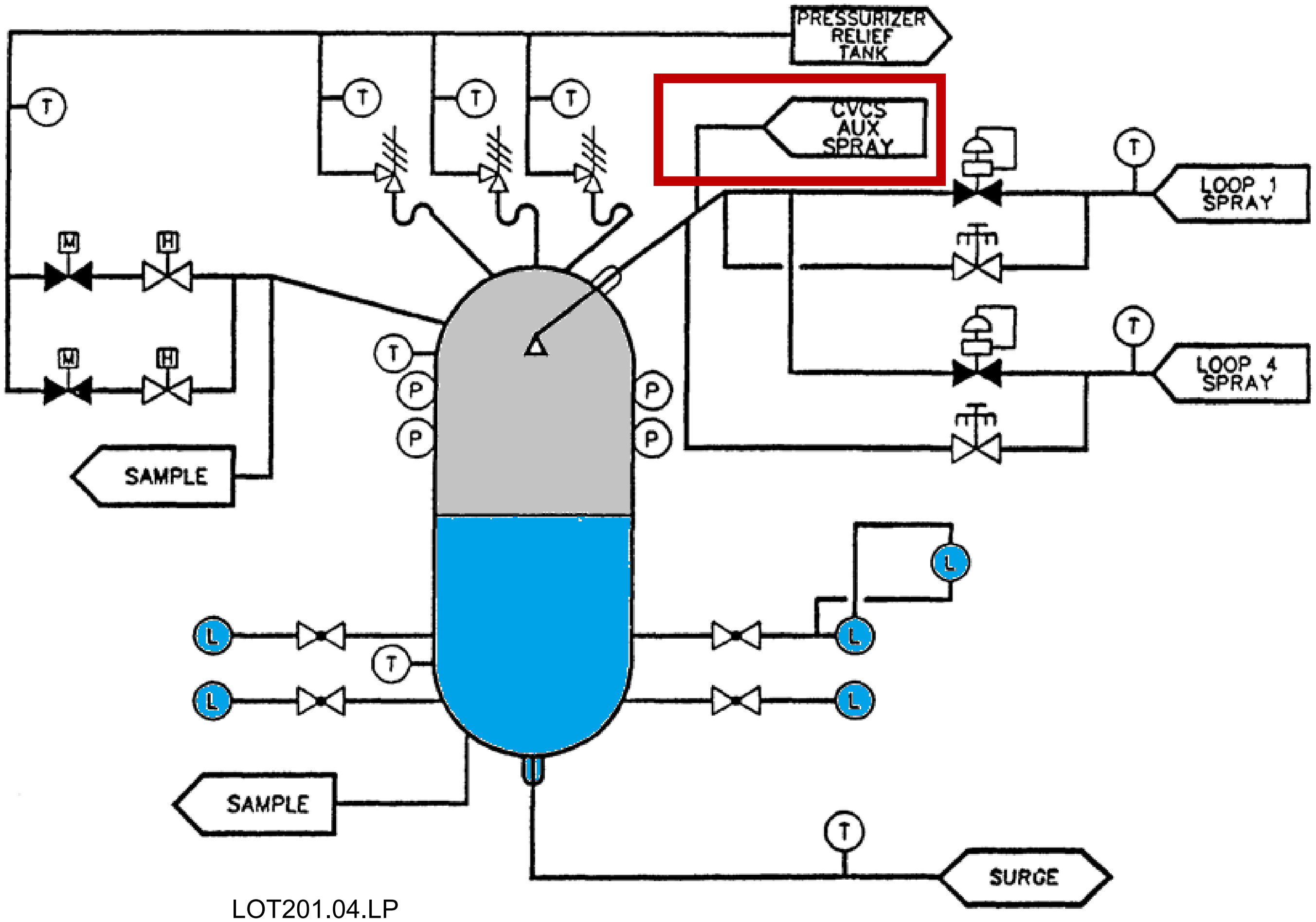
Question Level: F **Question Difficulty** 2

Justification:

The student must know the auxiliary spray flowpath and that it is used when RCPs are not running.

OBJ #1

Auxiliary Spray Line



LOT201.04.LP

OBJ #1

Auxiliary Spray

- Allows for spray flow and pressure reduction when the RCPs are NOT operating
- Line taps off the regenerative HX (RHX) outlet, same as normal makeup
- To force water up to the spray nozzle, both RCS loop isolation valves from RHX must be closed.

Addendum 2 Establishing Auxiliary Spray Page 1 of 1

1. ENSURE at least one CCP is in-service. (REFER TO Addendum 1, Establishing Charging Flow) ______
2. ENSURE Phase A is reset ______
3. ENSURE OPEN IA OCIV FV-8565. ______
4. ENSURE the normal Pressurizer spray valves are CLOSED.
   - SPRAY CONT LOOP A PCV-0655C ______
   - SPRAY CONT LOOP D PCV-0655B ______
5. OPEN Pressurizer "AUX SPRY" Valve, LV-3119. ______

NOTE

- Pressurizer AUX spray flow will **NOT** be established until the loop isolation valves are closed. Opening one of the loop isolation valves will secure Pressurizer AUX spray flow.
- Normal Pressurizer spray and CVCS loop isolation valves **CLOSED** will deliver full flow for maximum depressurization. Opening and closing normal Pressurizer spray valve(s) will lower or raise the depressurization rate.
- CCP flow should be maintained LESS THAN 240 gpm to prevent pump runout.

6. ENSURE CVCS loop isolation valves CLOSED.
   - CV-MOV-0003 ______
   - CV-MOV-0006 ______

CAUTION

WHEN directed to depressurize at "MAX" rate AND AUX spray is in use, THEN controlling AUX spray flow using the normal Pressurizer spray valves SHALL **NOT** be used.

7. Manually ADJUST normal Pressurizer spray valves to control depressurization rate. ______
8. WHEN ready to secure Pressurizer AUX spray, THEN PERFORM the following:
   a. OPEN one of the CVCS loop isolation valve ______
      - CV-MOV-0003

        OR
      - CV-MOV-0006

   b. CLOSE "AUX SPRY" LV-3119 ______

   c. RESTORE normal Pressurizer spray valves as directed by US. ______

Exam Bank No.: 3152 **Last used on an NRC exam:** Never

RO Sequence Number: 34

Given the following:

- Unit 1 is at 48% power following a load reduction.

Which of the following annunciators in alarm will result in an automatic Reactor Trip?

A. 6M05-C-2 “TURB THR BRG TRIP”

B. 5M02-E-2 “RCP TRIP”

C. 5M03-B-5 “ROD CONT URGENT ALARM”

D. 6M03-A-8 “QDPS ALARM SGWLCS”

Answer: B 5M02-E-2 "RCP TRIP"

Exam Bank No.: 3152 **Source:** New **Modified from** N/A

K/A Catalog Number: 012 G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm: Reactor Protection

RO Importance: 4.1 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.20 **Objective Number:** 91161

LIST the alarms associated with the solid state protection system.

Reference: 0POP09-AN-05M02 Window E-2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the alarm condition will cause a reactor trip but not for the conditions given in the stem.

B: CORRECT. If reactor power is above P-8 (40%) as given in the stem, a reactor trip occur, and the alarm response procedure directs the operator to enter 0POP05-EO-EO00.

C: INCORRECT. Plausible because the student may confuse the alarm condition that results in the rods locking up to prevent rod motion with a requirement instead to trip the reactor.

D: INCORRECT. Plausible because it will affect 4 SG NR level channels all of which provide inputs to the SSPS but is only affects one channel per SG and it takes 2/4 channels on an individual SG to get a trip.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the power level and conditions given and draw a conclusion.

RCP TRIP

Automatic Actions: 1) IF Reactor Power is above P-8 (40%), THEN a Reactor Trip will occur.

Immediate Actions: None

Subsequent Actions:

______ 1) IF the Reactor is critical AND an RCP trips, THEN PERFORM the following:

______ a) TRIP the reactor.

______ b) PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection.

______ c) CONTINUE actions of this procedure as resources permit.

______ 2) DISPATCH an Operator to the following locations to determine cause of trip:

______ a) Affected 13.8 KV bus RCP breaker (31 ft TGB N SWGR Rm.)

______ b) Associated Electrical Penetration Space RCP cubicle.

______ 3) TAKE appropriate action per Technical Specification 3.2.5, 3.4.1.1, 3.4.1.2 or 3.4.1.3.

______ 4) CONSULT Plant Management to DETERMINE RCP recovery actions.

Probable Causes:

1) RCP undervoltage
2) RCP underfrequency
3) RCP overcurrent

5M02-E-2

RCP
TRIP

Exam Bank No.: 1117 **Last used on an NRC exam:** 2018

RO Sequence Number: 35

Unit 2 is operating at 75% power.

- A steam pressure instrument on the steam line from SG ‘A’ is out of service and in the tripped condition.

An ESF actuation will occur if a...

A. second steam pressure instrument on steam line ‘A’ fails low.

B. steam pressure instrument from any other steam line fails low.

C. second steam pressure instrument on steam line ‘A’ detects steam pressure dropping at greater than 100 psig/50 seconds.

D. steam pressure instrument from any other steam line detects steam pressure dropping at greater than 100 psig/50 seconds.

Answer: A second steam pressure instrument on steam line ‘A’ fails low.

Exam Bank No.: 1117 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 013 K6.01 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors

RO Importance: 4.1 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.22 **Objective Number:** 80802

Describe the function of the instrumentation and controls available for operation and monitoring of the engineered safety features.

Reference: LOT 202.02 PPT slide 157

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: Tripping of more that 1 channel per Steam line at a time will initiate a Low Steam Line Tressure SI and a reactor trip.

B: INCORRECT: Plausible as this is a total of 2 channels failed, but must be on the same steam line.

C: INCORRECT: Plausible as this is a signal for MSLI but need 2 of the same type of signal and only applies below P-11 with Lo Steam Line Pressure SI Blocked

D: INCORRECT: Plausible as this is a signal for MSLI but need 2 of the same type of signal on 1 steam generator and only applies below P-11 with Lo Steam Line Pressure SI Blocked.

Question Level: H **Question Difficulty** 2

Justification:

Student must analyze the given conditions and know the logic for ESFAS actuations.

Main Steam Isolation Signals

- Low Compensated Steam Line Press
  - 735 psig
  - 2 of 3 detectors in 1 of 4 steam lines
  - Manually blocked below P-11 (RCS Pressure 1985 psig)

Exam Bank No.: 3153 **Last used on an NRC exam:** Never

RO Sequence Number: 36

Unit 1 is at 100% power.

- Reactor Containment Fan Coolers (RCFCs) 11A, 12A, 11B, and 12B are in service and RCFCs 11C and 12C are in standby.

Subsequently, the 13.8 KV Standby Bus 1H loses power.

Power will be temporarily unavailable to RCFCs _____(1)_____. 4 minutes later _____(2)_____ RCFCs will be running.

A. (1) 11B and 12B
(2) 6

B. (1) 11B and 12B
(2) 4

C. (1) 11C and 12C
(2) 6

D. (1) 11C and 12C
(2) 4

Answer: C (1) 11C and 12C (2) 6

Exam Bank No.: 3153 **Source:** New **Modified from** N/A

K/A Catalog Number: 022 K2.01 Knowledge of power supplies to the following: Containment cooling fans

RO Importance: 3.0 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 202.33 **Objective Number:** 51319

STATE the power supplies for the RCB-HVAC systems.

Reference: 0PSP03-EA-0002 pg 27; 0POP02-HC-0001 Pg 31, LOT 202.33 PPT slide 21

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student thinks that Bus 1H supplies normal power to the Train B Class 1E buses. The second part is correct.

B: NCORRECT. Plausible if the student thinks that Bus 1H supplies normal power to the Train B Class 1E buses and that the Train B RCFCs will sequence on after power is restored to the Train B Class 1E buses.

C: CORRECT. RCFCs 11C and 12C are powered from Class 1E 480 V Load Center E1C. 13.8 KV Standby Bus 1H supplies normal power to the Train C Class 1E buses. Losing that bus will cause Standby Diesel Generator 13 to start and load onto Class 1E 4160 V Bus E1C which restores power to 480 V Load Center E1C. The sequencer will then start the 11C and 12C RCFCs and all 6 RCFCs will be running.

D: INCORRECT. Plausible because student may believe that RCFCs 11C and 12C will not sequence on if they were not previously running. The first part is correct.

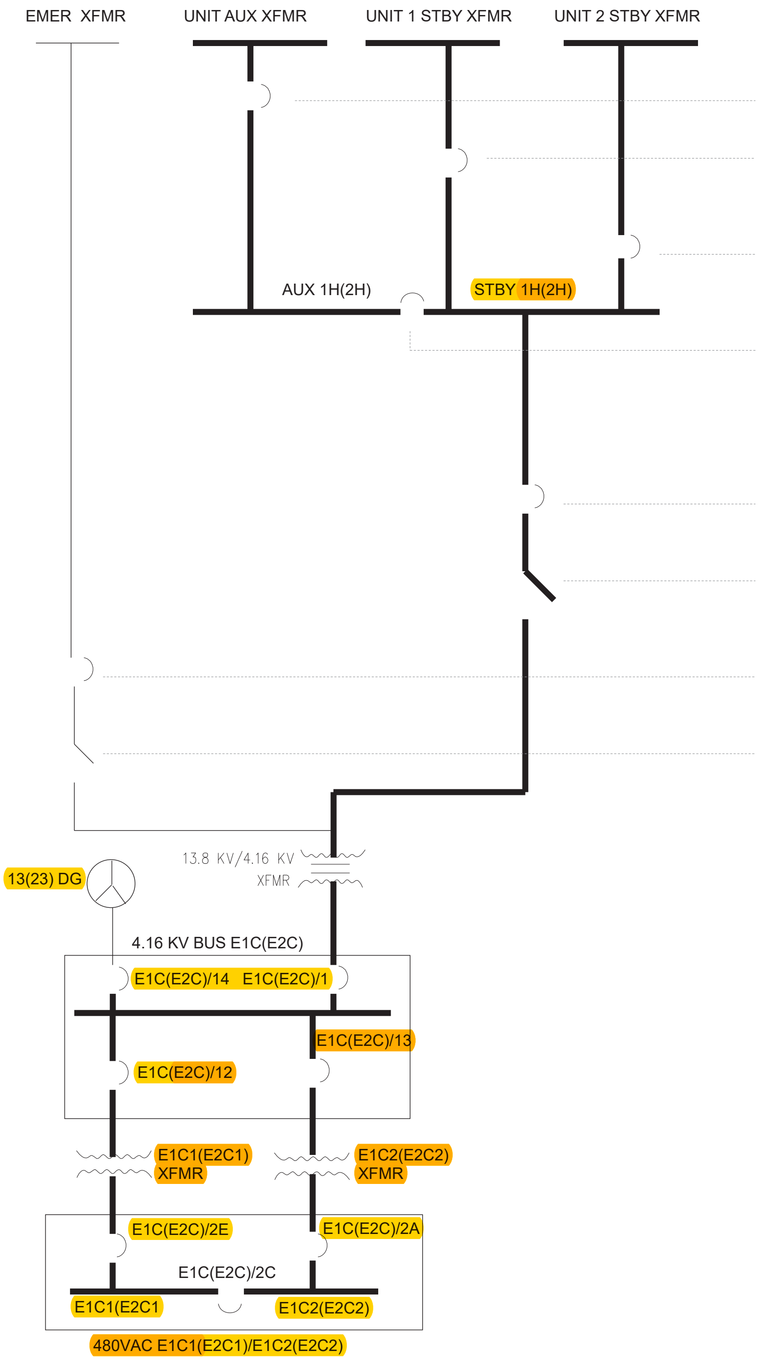
Question Level: H **Question Difficulty** 3

Justification:

The student must analyze given conditions and have knowledge of power distribution to the Reactor Containment Fan Coolers to determine the affected RCFCs and the number of RCFCs running.

Data Sheet 4	Required ESF Power Train C	Page 1 of 1

UNIT 1 (Circle Unit Performing Data Sheet) **UNIT 2**



DESIGNATE (✓) Class 1E 4160 VAC Bus Power Source (note 2)	(1) ____ UNIT AUX XFMR	(2) ____ UNIT 1 STBY XFMR	(3) ____ UNIT 1 STBY XFMR	(4) ____ UNIT 2 STBY XFMR	(5) ____ UNIT 2 STBY XFMR
UAT TO AUX BUS 1H(2H) SPLY	CLOSED	CLOSED	OPEN	CLOSED	OPEN
STBY XFMR 1 TO STBY BUS 1H(2H) SPLY	OPEN	CLOSED	CLOSED	OPEN	OPEN
STBY XFMR 2 TO STBY BUS 1H(2H) SPLY	OPEN	OPEN	OPEN	CLOSED	CLOSED
AUX TO STBY BUS 1H(2H) TIE	CLOSED	OPEN	N/A	OPEN	N/A

BREAKER/DISC POSITION	REQ'D	UNSAT
STBY BUS 1H(2H) TO XFMR E1C(E2C) BKR SW-NORM	CLOSED	OPEN
STBY BUS 1H(2H) TO XFMR E1C(E2C) DISC SW-NORM	CLOSED	OPEN
EMER BUS 1L(2L) TO XFMR E1C(E2C) BKR SW-EMER	N/A	N/A
EMER BUS 1L(2L) TO XFMR E1C(E2C) BKR DISC SW-EMER	N/A	N/A
SPLY BKR E1C(E2C)/1	CLOSED	OPEN
DG OUTP BKR E1C(E2C)/14	N/A	N/A
SPLY BKR E1C(E2C)/12	CLOSED	OPEN
SPLY BKR E1C(E2C)/13	CLOSED	OPEN
OUTP BKR E1C(E2C)/2E	CLOSED	OPEN
OUTP BKR E1C(E2C)/2A	CLOSED	OPEN
CROSS TIE E1C(E2C)/2C	OPEN	CLOSED

Note 1, ▬▬▬ Indicates Technical Specification related electrical lineup.
Note 2, Required per Technical Specifications, EITHER UNIT AUX XFMR OR UNIT 1 STBY XFMR OR UNIT 2 STBY XFMR **SHALL** be the power source for 4.16 KV Bus E1C(E2C).

____________________	__________	__________
PERFORMED BY	DATE	TIME

This procedure, when complete, SHALL be retained for five years.

Lineup 2	RCFC Electrical Lineup	Page 2 of 2

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY	NEW TAG NEEDED
E1A1(E2A1)/2D	RCFC FAN 11A(21A) 1(2)-HC-FN-001	EAB 10' SWGR Room LC E1A1(E2A1)	RACKED IN			
E1A2(E2A2)/3A	RCFC FAN 12A(22A) 1(2)-HC-FN-002	EAB 10' SWGR Room LC E1A2(E2A2)	RACKED IN			
DPA435 BKR 27	RCFC FAN 11A(21A) SPC HTR (HC-FN-001)	EAB 10' SWGR Room MCC E1A4(E2A4)/K1	ON			
DPA435 BKR 28	RCFC FAN 12A(22A) SPC HTR (HC-FN-002)	EAB 10' SWGR Room MCC E1A4(E2A4)/K1	ON			
E1B1(E2B1)/2E	RCFC FAN 11B(21B) 1(2)-HC-FN-003	EAB 35' SWGR Room LC E1B1(E2B1)	RACKED IN			
E1B2(E2B2)/3A	RCFC FAN 12B(22B) 1(2)-HC-FN-004	EAB 35' SWGR Room LC E1B2(E2B2)	RACKED IN			
DPB435 BKR 27	RCFC FAN 11B(21B) SPC HTR (HC-FN-003)	EAB 35' SWGR Room MCC E1B4(E2B4)/H1	ON			
DPB435 BKR 28	RCFC FAN 12B(22B) SPC HTR (HC-FN-004)	EAB 35' SWGR Room MCC E1B4(E2B4)/H1	ON			
E1C1(E2C1)/2D	RCFC FAN 11C(21C) 1(2)-HC-FN-005	EAB 60' SWGR Room LC E1C1(E2C1)	RACKED IN			
E1C2(E2C2)/3A	RCFC FAN 12C(22C) 1(2)-HC-FN-006	EAB 60' SWGR Room LC E1C2(E2C2)	RACKED IN			
DPC435 BKR 27	RCFC FAN 11C(21C) SPC HTR (HC-FN-005)	EAB 60' SWGR Room MCC E1C4(E2C4)/J1	ON			
DPC435 BKR 28	RCFC FAN 12C(22C) SPC HTR (HC-FN-006)	EAB 60' SWGR Room MCC E1C4(E2C4)/J1	ON			

Reactor Containment Fan Coolers

- Controls:
  - Remote - Main Control Room on CP002
  - Local - Transfer Switch Panels
    - ZLP700 – Train A
    - ZLP701 – Train B
    - ZLP709 – Train C

- Automatic Actuations:
  - Fans auto start on SI or LOOP.
  - Cooling Water switches from RCB Chilled Water to Component Cooling Water on SI. Cooling Water isolates on LOOP.

Reactor Containment Fan Coolers

- Flow path
  - Suction on ring duct
  - Discharge at RCB lower level
- With containment spray system, RCFCs makeup ESF heat removal system for the RCB.
- RCFC fan motors - powered from 480V Class 1E load centers.
  - Fan 11A – LC E1A, Fan 12A – LC E1A
  - Fan 11B – LC E1B, Fan 12B – LC E1B
  - Fan 11C – LC E1C, Fan 12C – LC E1C

Exam Bank No.: 3154 **Last used on an NRC exam:** Never

RO Sequence Number: 37

Unit 1 experiences a large break LOCA.

The Train ‘A’ Containment Emergency Sump Valve will open to supply CS Pump ‘A’ when RWST level lowers to ___(1)___ gallons and at least one Train ‘A’ HHSI AND one Train ‘A’ LHSI Pump Mini-flow valves are ___(2)___.

A. (1) 32,500
   (2) OPEN

B. (1) 75,000
   (2) OPEN

C. (1) 32,500
   (2) CLOSED

D. (1) 75,000
   (2) CLOSED

Answer: D (1) 75,000 (2) CLOSED

Exam Bank No.: 3154 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 026 K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CSS (Containment Spray) and the following systems: ECCS

RO Importance: 4.2 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.11 **Objective Number:**

LIST the automatic actions/interlocks associated with the Containment Spray System, components and/or controls.

Reference: LOT 201.11 PPT slide 33, LOT 201.10, PPT slide 41

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student confuses swapover level with level that requires pump shutdown and the required position of the HHSI/LHSI Pump Mini-Flow Valves.

B: INCORRECT. Plausible if the student confuses the position of the HHSI/LHSI Pump Mini-Flow Valves as open vs. closed. The first part is correct.

C: INCORRECT. Plausible if the student confuses swapover level with level that requires pump shutdown. The second part is correct.

D: CORRECT. To align the CS pump suctions for cold leg recirculation following a Safety Injection Signal requires a RWST LO-LO level of 75,000 gallons (~14%) and at least one each of two HHSI and LHSI Pump Miniflow Valves must be closed.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of the interlock conditions for auto swapover to the Containment Emergency Sump suction valves for Containment Spray and ECCS.

OBJ # 7 & 9

Automatic Actions

- Pump suction automatic switch-over to Recirculation occurs on SI Signal coincident with Lo-Lo RWST level
  - 14% (75,000 gal) approximate level
  - Signal also sent to LHSI and HHSI Pumps to isolate Mini-Flow Lines to the RWST (which must occur before sump valves open)
  - Must manually close RWST Outlet MOV's (CP-001)

AUTO-RECIRC SIGNAL

- SI AND RWST LO-LO SIGNAL (TRAIN DEPENDENT)
- INITIATES AN AUTO SWAPOVER OF HHSI/LHSI SUCTION TO THE SUMP
  - HHSI/LHSI MINI-FLOWS CLOSE
  - WHEN AT LEAST ONE OF EACH PAIR IS CLOSED, THE RESPECTIVE TRAIN EMERGENCY SUMP OUTLET VALVE OPENS

Exam Bank No.: 3155 **Last used on an NRC exam:** Never

RO Sequence Number: 38

A Unit 1 startup is in progress at End of Life (EOL) in accordance with 0POP03-ZG-0005, Plant Startup to 100%.

- Control Rods are in MANUAL.
- Reactor power is steady at 8%.

A Main Steam Safety Valve fails OPEN.

RCS Tavg will **INITIALLY** _______(1)______ and Reactor Power will **INITIALLY** ______(2)______.

A. (1) rise
(2) rise

B. (1) lower
(2) rise

C. (1) rise
(2) lower

D. (1) lower
(2) lower

Answer: B 1) lower, (2) rise

Exam Bank No.: 3155 **Source:** New **Modified from** N/A

K/A Catalog Number: 039 K5.08 Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

RO Importance: 3.6 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 204.01 **Objective Number:** 114534

Given plant or system conditions, PREDICT the response of the plant and/or systems.

Reference: Chapter 8, Reactor Operational Physics, slide 151 U1 NDR Figure 5-4 and Table 5-4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student understands that an increase in steam flow will add positive reactivity but answers the effect steam dumps will have to recover Tavg and that Reactor Power and steam flow will still be higher.

B: B.CORRECT. A main steam safety valve failing open will immediately increase steam flow by about 6% and cause Tavg to start lowering. As Tavg drops, positive reactivity is added to the core due to the negative MTC at EOL (see U1 NDR reference).. The positive reactivity addition results in Reactor Power initially rising.

C: INCORRECT. Plausible if the student thinks that an increase in steam flow will cause steam dumps to immediately close and total steam flow lowers.

D: INCORRECT. Plausible if the student thinks that an increase in steam flow will cause RCS Tave to lower but believes the steam dumps will close and lower steam flow.

Question Level: H **Question Difficulty** 3

Justification:

The student must have knowledge of the effect an increase in steam demand will have on reactivity and primary plant parameters.

Figure 5-4 FOP Moderator and Isothermal Temperature Coefficients vs. Vessel Average Moderator Temperature at EOL (19139 MWD/MTU), Eq. Xenon

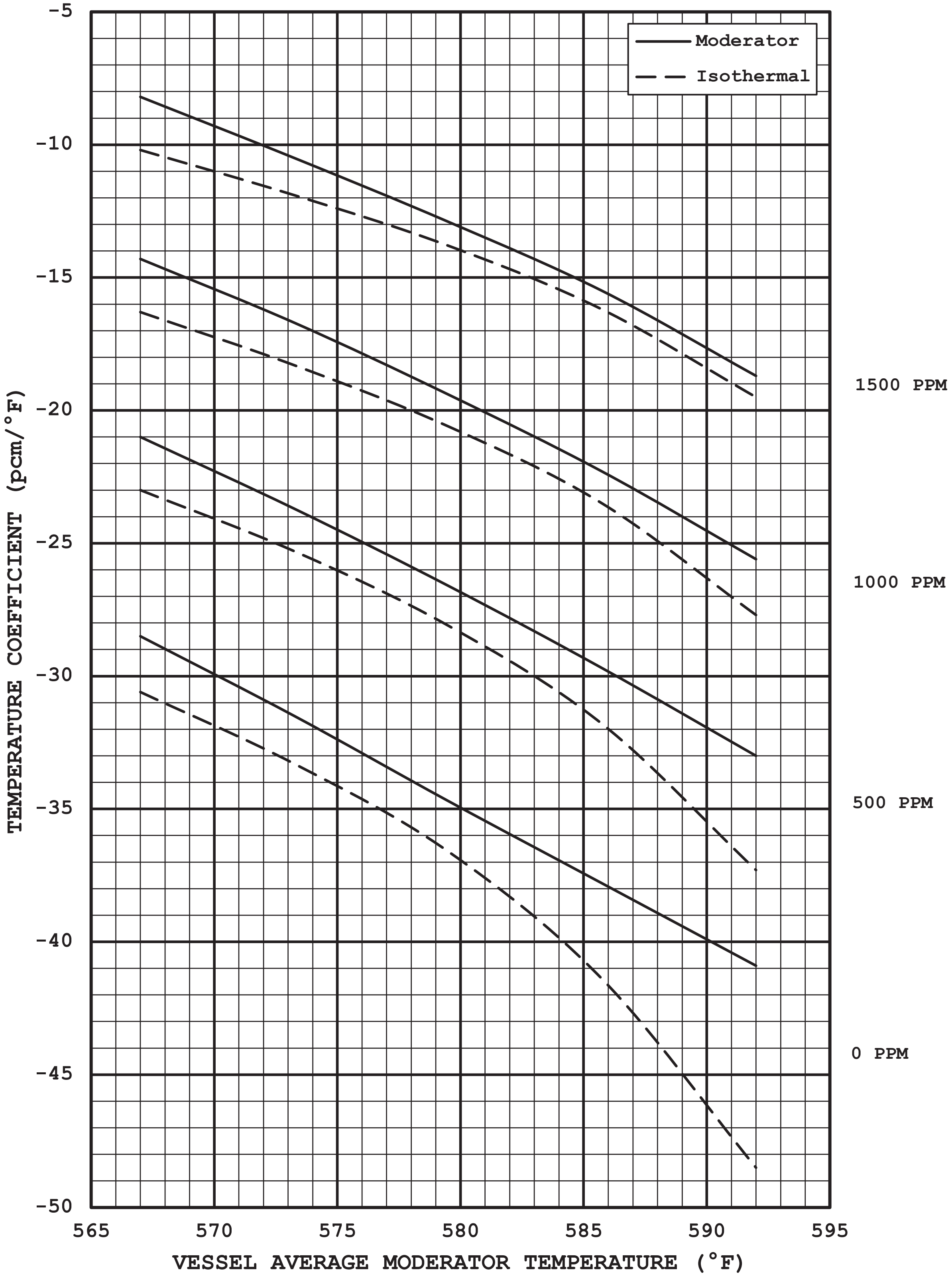


Table 5-4 Data for Figure 5-4

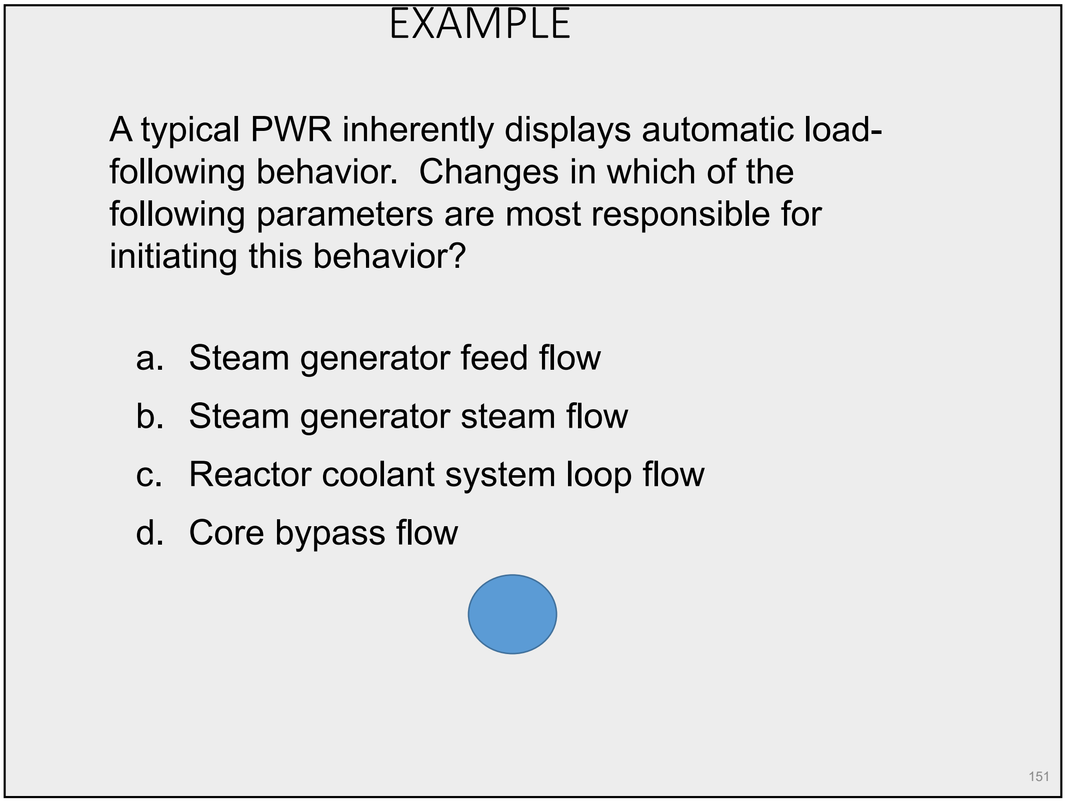
FOP Moderator and Isothermal Temperature Coefficients
as a function of Boron Concentration versus
Vessel Average Moderator Temperature at EOL (19139 MWD/MTU)

Moderator Temperature Coefficients (pcm/°F)

Boron Concentration (ppm)	Vessel Average Moderator Temperature (°F) 567.0	573.3	579.5	585.8	592.0
1500	-8.2	-10.5	-12.9	-15.5	-18.7
1000	-14.3	-16.7	-19.4	-22.3	-25.6
500	-21.0	-23.7	-26.6	-29.7	-33.0
0	-28.5	-31.5	-34.7	-37.8	-40.9

Isothermal Temperature Coefficients (pcm/°F)

Boron Concentration (ppm)	Vessel Average Moderator Temperature (°F) 567.0	573.3	579.5	585.8	592.0
1500	-10.2	-11.9	-13.8	-16.2	-19.5
1000	-16.3	-18.3	-20.6	-23.5	-27.7
500	-23.0	-25.3	-28.1	-31.8	-37.3
0	-30.6	-33.3	-36.6	-41.4	-48.5



EXAMPLE

A typical PWR inherently displays automatic load-following behavior. Changes in which of the following parameters are most responsible for initiating this behavior?

a. Steam generator feed flow
b. Steam generator steam flow
c. Reactor coolant system loop flow
d. Core bypass flow

B Changing steam flow results in changing reactor power. Increasing steam flow causes primary system temperature to decrease, resulting in core power increase. Decreasing steam flow causes primary system temperature to increase, resulting in core power decrease. Both effects are due to moderator temperature coefficient effects.

Exam Bank No.: 1668 **Last used on an NRC exam:** 2015

RO Sequence Number: 39

Unit 1 is operating at 100% power.

- SGFPT #13 is to be removed from service for pump maintenance.
- The crew has decided to run the Startup Feedpump (S/U SGFP) during this time to remain at 100% power.

In accordance with 0POP02-FW-0001, Main Feedwater, what additional action should be taken to account for SGFPT # 13 being removed from service?

A. A third FW Booster Pump must be started to reduce the load on the remaining two SGFPTs.

B. A third FW Booster Pump must be started to ensure S/U SGFP flow matches the flow from the secured SGFPT.

C. The SGFP Master Speed Controller must be placed in Manual to help keep the remaining two SGFPTs below 5400 rpm.

D. The SGFP Master Speed Controller must be placed in Manual to match each of the remaining SGFPT flows with the S/U SGFP flow.

Answer: A A third FW Booster Pump must be started to reduce the load on the remaining two SGFPTs.

Exam Bank No.: 1668 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 059 A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves

RO Importance: 2.7 **Tier:** **Group/Category:** **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 202.13 **Objective Number:** 5833

DISCUSS the following elements of 0POP02-FW-0001: 1. Purpose and Scope 2. Precautions and Limitations

Reference: 0POP02-FW-0001, Precaution 4.10

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: The third FW booster pump helps to ensure adequate suction pressure for all the feed pumps. By raising SGFP suction pressure the SGFPT speed does not require as much an increase to compensate for the lower capacity of the S/U SGFP. When using the S/U SGFP to replace a SGFPT at 100% power, then a third FW Booster pump will be required to reduce the load on the remaining two SGFPTs.

B: INCORRECT: When using the S/U SGFP to replace a SGFPT at 100% power, then a third FW Booster pump will be required to reduce the load on the remaining two SGFPS.

C: INCORRECT: If placing Main Feedwater flow on the S/U SGFP from a running SGFPT then it MAY be necessary to take manual control of the SGFP MASTER SPEED CONTROLLER OR Main or Low Power Feedwater Regulating Valves to maintain stable SG Water Levels; at the discretion of the Unit Supervisor or Shift Manager.

D: INCORRECT: If placing Main Feedwater flow on the S/U SGFP from a running SGFPT then it MAY be necessary to take manual control of the SGFP MASTER SPEED CONTROLLER OR Main or Low Power Feedwater Regulating Valves to maintain stable SG Water Levels; at the discretion of the Unit Supervisor or Shift Manager.

Question Level: F **Question Difficulty** 3

Justification:

Student must recall a precaution from the procedure to determine the correct procedural action.

4.0 Notes and Precautions

4.1 Extreme care SHOULD be used when pressurizing any portion of feedwater piping. To avoid potential for waterhammer, pressurize piping slowly by throttling valves. (Reference 2.6.2)

4.2 The Deareator (at normal operating temperature) SHALL **NOT** be used as a source of feedwater unless RCS temperature is GREATER THAN 340°F. This limitation is to ensure hot feedwater is **NOT** admitted to a relatively cold SG causing waterhammer.

4.3 The lineups of this procedure SHOULD only be performed when Main Feedwater System is shutdown. Operators MAY verify lineups during plant operations but SHALL **NOT** manipulate any valves or switches without Shift Manager/Unit Supervisor permission.

4.4 Sections and Subsections of this procedure MAY be performed independently at the direction of the Shift Manager/Unit Supervisor.

4.5 IF placing Main Feedwater flow on the Startup SGFP from a running SGFPT OR placing Main Feedwater flow on a SGFPT from the Startup SGFP, THEN it MAY be necessary to take MANUAL control of the "SGFP MASTER SPEED CONTROLLER" OR Main or Low Power Feedwater Regulating Valves to maintain stable SG Water Levels; at the discretion of the Shift Manager or Unit Supervisor

4.6 IF the critical bearing metal temperature of 190°F is reached on a Startup SGFP or FW Booster Pump, THEN corrective action SHALL be taken to cool the bearing. (Reference 2.15)

4.7 IF the critical bearing metal temperature of 200°F is reached on a Startup SGFP or FW Booster Pump, THEN the applicable pump SHALL be shutdown. (Reference 2.15)

4.8 A third FW Booster Pump MAY be operated to improve plant performance.

4.9 IF adjusting FW Booster Pump Lube Oil Pressure on a running FW Booster Pump, THEN as a precautionary measure, to evaluate starting the standby FW Booster Pump. (Reference 2.30)

4.10 WHEN using the S/U SGFP to replace a SGFP at 100% power, THEN a third FW Booster Pump will be required to reduce the load on the remaining two SGFPS.

4.11 The following are the Startup SGFP starting duties:

4.11.1 IF Motor Operating temperature is GREATER THAN or EQUAL to 160°F, THEN a 20 minute wait is required if motor runs between starts.

4.11.2 A 45 minute wait is required if the motor trips between starts (e.g. it starts but trips for some reason).

Exam Bank No.: 2677 **Last used on an NRC exam:** 2018

RO Sequence Number: 40

The Unit is at 100% power when the following occur at the same time:

- An inadvertent Safety Injection signal.
- QDPS APC A2 loses power.

Based on this occurrence,

the Train 'A' AFW Reg Valve will be __(1)__.

AND

the Train 'A' AFW Flow Indicator, FI-7525, will indicate __(2)__.

A. (1) throttled
(2) no flow

B. (1) throttled
(2) actual flow

C. (1) full open
(2) no flow

D. (1) full open
(2) actual flow

Answer: C (1) full open
(2) no flow

Exam Bank No.: 2677 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 061 K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners

RO Importance: 2.5 **Tier:** **Group/Category:** **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 202.44 **Objective Number:** 97939

Given a change in plant or system conditon, EXPLAIN the operation and indications of the QDPS System.

Reference: LOT 202.44 Powerpoint Presentation slide 61, 0POP04-AM-0001 Add. 13

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because a failure of different parts of QDPS affect different systems in different ways. The student has to know what systems are affected by QDPS and how a specific failure affects the system. The second part is correct.

B: INCORRECT: Plausible because a failure of different parts of QDPS affect different systems in different ways. The student has to know what systems are affected by QDPS and how a specific failure affects the system.

C: CORRECT: QDPS APC A2 being de-energized will cause the AFW flow control valve for that train to fail as is and cause the flow indication to read zero gpm. At 100% power the AFW Reg valves are aligned full open.

D: INCORRECT: Plausible because a failure of different parts of QDPS affect different systems in different ways. The student has to know what systems are affected by QDPS and how a specific failure affects the system.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to analyze the given condition to determine the effect on the AFW system.

Loss of QDPS APC-A2, may result in loss of the following control functions:

- Steam Generator 1A(2A) PORV, PV-7411 (Manual control available using **Addendum 3**)
- AFW Pump 11(21) Flow Controller, FT-7525
- Rx Head Vent Valve, HCV-0601

The following controllers may be affected if selected for control:

- Feedwater Control Valve, FCV-0551
- Feedwater Control Valve, FCV-0552
- Feedwater Control Valve, FCV-0553
- Feedwater Control Valve, FCV-0554
- Feedwater Bypass Control Valve, FV-7151
- Feedwater Bypass Control Valve, FV-7152
- Feedwater Bypass Control Valve, FV-7153
- Feedwater Bypass Control Valve, FV-7154

The following annunciators are unreliable:

•	Lampbox 2M03	Window A-7	"ESSEN CHLR 12A(22A) CNDSR PRESS HI/LO"
•	Lampbox 5M02	Window A-5	"OT DT RX PRETRP"
•	Lampbox 5M02	Window A-6	"OP DT RX PRETRP"
•	Lampbox 5M02	Window A-8	"QDPS ALARM TAS/QDPS FAIL"
•	Lampbox 5M02	Window C-6	"T AVG/AUCT T AVG DEV"
•	Lampbox 5M02	Window D-6	"DT/AUCT DT DEV"
•	Lampbox 5M02	Window F-5	"T AVG LO-LO NORM STM DUMP BLK ALERT"
•	Lampbox 5M03	Window A-4	"DELTA T ROD WTHDRWL BLK ALERT"
•	Lampbox 6M03	Window A-2	"SG 1A(2A) LVL HI-HI ALERT"
•	Lampbox 6M03	Window A-6	"SG 1B(2B) LVL HI-HI ALERT"
•	Lampbox 6M03	Window A-8	"QDPS ALARM SGWLCS"
•	Lampbox 6M03	Window D-2	"SG 1A(2A) LVL LO-LOALERT"
•	Lampbox 6M03	Window D-6	"SG 1B(2B) LVL LO-LO ALERT"`
•	Lampbox 6M04	Window A-2	"SG 1C(2C) LVL HI-HI ALERT"
•	Lampbox 6M04	Window A-6	"SG 1D(2D) LVL HI-HI ALERT"

Addendum 13	Affected Equipment And Unreliable Indications On Loss Of QDPS APC-A2	Addendum 13 Page 3 of 6

The following indicators are unreliable:

- AFW Loop A Flow, FI-7525
- AFW Loop A Flow, FI-7525A (ASP)
- SG A NR Compensated Level, LI-0571
- SG B NR Compensated Level, LI-0572
- SG C NR Compensated Level, LI-0573
- SG D NR Compensated Level, LI-0574
- RCS Loop A Hot Leg Temperature, TI-0410A
- RCS Loop A Cold Leg Temperature, TI-0410B
- RCS Loop A Delta T, TI-0411
- RCS Loop A Tavg, TI-0412A
- RCS Loop A OPDT Setpoint, TI-0412B
- RCS Loop A OTDT Setpoint, TI-0412C

The following recorders are unreliable:

- AFW Loop A Flow, FR-7525
- RCS OVERPWR/OVER TEMP/DELTA T, TR-0412

Addendum 13	Affected Equipment And Unreliable Indications On Loss Of QDPS APC-A2	Addendum 13 Page 4 of 6

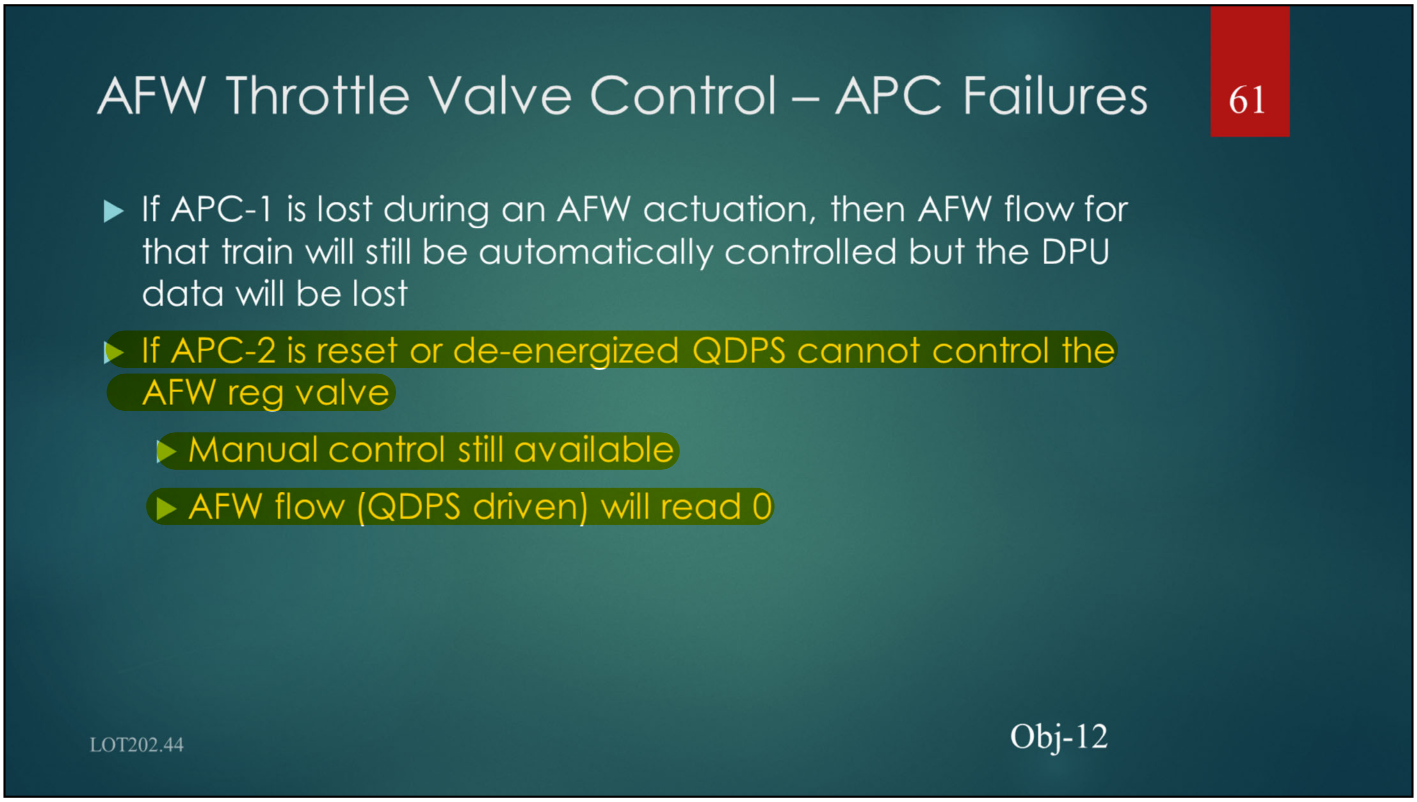
The following QDPS indications are unreliable:

- SG A PORV Pressure Loop, PV-7411
- SG A PORV Position Loop, ZT-7411
- AFW Loop A Flow, FT-7525
- SG A NR Level, LT-0571
- SG A NR Reference Temp, TE-0571
- SG A NR Reference Temp, TE-0571A
- SG A NR Compensated Level, LMY-0571A
- SG B NR Level, LT-0572
- SG B NR Reference Temp, TE-0572
- SG B NR Reference Temp, TE-0572A
- SG B NR Compensated Level, LMY-0572A
- SG C NR Level, LT-0573
- SG C NR Reference Temp, TE-0573
- SG C NR Reference Temp, TE-0573A
- SG C NR Compensated Level, LMY-0573A
- SG D NR Level, LT-0574
- SG D NR Reference Temp, TE-0574
- SG D NR Reference Temp, TE-0574A
- SG D NR Compensated Level, LMY-0574A
- RCS Loop A NR Hot Leg Temperature, TE-0410X
- RCS Loop A NR Hot Leg Temperature, TE-0410Y
- RCS Loop A NR Hot Leg Temperature, TE-0410Z
- RCS Loop A Hot Leg Average Temperature, TY-0410G
- Reactor Coolant Loop 1 Narrow Range Hot Leg Temperature Biases, TE-0410X/Y/Z

Addendum 13	Affected Equipment And Unreliable Indications On Loss Of QDPS APC-A2	Addendum 13 Page 5 of 6

The following computer points are unreliable:

- SG A Steam Line PORV Press, MSPE7411
- SG A Steam Line PORV Position, MSZE7411
- SG A Steam Line PORV Transfer Switch, MSHE7411
- SG A Steam Line PORV Valve Output, MSPE7411B
- SG A Steam Line PORV Setpoint, MSPE7411A
- SG A Steam Line PORV Auto-Man, MSPE7411C
- AFW Loop A Flow, AFFE7525
- AFW Loop A Flow Control Lo Limit, AFFE7525A
- AFW Loop A Flow Control Hi Limit, AFFE7525B
- Ess Chiller 12A(22A) Condenser Pressure, EWPE6904
- Ess Chiller 12A(22A) Compressor Running Signal, EWPB6904A
- Reactor Head Vent Valve Position, RCZE0601
- Reactor Head Vent Valve Transfer Switch, RCHE0601
- Reactor Head Vent Valve Output, RCHE0601A
- SG A NR Level, FWLE0571
- SG A NR Reference Temp, FWTE0571
- SG A NR Reference Temp, FWTE0571A
- SG A NR Compensated Level, FWLE0571A & L0403
- SG B NR Level, FWLE0572
- SG B NR Reference Temp, FWTE0572
- SG B NR Reference Temp, FWTE0572A
- SG B NR Compensated Level, FWLE0572A & LO423
- SG C NR Level, FWLE0573
- SG C NR Reference Temp, FWTE0573
- SG C NR Reference Temp, FWTE0573A
- SG C NR Compensated Level, FWLE0573A & L0443
- SG D NR Level, FWLE0574
- SG D NR Reference Temp, FWTE0574



AFW Throttle Valve Control – APC Failures

- If APC-1 is lost during an AFW actuation, then AFW flow for that train will still be automatically controlled but the DPU data will be lost
- If APC-2 is reset or de-energized QDPS cannot control the AFW reg valve
  - Manual control still available
  - AFW flow (QDPS driven) will read 0

LOT202.44

Obj-12

AFW flow will be zero on the board meters and the QDPS screens.

Exam Bank No.: 3233 **Last used on an NRC exam:** Never

RO Sequence Number: 41

AFW Pump #12 is powered from…

A. 480V Load Center E1B1

B. 480V Load Center E1C1

C. 4160V Bus E1B

D. 4160V Bus E1C

Answer: C 4160V Bus E1B

Exam Bank No.: 3233 **Source:** New **Modified from**

K/A Catalog Number: 062 K2.01 Knowledge of bus power supplies to the following: Major system loads

RO Importance: 3.3 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.36 **Objective Number:** 92396

LIST the major loads associated with the ESF Electrical System

Reference: LOT 202,28 Handout 1, page 7 of 28

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as other safety related pumps (RHR) are powered from a 480V LC and this is a "B" train load center.

B: INCORRECT: Plausible as other safety related pumps (RHR) are powered from a 480V LC and some safety related "B" train pumps have "C" train power supplies. (SFPCC pumps)

C: CORRECT: AFWP #12 is powered from 4160V Bus E1B.

D: INCORRECT: Plausible as this is also a 4160V Bus power supply and some safety related "B" train pumps have "C" train power supplies. (SFPCC pumps)

Question Level: F **Question Difficulty** 3

Justification:

The student must recall the power supply for AFWP #12.L

The motor-driven AFW pumps are maintained in standby readiness to take suction on the AFWST and discharge to their associated steam generator. See the section entitled Auxiliary Feedwater Pumps for pump information.

Trains A, B, and C use a motor to drive the pump. These motors are horizontally mounted induction type powered from their train's respective 4.16 kV ESF BUS.

Unit 1	Unit 2
#11 – E1A	#21 – E2A
#12 – E1B	#22 – E2B
#13 – E1C	#23 – E2C

The pumps can be operated from the Control Room at CP006 or at the Auxiliary Shutdown Panel (ASP) as selected at the transfer panels. When the Control Room is selected for control, the motor-driven pumps can receive an automatic strip signal from the sequencer.

These pumps automatically start on the following signals:

- Low-Low Level in Any Steam Generator
- AMSAC Actuation
- Safety Injection (Mode l)
- Loss of Offsite Power (LOOP) (Mode 2)
- Simultaneous LOOP with Safety Injection (Mode 3)

There are NO automatic trips or starts available when selected to the ASP. The transfer switches are on Panels 653 (#11 or #21) / 654 (#12 or #22) / 655 (#13 or #23) in the respective ESF Electrical Equipment Room and have positions of CR and ASP.

The control switches on CP006 have positions of PTL/STOP/AUTO/START and the switch positions on the ASP are STOP/NORMAL/START. The AFW pumps cannot be controlled after an automatic actuation unless SI and/or the sequencer (for LOOP) have been reset. Of course, PTL (pull-to-lock) could be used if necessary. In addition, the AFW Low-Low Level Block pushbuttons on CP006 allow block of the Steam Generator Low-Low Level auto start signal.

Closure of the pump motor's circuit breaker starts its associated IVC vent fan. The room thermostat setting controls cubicle temperature to ensure OPERABILITY is maintained. The cubicle thermostat is set and should not be changed without proper authorization.

Exam Bank No.: 3232 **Last used on an NRC exam:** Never

RO Sequence Number: 42

Unit 1 is at 50% power.

- A ground occurs on 125VDC Bus E1A11 and it de-energizes.

An impact of this failure is the unavailability of ______(1)_____.

The crew should respond using _____(2)_____.

A. (1) TDAFWP #14
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection

B. (1) TDAFWP #14
(2) 0POP04-DJ-0001, Loss of Class 1E 125 VDC Power

C. (1) Pressurizer PORV 655A
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection

D. (1) Pressurizer PORV 655A
(2) 0POP04-DJ-0001, Loss of Class 1E 125 VDC Power

Answer: D (1) Pressurizer PORV 655A (2) 0POP04-DJ-0001, Loss of Class 1E 125 VDC Power

Exam Bank No.: 3232 **Source:** New **Modified from**

K/A Catalog Number: 063 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Grounds

RO Importance: 2.5 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(5)

STP Lesson: LOT 201.37 **Objective Number:** 92986

STATE the local and MCR instrumentation available to monitor the Class 1E 125 VDC System.
92047 - STATE how the Class 1E 125 VDC System interfaces with other systems.

Reference: 0POP04-DJ-0001, Addendum 8, Page 1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as "A" and "D" DC power is similar in nature and both have power supplies from "A" train and the TDAFWP 14 comes from "D" train power. The loss of E1A11 does cause operational problems, but not does not result in a reactor trip.

B: INCORRECT: Plausible as "A" and "D" DC power is similar in nature and both have power supplies from "A" train and the TDAFWP 14 comes from "D" train power. 0POP04-DJ-0001 is the correct procedure.

C: INCORRECT: With a loss of Class 1E 125VDC Bus E1A11, the "A" PORV 655A will lose power. The loss of E1A11 does cause operational problems, but not does not result in a reactor trip.

D: CORRECT: With a loss of Class 1E 125VDC Bus E1A11, the "A" PORV 655A will lose power. The loss of E1A11 does cause operational problems, but not does not result in a reactor trip thus 0POP04-DJ-0001 is the correct procedure.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the given conditions and determine the impact on the plant and the appropriate AOP with which to respond.

125V DC Bus E1A11(E2A11)			
CUBICLE NUMBER	DESCRIPTION	REFERENCE DRAWINGS	FAILURE MODE
1B	"125V BATT E1A11(E2A11) TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
2A	"BATT CHGR E1A11(E2A11)-2 TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
3A	"BATT CHGR E1A11(E2A11)-1 TO 125V DC SWBD E1A11(E2A11)"	9-E-DJAA-01	N/A
3B	"TO 125V DC DIST PNL PL039A EAB 10' EL"	9-E-DJAA-01 9-E-DJAE-01	LOSS OF POWER TO PL039A
3C	"STBY DIESEL GEN 11(21) CONT PNL ZLP101"	9-E-DJAA-01 9-E-DG01-01	LOSS OF FLASHING POWER TO DIESEL
4A	"TRAIN A RX TRIP SWGR CONTROL PWR"	9-E-DJAA-01 0386-00189 9-E-SP18-03	DISABLES RTR 1B & BYR 2B SHUNT TRIP CKTS – REDUNDANT RTS 1C OR BYS 2C. NO IMMEDIATE EFFECT
4B	"TO 125V DC DIST PNL PL139A STBY D/G 11(21) RM 35' EL"	9-E-DJAA-01 9-E-DJAF-01	LOSS OF POWER TO DG CONTROL PANELS
4D	"4.16KV BUS E1A(E2A) DC CONTROL PWR"	9-E-DJAA-01 9-E-PK04-02	LOSS OF CONTROL PWR TO BKRs ON 4.16KV ESF BUS E1A(E2A)
5B	"480V ESF LC E1A(E2A) DC CONTROL PWR"	9-E-DJAA-01 9-E-PL01-01	LOSS OF CONTROL POWER TO BREAKERS ON 480V LC E1A(E2A)
5C	"ESF LOAD SEQUENCER CABINET A-ZLP801"	9-E-DJAA-01 4176-01020	DISABLES SEQUENCER
5D	"CH I INST/CONT PWR TMI INVERTER DC PWR"	9-E-DJAA-01 9-E-VAAA-01	ALTERNATE SOURCE SUPPLIES POWER
5E	"CH I INST/CONT PWR NSSS INVERTER DC PWR"	9-E-DJAA-01 9-E-VAAA-01	ALTERNATE SOURCE SUPPLIES POWER
7B	"B/U BKR-PRZR PORV 0655A 1(2)-RC-PCV-0655A"	9-E-DJAA-01	DISABLES PORV (TRAIN B REDUNDANT)
7C	"PRZR PORV 0655A 1(2)-RC-PCV-0655A"	9-E-DJAA-01 9-E-RC13-01	DISABLES PORV (TRAIN B REDUNDANT)

Exam Bank No.: 3213 **Last used on an NRC exam:** Never

RO Sequence Number: 43

A Reactor Trip and Safety Injection occurred in Unit 1.

- Due to an oil leak, the Shift Manager directs the crew to trip ESF DG 13.
- The RO reports that ESF DG 13 can NOT be tripped from the Control Room.

If ESF DG 13 must be tripped by manually isolating fuel, then the operator should _____(1)____ on the Fuel Rack Lever and _____(2)_____.

A. (1) pull down
(2) immediately release to the normal position

B. (1) pull down
(2) hold until the ESF DG 13 shaft stops rotating

C. (1) push up
(2) immediately release to the normal position

D. (1) push up
(2) hold until the ESF DG 13 shaft stops rotating

Answer: B (1) pull down (2) hold until the ESF DG 13 shaft stops rotating

Exam Bank No.: 3213 **Source:** New **Modified from**

K/A Catalog Number: 064 K4.08 Knowledge of the ED/G system design features(s) and/or interlock(s) which provide for the following: ED/G fuel isolation valves

RO Importance: 2.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.39 **Objective Number:** 44266

STATE the function of the Emergency Diesel Generator and major components (include location anf operation of components, power supplies, major flowpaths, interfaces, starting and control circuit, and interlocks)

Reference: 0POP02-DG-0001, Step 4.16

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as the lever is pulled down and with confusion about the the fuel rack mechanism.

B: CORRECT: The lever must be pulled down and held down to avoid re-admitting fuel to the cylinders and risking a restart.

C: INCORRECT: Plausible with confusion over lever operations and the fuel rack mechanism.

D: INCORRECT: Plausible with confusion over lever operations. The lever is held in position until the shaft stops rotating.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall information about manual trip features.

4.16 IF it is necessary to immediately stop a DG while operating in the Emergency Mode, THEN any of the following methods may be utilized:

- The "EMERGENCY STOP CIRCUIT 1" and "EMERGENCY STOP CIRCUIT 2" STOP pushbuttons **SHALL** be depressed. (ZLP102)
- The "EMER STOP" plunger **SHALL** be placed in the "PULL TO STOP" position. (CP003) (Reference Step 4.8)
- IF both of the above identified methods fail to stop the DG, THEN at least one of the following methods **SHALL** be utilized to stop the DG:
    - The fuel racks **SHALL** be manually closed by pulling down on the fuel rack lever until the DG shaft stops rotating. (Southwest Side of Engine Above Turning Gear)
    - An overspeed trip **SHALL** be manually initiated at the Engine Overspeed Governor. (Top of Diesel on Generator End)

4.17 IF the load current from ESF Bus E1A(E2A) to either Standby Bus 1F(2F) or Emergency Bus 1L(2L) exceeds 1008 Amperes (excessive bus reverse power) during parallel operation, THEN the DG output breaker will trip open.

4.18 IF Class 1E DC control power is lost at any time the DG is operating, THEN the DG will shutdown.

4.19 IF Non-Class 1E DC control power is lost at any time while the DG is operating in the Emergency Mode, THEN the DG will shutdown when released from the Emergency Mode. The DG shutdown will occur even if the Non-Class 1E DC power has been restored.

4.20 IF the DG is operating in the Emergency Mode AND one of the following non-emergency mode engine trips (**NOT** test simulations) is received, THEN the DG will trip when released from the Emergency Mode: (ZLP102)

- "HIGH TEMP MAIN & CONN. ROD OR GEN. BRG.", Window A-3
- "EXCESSIVE VIBRATION ENG. & GEN. BRG.", Window A-4
- "FAILURE TURBO THRUST BEARING", Window A-5
- "JACKET WATER HIGH TEMPERATURE", Window A-6

Exam Bank No.: 1488 **Last used on an NRC exam:** 2019

RO Sequence Number: 44

If RT-8032, GWPS Outlet, reaches the HIGH alarm setpoint then…

A. WG-FV-4657, GWPS Inlet Header Valve, closes.

B. WG-FV-4671, GWPS Discharge Flow Valve, goes to recirculation mode.

C. GWPS Bellows Compressor, receives a start signal.

D. WG-FIC-4653, Volume Control Tank Flow, closes.

Answer: A WG-FV-4657, GWPS INLET HEADER VALVE, closes

Exam Bank No.: 1488 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 073 K4.01 Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint

RO Importance: 4.0 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(11)

STP Lesson: LOT 202.41 **Objective Number:** 92122

LIST the initiating condition and resultant automatic action for the PERMS radiation monitors associated with the following systems:
B. Gaseous Waste Processing System

Reference: LOT202.41, handout #1, page 26, PPT slide 117

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

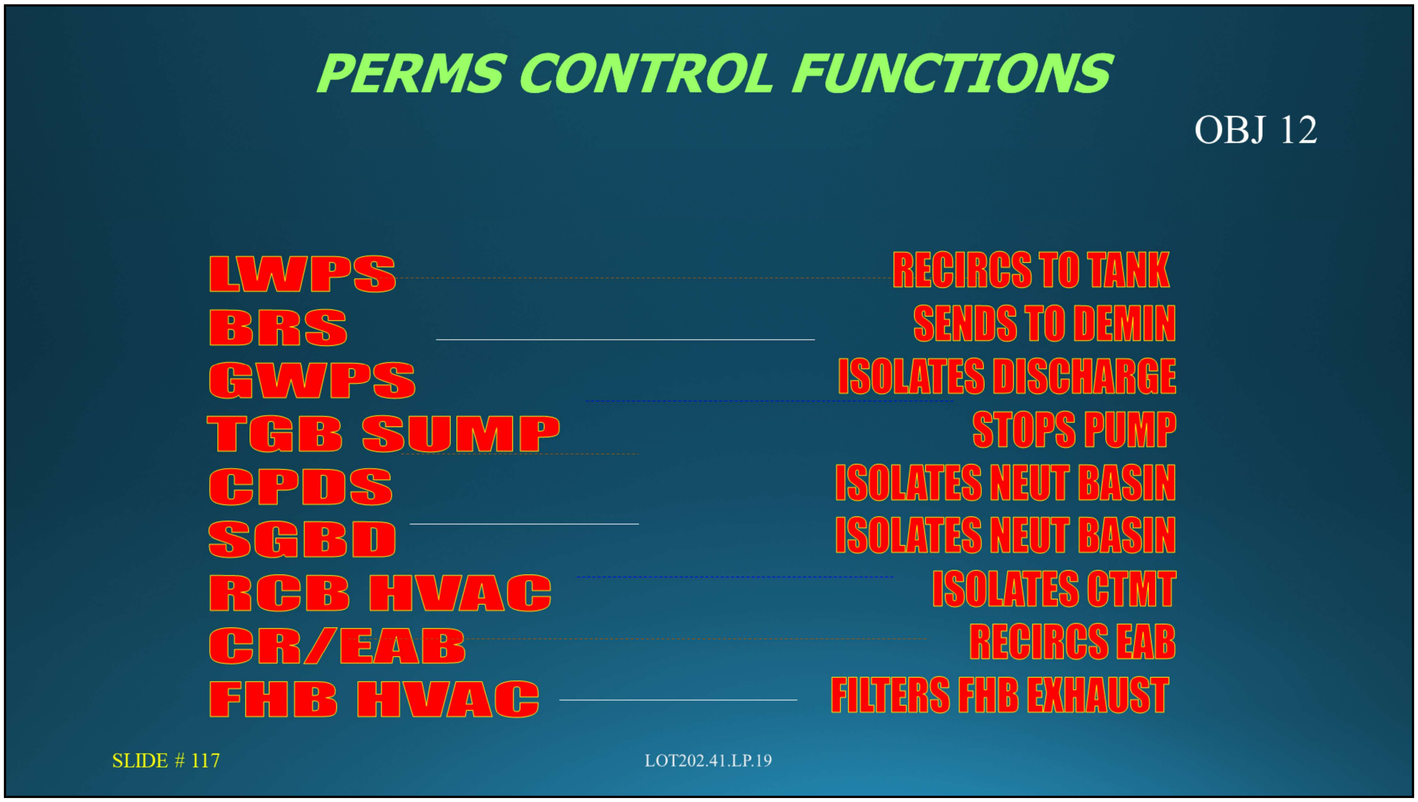
Distractor Justification

A: CORRECT: One of the auto actions for this monitor is to close the inlet valve.
B: INCORRECT: Plausible as some other radiation releases by placing the discharge valve in recirc on a high rad signal..
C: INCORRECT: Plausible as the bellows compressor is affected but it trips instead of starts.
D: INCORRECT: Plausible as the VCT is a significant input to the system, but individual valves do not close.

Question Level: F **Question Difficulty** 2

Justification:

The applicant must have a knowledge of the protection afforded by the PRMs to the waste gas system.



PERMS CONTROL FUNCTIONS

OBJ 12

LWPS	RECIRCS TO TANK
BRS	SENDS TO DEMIN
GWPS	ISOLATES DISCHARGE
TGB SUMP	STOPS PUMP
CPDS	ISOLATES NEUT BASIN
SGBD	ISOLATES NEUT BASIN
RCB HVAC	ISOLATES CTMT
CR/EAB	RECIRCS EAB
FHB HVAC	FILTERS FHB EXHAUST

SLIDE # 117

LOT202.41.LP.19

Liquid Waste Processing System (LWPS) – RT-8038 In the event of high radiation in this system, or a monitor failure condition a diversion valve will send the liquid effluent from the system back to the waste monitor tanks

Boron Recycle System – RT-8037 Monitor serves to divert flow back to the BRS Evaporator Feed Demineralizers on a high radiation signal or monitor failure signal. BR-RCV-4204

Gaseous Waste Processing System (GWPS)- RT-8032 sends a signal to the GWPS shutdown circuitry to close the discharge valve, the inlet valve , the BRS vent and secure the Bellows Compressor.

Turbine Generator Building Sump and Drain System – RT-8041 High radiation at the sump pump discharge or a monitor failure condition will stop the sump pump.

Condensate Polishing System (CPS) – RT-8042 High radiation at the discharge of the CPS to the neutralization basin or a monitor failure condition will close this discharge valve. CP-FV-5804

Steam Generator Blowdown (SGBD) System RT-8043 High radiation in the steam generator blowdown liquid or a monitor failure condition closes the SGBD discharge to the neutralization basin. SB-FV-5019 closes.

Containment Building Ventilation System RT-8012 & 8013 -High radiation in the RCB Purge System Exhaust sends a signal to the Solid State Protection

System (SSPS) for Containment Ventilation Isolation (CVI). **EAB and Control Room Envelope (HVAC) – RT-8033 & 8034** High radiation level at the EAB air intake initiates Control Room/EAB emergency ventilation.

Fuel Handling Building HVAC System –RT-8035 & 8036 High radiation at the exhaust initiates FHB exhaust filtration.

Turbine Generator Building (TGB) Sump 1 monitor

Failed Fuel Monitor (CVCS)

Condensate Polishing System Monitor

Gaseous Waste Processing System (GWPS) Inlet Monitor

MAB HVAC (7)

GWPS Discharge Monitor

Main Steam Line Monitors (4) (Class 1E)

SG Blowdown Monitors (4) (Class 1E)

PERMS CONTROL FUNCTIONS

The PERMS monitor control functions are also outlined in Table 1 for those monitors possessing control functions. Some examples of typical control functions from these monitors can be shown as follows:

Liquid Waste Processing System (LWPS) – RT-8038 In the event of high radiation in this system, or a monitor failure condition a diversion valve will send the liquid effluent from the system back to the waste monitor tanks. WL-FV-4077

Boron Recycle System – RT-8037 Monitor serves to divert flow back to the BRS Evaporator Feed Demineralizers on a high radiation signal or monitor failure signal. BR-RCV-4204

Gaseous Waste Processing System (GWPS)- RT-8032 High radiation as measured at the GWPS discharge or a monitor failure condition results in the shutdown of the GWPS. The High Rad or Monitor Failure sends a signal to the GWPS shutdown circuitry to close the discharge valve, the inlet valve , the BRS vent and secure the Bellows Compressor.

Turbine Generator Building Sump and Drain System – RT-8041 High radiation at the sump pump discharge or a monitor failure condition will stop the sump pump.

Condensate Polishing System (CPS) – RT-8042 High radiation at the discharge of the CPS to the neutralization basin or a monitor failure condition will close this discharge valve. CP-FV-5804

Steam Generator Blowdown (SGBD) System RT-8043 High radiation in the steam generator blowdown liquid or a monitor failure condition closes the SGBD discharge to the neutralization basin. SB-FV-5019 closes.

Containment Building Ventilation System RT-8012 & 8013 -High radiation in the RCB Purge System Exhaust sends a signal to the Solid State Protection System (SSPS) for Containment Ventilation Isolation (CVI). (Normal and supplementary purge)

Electrical Auxiliary Building and Control Room Envelope (HVAC) – RT-8033 & 8034 High radiation level at the EAB air intake initiates Control Room/EAB emergency ventilation.

Exam Bank No.: 3159 **Last used on an NRC exam:** Never

RO Sequence Number: 45

Unit 1 at 100% power.

Which of the following would require entry into a one hour or less Technical Specification LCO action? (Assume no allowance for the CRMP)

A. Essential Cooling Water intake temperature is 97ºF.

B. Essential Cooling Pond Level is at 26.0' Mean Sea Level.

C. ECW Pump "C" is out of service and ECW Pump 'B' is declared inoperable.

D. Standby Diesel Generator 11 has been out of service for 4 hours and 'A' ECW Pump is declared inoperable.

Answer: C ECW Pump "C" is out of service and ECW Pump "B" is declared inoperable.

Exam Bank No.: 3159 **Source:** New **Modified from** N/A

K/A Catalog Number: 076 G2.2.39 Knowledge of less than or equal to one-hour Technical Specification action statements for systems: Service Water

RO Importance: 3.9 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 503.01 **Objective Number:** 80056

DETERMINE the applicable Technical Specification and/or the Technical Requirements Manual (TRM) Limiting Conditions for Operation (LCOs) and the required actions to be taken.

Reference: Technical Specification 3.7.4, 3.7.5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because Technical Specification 3.7.5 has a maximum temperature but the temperature given in the distractor still meets the LCO.

B: INCORRECT. Plasuible because pond level is part of Technical Specification 3.7.5 but the minimum water level for the Essential Cooling Pond is 25.5 ft.

C: CORRECT. Per Tech Spec 3.7.4 with two inoperable trains of ECW you are in a one hour action to restore at least two to operable or be in hot standby in 6 hrs.

D: INCORRECT. Plausible because student may think that two ECW pumps are inoperable with misunderstanding of power sources and that the diesel being out makes two ECW pumps inoperable but it does not. The action in 3.7.4 with one pump inoperable is 7 days.

Question Level: H **Question Difficulty** 3

Justification:

Student must analyze condition given in the selections to determine if entry is required into a one hour or less TS Action.

3/4.7.4 ESSENTIAL COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least three independent essential cooling water loops shall be OPERABLE.

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

ACTION:

a. With only two essential cooling water loops OPERABLE, within 7 days restore at least three loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more essential cooling water loops inoperable, within 1 hour restore at least two loops to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least three essential cooling water loops shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position;

b. At a frequency in accordance with the Surveillance Frequency Control Program during shutdown, by verifying that:

   1) Each Essential Cooling Water automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal, and

   2) Each Essential Cooling Water pump starts automatically on an actual or simulated signal.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

a. A minimum water level at or above elevation 25.5 feet Mean Sea Level, USGS datum, and

b. An Essential Cooling Water intake temperature of less than or equal to 99°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION is applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by verifying the intake water temperature and water level to be within their limits.

Exam Bank No.: 2586 **Last used on an NRC exam:** 2018

RO Sequence Number: 46

Unit 1 is at 100% power.

The Reactor Operator can monitor Instrument and Service Air header pressures on _____(1)____ control board indicators and ICS.

The normal operating pressure is approximately _____(2)_____.

A. (1) CP-008
(2) 110 psig

B. (1) CP-002
(2) 110 psig

C. (1) CP-008
(2) 125 psig

D. (1) CP-002
(2) 125 psig

Answer: C (1) CP-008 - (2) 125 psig

Exam Bank No.: 2586 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 078 A4.01 Ability to manually operate and/or monitor in the control room: Pressure gauges

RO Importance: 3.1 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(4)

STP Lesson: LOT 202.26 **Objective Number:** 80556

Describe the instrumentation and controls available to monitor and operate the Instrument Air and Service Air systems.

Reference: LOT202.26 PPT slide 24; 0POP04-IA-0001, Page 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because 110 psig is above the pressure for parts of IA/SA to start isolating. Incorrect because at this pressure all of the IA/SA Compressors would be running and loaded.

B: INCORRECT: Plausible because IA OCIV is operated on CP-002 and this is the only control for IA/SA on the control panels. Incorrect because the IA/SA header pressure indication is on CP-008. Plausible because 110 psig is above the pressure for parts of IA/SA to start isolating. Incorrect because at this pressure all of the IA/SA Compressors would be running and loaded.

C: CORRECT: IA Header and SA Header Indications can be found on CP-008 and the normal operating pressure for both is about 120 to 125 psig.

D: INCORRECT: Plausible because IA OCIV is operated on CP-002 and this is the only control for IA/SA on the control panels. Incorrect because the IA/SA header pressure indication is on CP-008.

Question Level: F **Question Difficulty** 2

Justification:

The student must have fundamental knowledge of location of control room indication and IA/SA System pressures.

MONITORING

monitor plant parameters at a frequency based on importance & plant conditions.

CP008

OBJ 7

PURPOSE

This procedure provides the necessary operator actions for responding to a significant degradation or loss of Instrument Air (IA) capacity.

Instrument Air Pressure (Decreasing)	**Automatic Actuation**
122 psig	IA Compressor 11(21) Starts/Loads in Local Control
119 psig	IA Compressor 12(22) Starts/Loads in Local Control
116 psig	IA Compressor 13(23) Starts/Loads in Local Control
113 psig	IA Compressor 14(24) (air cooled and BOP DG powered) Starts/Loads
100 psig	Service Air Isolation Valve N1(2)IA-PV-9785 Closes
90 psig	Instrument Air to Yard Valve N1(2)IA-PV-8568 Closes
80 psig	Instrument Air Dryer Bypass N1(2)IA-PV-9983 Opens

SYMPTOMS OR ENTRY CONDITIONS

1. The following Control Room annunciator alarms:
   - “SAS ISOL VLV CLOSE” Lampbox 08M3, Window F-3
   - “SAS HDR PRESS LO” Lampbox 08M3, Window E-3
   - “IAS HDR PRESS LO” Lampbox 08M3, Window D-3
2. All operable IA compressors running continuously.
3. No IA compressors running.
4. Various air operated valves observed to be drifting to failure positions.

This Procedure is Applicable in All Modes

Exam Bank No.: 2685 **Last used on an NRC exam:** 2017

RO Sequence Number: 47

Unit 1 is in Mode 6.

- Containment closure is required to be established.

The Equipment Hatch must be closed with a MINIMUM of _____(1)_____ bolts.

If the Equipment Hatch can NOT be closed as required, movement of irradiated fuel in the containment must be suspended _____(2)_____.

A. (1) 4
(2) immediately

B. (1) 14
(2) immediately

C. (1) 4
(2) within 2 hours

D. (1) 14
(2) within 2 hours

Answer: A (1) 4
(2) immediately

Exam Bank No.: 2685 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 103 K3.03 Knowledge of the effect that a loss or malfunction of the containment system will have on the following:
Loss of containment integrity under refueling operations.

RO Importance: 3.7 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(9)

STP Lesson: LOT 503.01 **Objective Number:** 92102

Given the topic or title of a specification included in the Technical Specifications, or the Technical Requirements Manual (TRM), DESCRIBE the general requirements of the specification to include components or administrative requirements affected, limitations, major time frames involved, major surveillance in order to comply, and the bases for the specification.

Reference: CR 16-12260 Lessons Learned, TRM 3.9.4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: The TRM states that a minimum of 4 bolts must be in place to secure the equipment hatch during refueling. If it is NOT closed with 4 bolts or capable of being closed with 4 bolts within 2 hours, then suspend movement of irradiated fuel in containment immediately.

B: INCORRECT: Plausible because 14 bolts would be 50% of the total equipment hatch bolts (28).

C: INCORRECT: Plausible because the equipment hatch has to have the manpower and equipment staged to be able to close and bolt the equipment hatch within 2 hours. Incorrect because if at anytime it is determined that this capability does not exist then the action is to suspend movement of irradiated fuel in the containment immediately.

D: INCORRECT: Plausible because 14 bolts would be 50% of the total equipment hatch bolts (28). Plausible because the equipment hatch has to have the manpower and equipment staged to be able to close and bolt the equipment hatch within 2 hours. Incorrect because if at anytime it is determined that this capability does not exist then the action is to suspend movement of irradiated fuel in the containment immediately.

Question Level: F **Question Difficulty** 2

Justification:

The student must have knowledge of what constitutes a loss of containment integrity during refueling operations. Knowing the number of bolts required for the equipment hatch during refueling would determine if the containment system had a loss or malfunction. The effect of the loss would be to immediately suspend movent of irradiated fuel in the containment which would be a requirement of less than 1 hour and therefore RO knowledge.

Station Clock Reset

Oct. 10, 2016

Background

On Sunday, October 9, the Unit 2 Reactor Containment Building Equipment Hatch was inappropriately reconfigured while in a mode not allowed by technical specifications. The investigation is ongoing, but it appears we gave permission to "hot bolt" the containment equipment hatch prior to Mode 5, the intended time to unbolt the hatch. Hot bolting involves loosening 24 of 28 bolts on the equipment hatch. The hot bolting was done in Mode 3 and is not allowed in Modes 1-4. The scheduled activity was for after the unit entered Mode 5.
Prior to Mode 5 entry, containment integrity is required to be maintained.

This event resulted in a Station Human Performance Clock Reset and a level 2 Plant Status Control event for a human performance error leading to an unplanned entry into a limiting condition of operations of <72 hours.

Key Takeaways

Nuclear Professionals have high ownership for the preparation and safe execution of assigned work activities. We consider the most likely undesired consequences of our activities and validate contingency actions. We actively participate in briefings and are focused and engaged in the tasks we execute.

Procedure use

The procedure for unbolting the equipment hatch specifically states to ENSURE the unit is in Mode 5 or 6 before performing work. Strict adherence to approved procedures and written instructions is required. Approved written instructions SHALL be followed as written and users SHALL NOT deviate from or omit steps except where specifically allowed by the written instruction being used or the allowances specified in this procedure. If a step cannot be performed as written, we must stop and contact our supervisor to resolve the issue.

Questioning Attitude

During this event, the workers communicated with the One Stop Shop and there was not a clear understanding of the procedure requirements and when the work was to be performed. We need to use a questioning attitude when we are working an activity to fully understand the predecessors and any mode restrictions prior to starting the activity. We must ask questions to eliminate assumptions and ensure parties involved in the communications fully understand what and when the activity is taking place.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITIONS FOR OPERATIONS

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment hatch closed and held in place by a minimum of four bolts
OR
1) The Reactor has been subcritical for ≥ 42 hours, AND
If open, the equipment hatch is capable of being closed within 2 hours.

b. A minimum of one door in the containment Auxiliary Airlock (AAL) and a minimum of one door in the containment Personnel Airlock (PAL) are closed.
OR
The water level is ≥ 23 feet above the reactor vessel flange.
AND
The Reactor has been subcritical for ≥ 42 hours
AND
Individuals are available to close a PAL door and AAL door when directed as soon as possible but within 2 hours.

c. All other penetrations providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed as soon as possible but within 2 hours.

APPLICABILITY: During movement of irradiated fuel within the containment.

ACTION:

If the requirements above are not satisfied, immediately suspend all operations involving movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its required condition or capable of being closed as required in Requirement 3.9.4 at least once per 7 days during movement of irradiated fuel in the containment building by (as applicable):

a. Verifying the penetrations are in their required condition or capable of being placed in their required condition.

b. Staging proper tools and designating trained personnel to close the equipment hatch if open.

Exam Bank No.: 2824

Last used on an NRC exam: 2019

RO Sequence Number: 48

If a relief lifts on an RCP Number 1 Seal Leakoff line inside containment, then level in the ____(1)____ would rise. This relief valve lifts at a setpoint of _____(2)_____ psig.

A. (1) RCDT
(2) 600

B. (1) RCDT
(2) 150

C. (1) PRT
(2) 600

D. (1) PRT
(2) 150

Answer: D (1) PRT (2) 150

Exam Bank No.: 2824 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 007 A3.01 Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT.

RO Importance: 2.7 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.04 **Objective Number:** 32335

List all the reliefs which discharge into the PRT.

Reference: LOT201.06 Student Handout, page 28 of 50

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as this tank is inside containment and does receive valve leakoffs and 600 psig is the setting for the RHR relief valves.

B: INCORRECT: Plausible as this tank is inside containment and does receive valve leakoffs . This is the correct setpoint.

C: INCORRECT: Plausible as this is the correct destination and 600 psig is the setting for the RHR relief valves.

D: CORRECT: This relief is routed to the PRT and lifts at 150 psig.

Question Level: F **Question Difficulty** 3

Justification:

The student must know the inputs to the PRT and the setpoints of the relief valve in question.

The Seal Water Leakoff Header Relief Valve is located inside the containment. It is set to relieve to the pressurizer relief tank (PRT) at a pressure of 150 psig. Seal water return header containment isolation valves (MOV-077 and MOV-079) close on a Phase A isolation signal and can be operated from CP-004.

The Seal Water Return Filter is a 25 micron disposable filter and has a maximum designed flow of 250 gpm. The filter is replaced if the pressure across it increases to 20 psid or if it has a radiation level of 5 R/Hour on contact. A manual bypass valve is provided to allow continued operation during filter replacement.

The Seal Water Heat Exchanger (see Figure 12) receives flow from the RCP #1 seal leakoff (~12 gpm), the excess letdown heat exchanger (~20 gpm) and the centrifugal charging pumps recirculation flow (~60 gpm). The seal water heat exchanger is cooled by component cooling water. A relief valve on the inlet to the heat exchanger is set to relieve to the VCT at a pressure of 150 psig. The outlet of the seal water heat exchanger is normally directed to the suction of the charging pumps, but can also be directed back to the VCT through a spray nozzle. The seal water return flow would be directed to the VCT if it became necessary to maintain the hydrogen concentration of the RCS when the normal letdown path is not available. A manual bypass valve and line is provided around the heat exchanger for use during maintenance or leak conditions.

Excess Letdown System (Figure 11)

The Excess Letdown System is used if the normal letdown path is inoperable, to maintain the flow balance between the letdown and charging systems or for additional letdown when necessary. Excess letdown is taken from the reactor coolant loop 4 intermediate leg upstream of the RCP and flows to the excess letdown heat exchanger.

Excess Letdown Isolation Valves (MOV-082 and MOV-083) are motor operated and controlled from CP-004 by "CLOSE, NORMAL, OPEN" spring return to "NORMAL" switches. Valve position status lamps are located above the switch. These valves fail "as is" on loss of electrical power. On a SI or Phase A signal these valves must be closed immediately by the operator to prevent damage to the CCW side of the Excess Letdown Heat Exchanger and loss of reactor coolant through the Seal Water Leakoff Header Relief Valve since there are no automatic closure signals.

The Excess Letdown Heat Exchanger is a stainless steel, tube and shell heat exchanger, cooled by Component Cooling System. It reduces the letdown water temperature to approximately 160°F. A high temperature alarm sound on CP-004 if the temperature rises to 175°F. Letdown flows through the tube side and component cooling water flows through the shell side. Excess letdown heat exchanger outlet pressure and temperature can be read on CP-004. The excess letdown heat exchanger is located inside containment in the northeast section of the 52 foot elevation.

The Excess Letdown Flow Control Valve (HCV-227) is used to control letdown flow to a maximum of approximately 20 gpm. HCV-227 is a motor operated valve and fails as is. Caution should be observed when establishing excess letdown flow because of the effect it can have on #1 seal water leakoff backpressure. The seal return flow shares a common line. Increased backpressure could cause a change in the #1 seal water flow.

Exam Bank No.: 3161 **Last used on an NRC exam:** Never

RO Sequence Number: 49

A large break LOCA has occurred in Unit 1.

- Containment pressure is 15 psig.
- Containment Spray has not actuated.

Containment Spray can be manually actuated from ___(1)___. To get a manual actuation ___(2)___ CS/CIB/CVI switches must be placed in ACTUATE.

A. (1) CP-002 OR CP-005
(2) 1 of 2

B. (1) CP-002 OR CP-005
(2) 2 of 2

C. (1) CP-002 ONLY
(2) 1 of 2

D. (1) CP-002 ONLY
(2) 2 of 2

Answer: B (1) CP-002 OR CP-005
(2) 2 of 2

Exam Bank No.: 3161 **Source:** New **Modified from** N/A

K/A Catalog Number: 026 A4.01 Ability to manually operate and/or monitor in the control room: CSS controls

RO Importance: 4.5 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.11 **Objective Number:** 81164

Describe the instrumentation and controls available to operate and monitor the Containment Spray System.

Reference: LOT 201.11 PPT slide 35

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because SI, Steam Line Isolation, and CIA/CVI only require one switch to manually actuate. Part 1 is correct.

B: CORRECT. Containment Spray can be manually actuated from CP-002 or CP-005 using 2 of 2 CS/CIB/CVI switches simultaneously at either panel. There are four total manual switches.

C: INCORRECT. Plausible because the actuation can be reset on only CP-002 but can be manually actuated from either panel. Also the manual controls for the CS Pumps and Discharge valves are also located on CP-002. Part 2 is plausible because SI, Steam Line Isolation, and CIA/CVI only requires one switch to manually actuate.

D: INCORRECT. Plausible because the Containment Spray actuation can only be reset on CP-002 but can be manually actuated from either panel. Also the manual controls for the CS Pumps and Discharge valves are located on CP-002. Part 2 is correct.

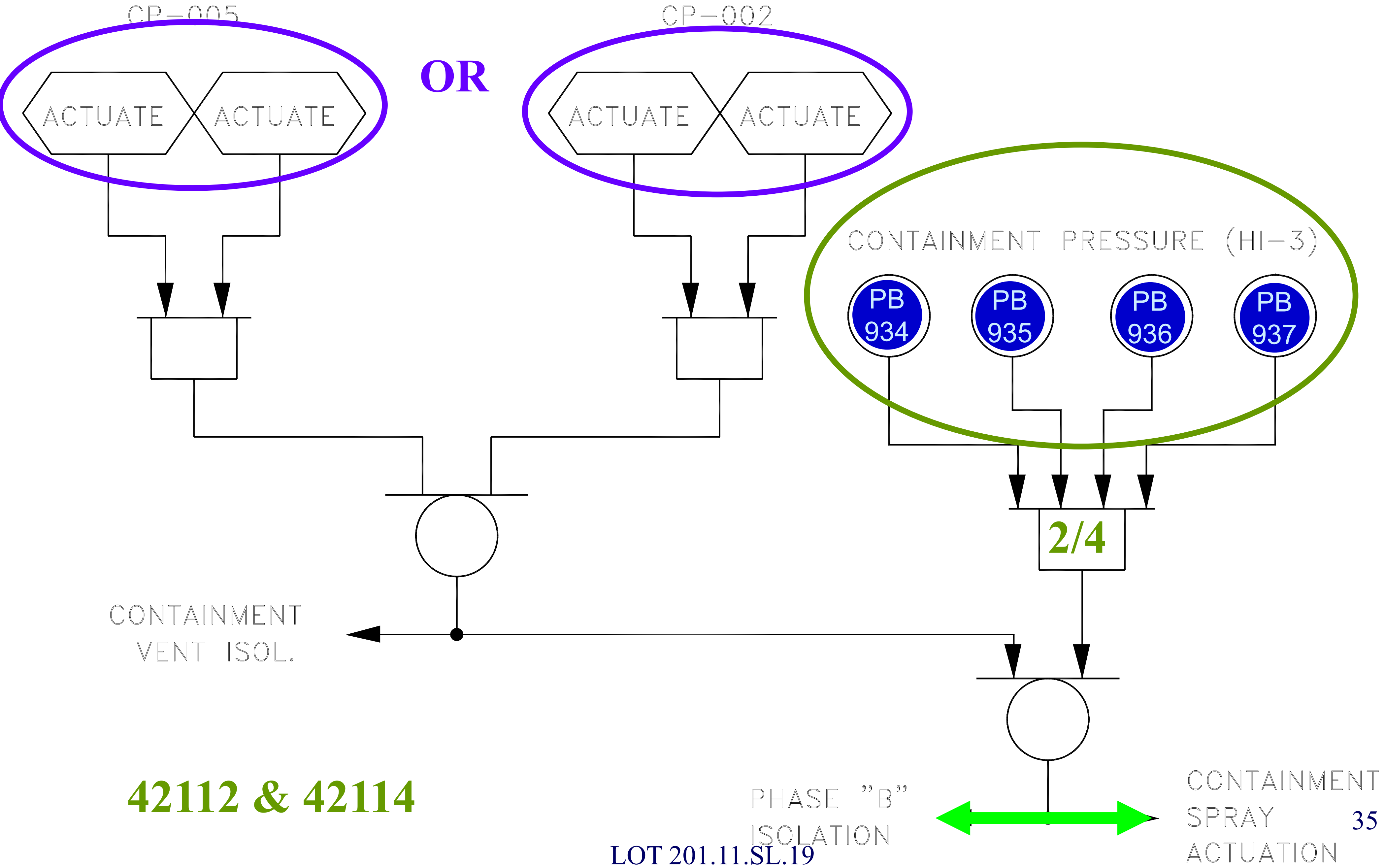
Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of method for manually actuating Containment Spray and the location of controls.

OBJ # 7 & 9

CONTAINMENT SPRAY ACTUATION



42112 & 42114

Exam Bank No.: 3231 **Last used on an NRC exam:** Never

RO Sequence Number: 50

_____(1)_____ supplies the steam for TDAFWP #14 and _____(2)_____ supplies the steam for the SGFPTs.

A. (1) Main Steam
(2) Extraction Steam

B. (1) Main Steam
(2) Main Steam and/or Reheat Steam

C. (1) Reheat Steam
(2) Extraction Steam

D. (1) Reheat Steam
(2) Main Steam and/or Reheat Steam

Answer: B (1) Main Steam; (2) Main Steam and/or Reheat Steam

Exam Bank No.: 3231 **Source:** New **Modified from** N/A

K/A Catalog Number: 039 G2.1.27 Knowledge of system purpose and/or function. Main and Reheat Steam

RO Importance: 3.9 **Tier:** **Group/Category:** **10CFR Reference:** 55.41(b)()

STP Lesson: LOT 201.01 **Objective Number:** 29949

STATE the functions of the Main Steam System.

Reference: LOT 202.02.PPT SL 19 & 20, LOT202.14.HO.1, page 6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as the TDAFWP is supplied from MS, and with a misconception between steam sources and where lines tap off.

B: CORRECT: The TDAFWP is supplied from the "D" MS Header upstream of the MSIV. The SGFPTs are supplied from Main Steam or Reheat Steam depending on the operation of the pump.

C: INCORRECT. Plausible with a misconception of the steam supply source for the TDAFWP and SGFPTs

D: INCORRECT. Plausible with a misconception of the steam supply source for the TDAFWP. This is the correct supply for the SGFPTs.

Question Level: F **Question Difficulty** 2

Justification:

Student must recall functions of the Main Steam system.

TURBINE STEAM SUPPLY AND EXHAUST

A flow diagram of the steam supply to and the steam exhaust from Number 11 Steam Generator Feedwater Pump Turbine is show on Slide 19. Main Steam (high pressure steam) is used during a plant startup to approximately 40% plant load.

At about 40% plant load, the Hot Reheat Steam (low pressure steam) pressure is high enough to start supplying steam to the turbine.

The steam supply drains prevent condensate accumulation that could cause turbine damage or water hammer as steam flow is admitted into the lines or the turbine.

Turbine exhaust valve must be open (ZSO-7467 indicates valve is fully open) before the "MAN SPD CONTROL" signal in the Woodward 509 Digital Control System can be increased.

LOW PRESSURE STEAM NON-RETURN VALVE

The function of the non-return valve is to prevent back flow of steam to the low pressure turbines following a main turbine trip. When the main turbine trips, the low pressure steam to the feed pump turbine will decrease to condenser vacuum. As the low pressure steam pressure decreases, the high pressure governor valve will start to open admitting steam to the feed pump turbine first stage to maintain pump speed. Since the feed pump turbine low pressure governor and stop valves are open, steam could flow from the feed pump turbine first stage to the main turbine low pressure turbines if it were not for the non-return valve.

Slide 20 is a logic diagram showing the operation of the low pressure steam non-return valve for Number 13 Steam Generator Feedwater Pump turbine. The solenoid valve is energized to vent air from the operating cylinder. A spring above the operating piston pushes the operating piston down when air is vented from the bottom of the operating cylinder. The air operator cannot close the non-return valve against flow. It gives an assist to insure valve closure as back flow is about to start.

The local test valve, when actuated, supplies instrument air on top of the operating cylinder. With air pressure on the top and bottom of the piston, the spring on the top of the piston will move the piston downward. A small amount of rotational movement in the non-return valve shaft should be observed as the operating piston moves downward. This indicates that the non-return valve is not stuck in the open position and is free to close.

Auxiliary relay panel ERR101 is located on the 29' elevation of the TGB in the southeast corner by Battery Charger Room 105. Distribution panel PL125F is located on the 55' elevation of the TGB near the enclosure room around ZLP-609. ZLP-609 contains the SGFPT Woodward 509 Digital Control Systems.

Distribution panel PL125F receives power from cubicle 3E on 125 VDC Distribution Switchboard 2A which is located in Battery Charger Room 105 which, in turn, is located in the southeast corner of the TGB on the 29' elevation.

STEAM SUPPLY DRAIN VALVES

OBJ 01

Functions

- MS conducts steam from the four S/Gs through the containment wall to the following:
  - Main Turbine
  - Moisture Separator Reheaters (MSR)
  - Steam Dumps (Turbine Bypass System)
  - Steam Generator Feed Pump Turbines (SGFPT)

OBJ 01

Functions (Continued)

  - Gland Seal Steam System (GS)
  - Auxiliary Steam (AS)
  - Deaerator
  - AFW Pump #14
- Also provides a means to dissipate heat generated in primary system during plant startup, hot standby, cooldown, step load reduction, and off-normal conditions.

Exam Bank No.: 3230 **Last used on an NRC exam:** Never

RO Sequence Number: 51

Unit 1 is at 100% power.

250 VDC SWBD 1A loses power. This affects the ______(1)______ Breaker. The breaker trip function is _______(2)_______.

A. (1) 13.8 KV AUX BUS 1F Feeder
(2) disabled

B. (1) 13.8 KV AUX BUS 1F Feeder
(2) unaffected

C. (1) Main Generator Output
(2) disabled

D. (1) Main Generator Output
(2) unaffected

Answer: C (1) Main Generator Output; (2) disabled

Exam Bank No.: 3230 **Source:** New **Modified from** N/A

K/A Catalog Number: 062 K1.03 Knowledge of the physical connections and/or cause-effec
relationships between the ac distribution system
and the following systems: DC distribution

RO Importance: 3.5 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 202.17 **Objective Number:** 91615

Given a plant or system condition, PREDICT the operations of the Main Generator and Exciter System.

Reference: 0POP04-DC-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the 13.8 KV Feeder Breaker use DC Control Power but it is 125 VDC vice 250 VDC, The second part is correct.

B: INCORRECT. Plausible because the 13.8 KV Feeder Breakers use DC Control Power but it is 125 VDC vice 250 VDC and there are malfunctions that don't affect both the trip and close circuits (trip of 250 VDC SWGR breaker 3B).The student can easily reason that keeping the trip function available is more desirable than keeping the closure function available.

C: CORRECT. Per Addendum 1 of 0POP04-DC-0001 loss of 250 VDC SWGR 1A would disable trip functions of the Main Generator Output Breaker.

D: INCORRECT. Plausible because there are malfunctions that don't affect both the trip and close circuits (trip of 250 VDC SWGR breaker 3B). The student can easily reason that keeping the trip function available is more desirable than keeping the closure function available.

Question Level: H **Question Difficulty** 2

Justification:

Student must recall functions of control power and predict the effect on the breaker.

Addendum 1 | **250V DC Switchboard Loads** | Addendum 1 Page 1 of 1

DEVICE	COMPONENT NAME	AFFECTED EQUIPMENT
SWBD 1A(2A)/2C	SGFP 11(21) EMER DC LUBE OIL PUMP	Deenergizes SGFP 11(21) EMER DC LUBE OIL PUMP
SWBD 1A(2A)/2D	SGFP 12(22) EMER DC LUBE OIL PUMP	Deenergizes SGFP 12(22) EMER DC LUBE OIL PUMP
SWBD 1A(2A)/2E	SGFP 13(23) EMER DC LUBE OIL PUMP	Deenergizes SGFP 13(23) EMER DC LUBE OIL PUMP
SWBD 1A(2A)/2F	MAIN GEN CKT BKR CONT PNL DC POWER	Disables trip and close circuit of Main Generator Circuit Breaker
SWBD 1A(2A)/3B	MAIN GEN CKT BKR CONT PNL DC POWER	Disables trip circuit of Main Generator Circuit Breaker
SWBD 1A(2A)/3C	MAIN GEN AIR SIDE SEAL OIL B/U PUMP	Deenergizes MAIN GEN AIR SIDE SEAL OIL B/U PUMP
SWBD 1A(2A)/3D	MAIN TURB EMER DC OIL PUMP	Deenergizes MAIN TURB EMER DC OIL PUMP

Exam Bank No.: 2220 **Last used on an NRC exam:** 2013

RO Sequence Number: 52

Complete the following regarding the Emergency Diesel Generator Trip Solenoids:

Control power for the Class 1E solenoids comes from _____(1)_____

Control power for the Non-Class solenoid comes from _____(2)_____.

A. (1) Class 1E 120 Volt Vital AC
(2) Non-Class 125 Volt DC

B. (1) Class 1E 125 Volt DC
(2) Non-Class 125 Volt DC

C. (1) Class 1E 120 Volt Vital AC
(2) Non-Class 120 Volt Vital AC

D. (1) Class 1E 125 Volt DC
(2) Non-Class 120 Volt Vital AC

Answer: B (1) Class 1E 125 Volt DC
(2) Non-Class 125 Volt DC

Exam Bank No.: 2220 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 064 K2.03 Knowledge of bus power supplies to the following: Control power

RO Importance: 3.2 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.39 **Objective Number:** 44288

STATE the normal source of power for the Emergency Diesel Generator system, sub systems and components.

Reference: LOT 201.39 Powerpoint

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Credible because solenoids can be either AC or DC powered. The applicant must have knowledge of system design to correctly respond.

B: CORRECT: Emergency Diesel Generator Emergency Trip Solenoids are powered from Class 1E 125 VDC and the Non-Emergency Trip Solenoids are powered from Non-Class 125 VDC.

C: INCORRECT: Credible because solenoids can be either AC or DC powered. The applicant must have knowledge of system design to correctly respond.

D: INCORRECT: Credible because solenoids can be either AC or DC powered. The applicant must have knowledge of system design to correctly respond.

Question Level: F **Question Difficulty** 3

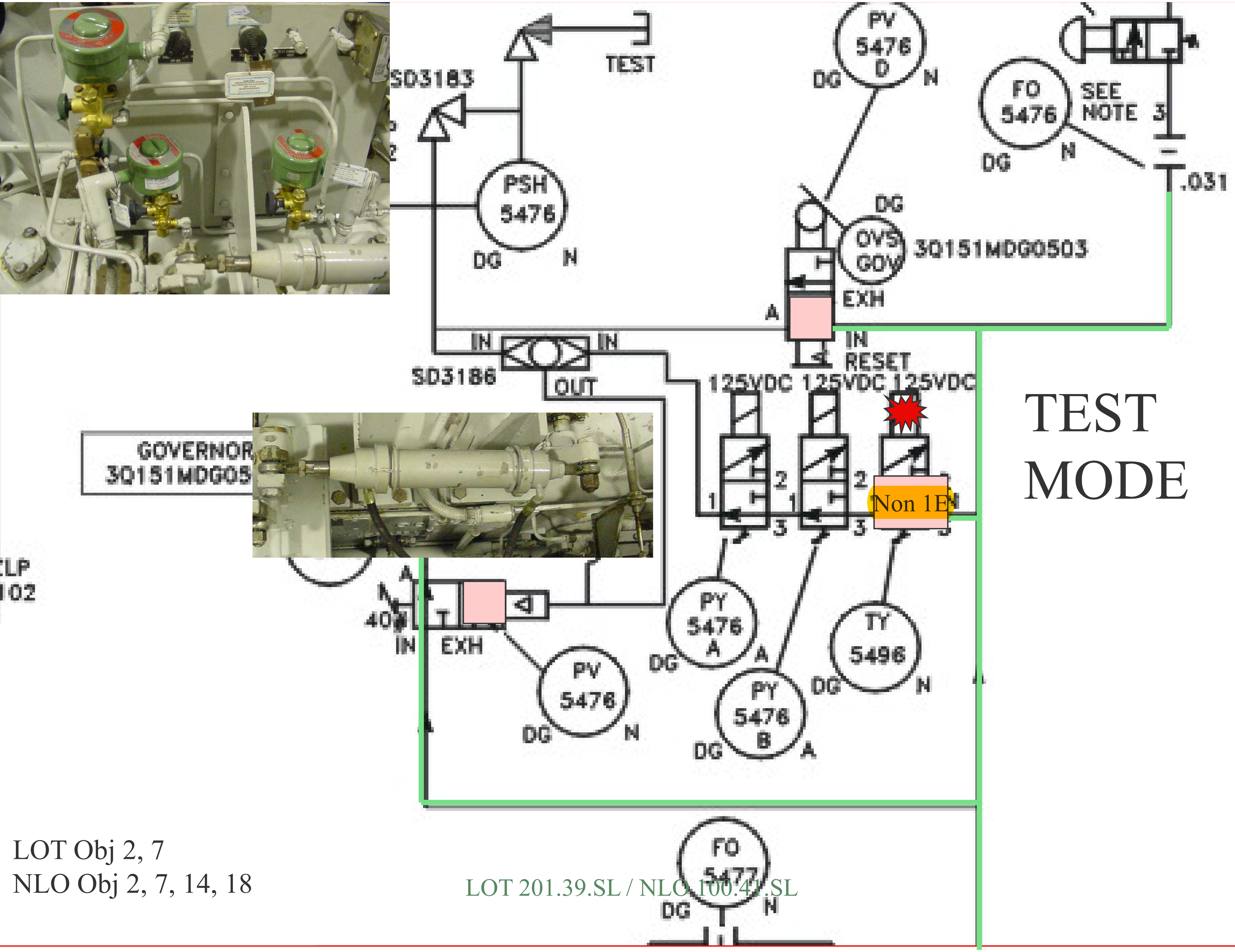
Justification:

The Reactor Operator must have knowledge of power supplies to different components of the Emergency Diesel Generator system.

Power Supply Listings

- 125 VDC Class 1E Distribution
  - Control power for emergency mode
  - Annunciator power
  - Field flashing power
- 125 VDC Non-class Distribution
  - Control power for test mode
  - Standby Fuel Oil Booster Pump
  - Panel indicating lights
  - Engine isolation relays
  - Engine mode reset power

LOT Obj 1, 9
NLO Obj 1, 12



TEST MODE

LOT Obj 2, 7
NLO Obj 2, 7, 14, 18

Exam Bank No.: 3180 **Last used on an NRC exam:** Never

RO Sequence Number: 53

Unit 1 is at 100% power.

- CCW Pump 1A is in service.
- ECW Pump 1A and 1B are in service, ECW Pump 1C selected for STANDBY.
- Essential Chillers 12A and 12B are in service.
- MAB Chillers 11A, 11B, and 11C are in service.
- Centrifugal Charging Pump 1B is running.
- RCFCs 11A, 11B, 12A, and 12B are running.

ECW Pump 1B trips.

Which of the following will occur?

A. MAB Chiller 11B will trip.

B. Essential Chiller 12B will trip.

C. Centrifugal Charging Pump 1B motor temperatures start to rise.

D. Reactor Containment Building temperature and pressure start to rise.

Answer: B Essential Chiller 12B will trip.

Exam Bank No.: 3180 **Source:** New **Modified from** N/A

K/A Catalog Number: 076 K3.07 Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads

RO Importance: 3.7 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.13 **Objective Number:** 30114

DISCUSS how the ECW system interfaces with the following plant systems:
A. Component Cooling Water
B. EAB HVAC Chillers
C. Diesel Generator

Reference: 0POP04-EW-0001 Addendum 1, 0POP09-AN-02M4-A-3, LOT202.36, slide 36

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because MAB Chillers trip on HI refrigerant pressure but condenser water flow is supplied by OL-ACW not ECW.

B: CORRECT. If ECW 1B trips the 12B Essential Chiller will lose condenser cooling and refrigerant pressure will rise to the trip setpoint and trip Essential Chiller 12B.

C: INCORRECT. Plausible as the ECW Train B cools a supplementary cooler but it is for the CCW pump. Centrifugal Charging pumps are cooled by CCW common header which is unaffected.INCORRECT. Plausible as RCFCs are cooled by chill water and the Essential Chiller B is affected however the RCFCs are cooled by RCB chill water. Without chill water to the Train B RCFCs the RCB would heat up and pressure would rise.

D: INCORRECT. Plausible as RCFCs are cooled by chill water and the Essential Chiller B is affected however the RCFCs are cooled by RCB chill water. Without chill water to the Train B RCFCs the RCB would heat up and pressure would rise.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to assess plant conditions and predict the impact on the ECW/CCW system.

Addendum 1	Components Cooled By Essential Cooling Water	Addendum 1 Page 1 of 1

Essential Cooling Water provides cooling water to safety-related equipment in the Mechanical Auxiliary Building and Diesel Generator Building.

- Component Cooling Water Heat Exchanger
- Essential Chillers
- ESF Diesel Generators
- Component Cooling Water Pump Supplementary Coolers

Essential Chiller Description

Chiller Trips

Chillers do NOT automatically restart following these trips:

- Compressor high discharge temperature - 220^{0}F
- Compressor high oil temperature - 180^{0}F
- Refrigerant low temperature (evaporator) - 32.5^{0}F
- Refrigerant high pressure (condenser) - 30 psig
- Compressor low oil pressure - 15 psid
- Loss of 120 VAC control power

Objective 5,6

ESSEN CHLR 12B(22B) CNDSR PRESS HI/LO

Automatic Actions: None

Immediate Actions: None

> NOTE
>
> - This alarm is not linked to any chiller automatic control functions. The chiller has its own protective safety devices that protect against unsafe operating conditions. Given the current cold weather chiller operating guidelines a condenser low pressure alarm is not expected to ever take place.
> - If this is a valid condition it is most likely that the chiller is near or in a stall or surge condition. Look for unusually loud chiller noise, possibly in a cyclic pattern. Chiller operation under this condition should not be sustained.
> - Any time an essential chiller is operated with HVAC loads bypassed or with very light loads, the condenser low pressure alarm may occur. The time spent operating a chiller in this condition should be minimized. Additionally the chiller may cycle off due to low load and low chilled water outlet temperature.

Subsequent Actions: _____ 1) DISPATCH an Operator to investigate cause of alarm as necessary.

_____ 2) IF parameters support operating the Essential Chiller per Cold Weather Guidelines, THEN PLACE Essential Chiller in cold weather alignment per 0POP02-CH-0005, Essential Chiller Operation, Essential Chiller Cold Weather Operations Section.

_____ 3) IF Essential Chiller compressor discharge pressure is normal AND noise level is normal, THEN generate a Condition Report to verify calibration of condenser pressure transmitter that supplies the annunciator.

ESSEN CHLR 12B(22B) CNDSR PRESS HI/LO

Subsequent Actions: (Continued)

______ 4) IF Essential Chiller compressor discharge pressure is high, THEN PERFORM the following:

______ a) VERIFY condenser pressure is within the operability guidelines listed for given ECW outlet temperature.

______ b) CHECK for excess purge light on chiller.

______ c) CHECK the purge counter trends for indication for inleakage.

______ d) GENERATE Condition Report for either condition above.

______ 5) SECURE the affected Essential Chiller per 0POP02-CH-0005, Essential Chiller Operation.

______ 6) START additional Essential Chillers as required per 0POP02-CH-0005, Essential Chiller Operation.

______ 7) NOTIFY the System Engineer of the alarming Essential Chiller.

______ 8) TAKE appropriate actions per Technical Specification 3.5.2, 3.6.2.1, 3.7.7 and 3.7.14.

Probable Causes:

1) ECW Temperature high
2) ECW Temperature low
3) ECW Flow high
4) ECW Flow low
5) Essential Chiller 12B(22B) high condenser pressure
6) Essential Chiller 12B(22B) low condenser pressure

Origin and Setpoint: 1) 1(2)-EW-PT-6905 Hi: 20 psig/34.7 psia Lo: 11.2 inHg/9.2 psia

References:

1) 0POP02-CH-0005, Essential Chiller Operation
2) Technical Specification 3.5.2, 3.6.2.1, 3.7.7 and 3.7.14.

3)

Exam Bank No.: 3215 **Last used on an NRC exam:** Never

RO Sequence Number: 54

The crew is unable to open FV-8565, IA Containment Isolation Valve, from the Control Room.

Complete the following:

Instrument Air may be manually supplied to Containment by ________(1)_______, FV-8565, IA Containment Isolation Valve.

This action is accomplished on the _______(2)_______.

A. (1) bypassing
(2) 60’ MAB

B. (1) bypassing
(2) 10’ MAB

C. (1) manually overriding
(2) 60’ MAB

D. (1) manually overriding
(2) 10’ MAB

Answer: D (1) manually overriding FV-8565, IA Containment Isolation Valve (2) 10' MAB

Exam Bank No.: 3215 **Source:** New **Modified from**

K/A Catalog Number: 078 K4.01 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control

RO Importance: 2.7 **Tier:** 2 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 202.26 **Objective Number:** 102100

Given a plant or system condition, PREDICT the operation of the Instrument and Service Air system.

Reference: 0POP02-IA-0001, Section 9.0

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as many valves do have bypass lines that may be opened upon valve failure and with confusion over the location. The 60' MAB also has Instrument Air provided to Containment Isolation Valves.

B: INCORRECT: Plausible as many valves do have bypass lines that may be opened upon valve failure. This is the correct location.

C: INCORRECT: Plausible as this is the correct method and with confusion over the location. The 60' MAB also has Instrument Air provided to Containment Isolation Valves.

D: CORRECT: This is the correct method and location.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall how to manually operate the IA Containment Isolation Valve

9.0 Manually Overriding IA OCIV Locally

CAUTION

- Manually overriding FV-8565, IA OCIV, to Containment **SHALL NOT** be performed during Modes 1 through 4.
- Containment Closure may be affected if Instrument Air is lost while FV-8565, IA OCIV, is manually overridden. (TRM 3.9.4)
- The time FV-8565, IA OCIV, is manually overridden SHALL be minimized.

9.1 OPEN FV-8565, IA OCIV, locally by performing the following:

9.1.1 ENSURE FV-8565, IA OCIV, is in the AUTO/OPEN position. (CP-002) ______

9.1.2 UNLOCK and OPEN 1(2)-IA-1516, RCB INSTRUMENT AIR HEADER FV-8565 MANUAL OVERRIDE VALVE. (10’ MAB Penetration) ______

9.1.3 UNLOCK and CLOSE 1(2)-IA-1515, RCB INSTRUMENT AIR HEADER FV-8565 MANUAL OVERRIDE VALVE. (10’ MAB Penetration) ______

9.1.4 INITIATE an OPERABILITY ASSESSMENT SYSTEM (OAS) entry for FV-8565 to ensure restoration prior to entry into Mode 4. ______

9.2 WHEN FV-8565, IA OCIV, is **NO** longer required to be bypassed open, THEN PERFORM the following:

NOTE

The following step will reset the seal-in relay if power was removed from FV-8565, IA OCIV. FV-8565 is powered from “B” train 125 vdc.

9.2.1 Momentarily PLACE FV-8565, IA OCIV, handswitch to the OPEN position to reset the valve circuitry. (CP-002)

_____ Perform

_____ Dual Verif

9.2.2 OPEN and LOCK 1(2)-IA-1515, RCB INSTRUMENT AIR HEADER FV-8565 MANUAL OVERRIDE VALVE. (10’ MAB Penetration)

_____ Perform

_____ Ind. Verif

9.2.3 CLOSE and LOCK 1(2)-IA-1516, RCB INSTRUMENT AIR HEADER FV-8565 MANUAL OVERRIDE VALVE. (10’ MAB Penetration)

_____ Perform

_____ Ind. Verif

Exam Bank No.: 3208 **Last used on an NRC exam:** Never

RO Sequence Number: 55

In accordance with 0POP03-ZG-0007, Plant Cooldown, subcooling shall be at least a MINIMUM of ___(1)___ ºF with RCS pressure between 1000 and 1900 psig during the cooldown.

To ensure the limits are met RCS pressure should be controlled using Pressurizer Heaters and ____(2)____ Spray.

A. (1) 50
(2) Normal

B. (1) 50
(2) Auxiliary

C. (1) 100
(2) Normal

D. (1) 100
(2) Auxiliary

Answer: A (1) 50; (2) Normal

Exam Bank No.: 3208 **Source:** New **Modified from** N/A

K/A Catalog Number: 002 A1.04 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCS controls including: Subcooling Margin

RO Importance: 3.9 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 506.01 **Objective Number:** 92158

In regards to the referenced procedure, DISCUSS the following: Purpose, Scope, Notes, and Precautions.

Reference: 0POP03-ZG-0007 Pgs. 30,45,165,166

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. 0POP03-ZG-0007 requires a minimum of 50 degrees F Subcooled Margin for the conditions given in the stem. Some combination of RCPs will be run during the RCS cooldown and RCS Pressure will be controlled manually using PZR heaters and Normal Spray.

B: INCORRECT. Plausible because Auxiliary Spray is used much later in the procedure in MODE 5 when RCPs are secured. The first part is correct.

C: INCORRECT. Plausible because 100 degrees F is the the maximum value allowed by the procedure for the given conditions. The second part is correct.

D: INCORRECT. Plausible because 100 degrees F is the the maximum value allowed by the procedure for the given conditions and Auxiliary Spray is used much later in the procedure in MODE 5 when RCPs are secured.

Question Level: H **Question Difficulty** 3

Justification:

Student must have knowledge of subcooled margin limits during a normal plant cooldown and analyze conditions given to select method to control it.

Initials

NOTE

- The RCP configuration recommended for RCS Cooldown conditions are as follows:
  - (Preferred) RCP 1D(2D) [RCP 1D(2D) is the only pump with adequate spray flow in single pump configuration]
  - (Alt 1) RCP 1D(2D) and RCP 1A(2A)
  - (Alt 2) RCP 1D(2D) and RCP 1B(2B) or RCP 1C(2C)
  - (Alt 3) RCP 1A(2A) and RCP 1B(2B) or RCP 1C(2C)
  - (Alt 4) RCP 1B(2B) and RCP 1C(2C)
- MAXIMIZING the Reactor Coolant System flowrate during CRUD Cleanup may reduce DOSE rates for the remainder of the Outage due to faster cleanup time.
- MAXIMIZING the Reactor Coolant System flowrate during cooldown may raise DOSE rates for the remainder of the Outage due to large CRUD burst.
- For Planned Outages, the RCP configuration recommended by the Outage Planning Team should be considered; the Shift Manager/Unit Supervisor has final selection judgment.
- For Un-Planned Outages, the RCP configuration recommended by Health Physics and Chemistry should be considered; the Shift Manager/Unit Supervisor has final selection judgment.
- IF **NO** RCP configuration recommendation is provided, THEN the default RCP configuration is RCP 1D(2D) running and 3 RCPs secured, the Shift Manager/Unit Supervisor has final selection judgment.
- The Shift Manager/Unit Supervisor may authorize changes to the operating RCP configuration during the cooldown if conditions warrant (i.e. large CRUD burst, pump problems, cleanup issues, etc.).

5.15 ESTABLISH the operating RCP(s) configuration for the cooldown, by PERFORMING the following:

5.15.1 IF Planned Outage, THEN CONTACT Outage Planning Team for RCP configuration for cooldown recommendation. ____

5.15.2 CONTACT Chemistry for RCP configuration for cooldown recommendation and to verify that RCS activity level (crud) is acceptable and full flow clean-up **NOT** required. ____

5.15.3 CONTACT Health Physics for RCP configuration for cooldown recommendation. ____

Initials

CAUTION

Pressurizer pressure SHALL **NOT** be allowed to drop to less than 1900 psig until the Pressurizer pressure safety injection signal has been blocked per Step 5.25.

5.36 ENSURE Safety Injection blocked per Step 5.25. _____

5.37 ENSURE 50 to 100°F subcooling is maintained continuously while maintaining RCS pressure between 1000 and 1900 psig during RCS cooldown. _____

5.38 WHEN RCS Tavg is 563°F (P-12) AND the steam dumps are being used for RCS cooldown, THEN PERFORM the following (CP007):

5.38.1 RECORD steam dump current demand on “DEMAND UI-0555”. _____

Demand __________%

5.38.2 ENSURE steam dump “DEMAND UI-0555” is 0% using Steam Dump “HDR PRESS CONT PK-0557. _____

5.38.3 Momentarily PLACE steam dump "INTLK SEL" switches in the BYPASS INTERLCK position.

- Train A "INTLK SEL" _____
- Train B "INTLK SEL" _____

5.38.4 SLOWLY RESTORE steam dump “DEMAND UI-0555” recorded in Step 5.38.1 OR as directed by US/SM using Steam Dump “HDR PRESS CONT PK-0557. _____

5.39 ADJUST cooldown rate as required to maintain a differential temperature of less than 25°F between the RCS loops. _____

5.40 MAINTAIN RCP seal injection flow to each RCP between 8 and 13 gpm. _____

5.0 Pressurizer Master Pressure Controller Operations Guideline

5.1 WHEN Operating Below Normal Operation Pressure, THEN ENSURE the following Controllers are in MANUAL:

- Pressurizer Pressure Controller RC-PK-0655A (PZR Press Master Controller)
- Pressurizer Spray Valve Controller "PRZR SPR PCV-0655B"
- Pressurizer Spray Valve Controller "PRZR SPR PCV-0655C"

5.2 RCS/PZR Pressure Control during Heatup/Cooldown Operations SHOULD be performed by varying the Controller Output the following:

- Pressurizer Pressure Controller RC-PK-0655A (PZR Press Master Controller) to Control the output of the PZR CONTROL HTRS
- Pressurizer Spray Valve Controller "PRZR SPR PCV-0655B" and "PRZR SPR PCV-0655C" to Control RCS/PZR Pressure
- Backup Heaters may be cycled as necessary to aid in PZR turnover flow

5.3 The following is the Master Controller outputs in Manual Operation:

Function	Controller Output VDC	Controller Output %	Signal Direction
PCV-0655A OPENS	8.75	87.5	INC (↑)
PCV-0655A CLOSES	7.50	75.0	DEC (↓)
PZR PRESS DEV HI ALARM	7.19	72.0	INC (↑)
SPRAY FULL OPEN	7.19	72.0	INC (↑)
PRES PRESS DEV HI ALARM RESETS	6.56	65.5	DEC (↓)
SPRAY FULL CLOSED	4.06	40.5	DEC (↓)
CONT HTRS 0% PWR	3.44	34.5	INC (↑)
CONT HTRS 50% PWR	2.50	25.0	N/A
CONTROL HTRS 100% PWR	1.56	15.5	DEC (↓)
PZR PRESS DEV LO ALARM & BU HTRS ON	0.94	9.5	DEC (↓)

6.0 Pressurizer Spray Valve Controller Operations Guideline

6.1 Pressurizer Spray Valve Controllers "PRZR SPR PCV-0655B" and "PRZR SPR PCV-0655C" SHOULD remain in Manual and throttled while the associated Spray Valve's RCP is operating. This allows the Pressurizer Spray Valve Controllers operating in Manual to control RCS/PZR pressure.

6.2 WHEN RCP 1A(2A) is **NOT** running, THEN ENSURE Pressurizer Spray Valve Controller "PRZR SPR PCV-0655C" in "MAN" and CLOSED.

6.3 WHEN RCP 1D(2D) is **NOT** running, THEN ENSURE Pressurizer Spray Valve Controller "PRZR SPR PCV-0655B" in "MAN" and CLOSED.

6.4 IF using Auxiliary Spray, THEN control Pressurizer Spray Valves per the Auxiliary Spray Operations Guideline.

7.0 Auxiliary Spray Operations Guideline

7.1 Auxiliary Spray SHALL **NOT** be initiated with a temperature differential greater than 621°F. (TRM 3.4.9.2.c)

7.2 The temperature differential between the pressurizer liquid and the reactor coolant SHALL **NOT** exceed 320°F to minimize the effects of surge line thermal stratification.

7.3 WHEN using Auxiliary Spray, THEN THROTTLE with the Normal Pressurizer Spray Valves to establish the desired flow rate into the Pressurizer. This minimizes thermal cycles on the Auxiliary Spray nozzle and prevents unwanted short cycle of the CVCS flow into the RCS Loops.

7.4 THROTTLING with CHG FLOW CONT VLV FCV-0205 will impact Auxiliary Spray flow. CHG FLOW CONT VLV FCV-0205 should be used to maintain the desired PZR Level and Normal Pressurizer Spray Valves should be used to establish the desired flow rate into the Pressurizer.

Exam Bank No.: 3229 **Last used on an NRC exam:** Never

RO Sequence Number: 56

The power supply for Pressurizer Heater Control Group 1C is 480V Load Center ____.

A. E1A1

B. E1C1

C. 1J2

D. 1N

Answer: D 1N

Exam Bank No.: 3229 **Source:** Modified **Modified from** 2372

K/A Catalog Number: 011 K2.02 Knowledge of bus power supplies to the following: PZR heaters

RO Importance: 3.1 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.14 **Objective Number:** 8860

List the power supplies to the pressurizer heaters.

Reference: LOT201.14 PPT slide 17

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because this is the power supply to another group of Pressurizer Heaters (Backup Heater Group 1A).

B: INCORRECT. Plausible because this is the power supply to another group of Pressurizer Heaters (Backup Heater Group 1B).

C: INCORRECT. Plausible because this is the power supply to another group of Pressurizer Heaters (Backup Heater Group 1E).

D: CORRECT. The power supply to Pressurizer Heater Control Group 1C is 480 V Load Center 1N.

Question Level: F **Question Difficulty** 2

Justification:

Student must recall the power supplies to Pressurizer heaters.

Source For RO #56

Exam Bank No.: 2372 Last used on an NRC exam: 2015

Which of the following Load Centers provide power for the Pressurizer Backup Heater Group B?

A. E1A1

B. E1B1

C. E1B2

D. E1C1

Answer: D E1C1

Exam Bank No.: 2372 **Source:** New **Modified from:**

K/A Catalog Number: 010 K2.01 Knowledge of bus power supplies to the following:

K2.01 PZR heaters

RO/SRO Importance: 3.0 / 3.4 **Tier:** 2 **Group/Category:** 1

RO-10CFR55.41 # 7 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 201.36 **Objective Number:** 92396

LIST the major loads associated with the ESF Electrical System.

Reference: LOT 201.36

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: E1A1 supplies power for the A Group Pressurizer Backup Heaters.
B: INCORRECT: E1B1 does not supply power to any Pressurizer Backup Heaters.
C: INCORRECT: E1B2 does not supply power to any Pressurizer Backup Heaters.
D: CORRECT: E1C1 supplies power for the B Group Pressurizer Backup Heaters.

Level F **Difficulty** 2

Justification:

Must have fundamental knowledge of power supplies for PZR Heaters.

Objective 2

HEATERS – POWER SUPPLIES

- Control Group C – 480V LC 1N
- **Backup Group A – 480V LC E1A1**
- **Backup Group B – 480V LC E1C1**
- Backup Group D – 480V LC 1P
- Backup Group E – 480V LC 1J2

Exam Bank No.: 3165 **Last used on an NRC exam:** Never

RO Sequence Number: 57

A Unit 1 reactor startup is being performed in accordance with 0POP03-ZG-0004, Reactor Startup.

- Initial readings on Source Range Channels NI-31 and NI 32 Counts, were $5X10^1$ CPS

Currently:

- Control Bank C step counter is reading 60 and constant
- Source Range Channels NI-31 and NI-32 Counts are reading $4X10^3$ CPS and rising
- Source Range Channels NI-32 and NI-32 SUR is 0.5 DPM and stable

The reactor is ___(1)___.

The crew should ___(2)___.

A. (1) subcritical
   (2) initiate emergency boration ONLY

B. (1) critical
   (2) initiate emergency boration and fully insert all control banks

C. (1) subcritical
   (2) initiate emergency boration and fully insert all control banks

D. (1) critical
   (2) initiate emergency boration ONLY

Answer: B (1) critical, (2) initiate emergency boration and fully insert all control banks

Exam Bank No.: 3165 **Source:** New **Modified from** N/A

K/A Catalog Number: 015 K5.05 Knowledge of the operational implications of the following concepts as they apply to the NIS: Criticality and its indications

RO Importance: 4.1 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(1)

STP Lesson: LOT 506.01 **Objective Number:** 92158

In regard to the referenced procedure, DISCUSS the following:
2.Precautions
3.Notes and Cautions

Reference: LOT 506. 01 Power Point slide 14; 0POP03-ZG-0004

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as the student could misinterpret whether rod motion is stopped and because if above the POAH emergency boration would be performed but the rods would not be inserted.

B: CORRECT. Per the indications given, the reactor is critical (stable positive SUR with no rod motion and the thumb rule of 5-7 doublings is also met. The action is required by 0POP03-ZG-0004 if you are critical below 65 steps on Control Bank C which means that rods are below the rod insertion limits..

C: INCORRECT. Plausible as the student could misinterpret whether rod motion is stopped. The second part is correct.

D: INCORRECT. Plausible because if above the POAH emergency boration would be performed but the rods would not be inserted. The first part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student has to analyze given conditions to determine reactor criticality and required procedure action.

Initials

CAUTION

- A steady startup rate of 1.0 dpm SHALL **NOT** be exceeded. (Reference 2.5)
- Criticality SHALL be anticipated continually whenever control rods are being withdrawn.
- IF criticality occurs with Control Banks below Technical Specification Rod Insertion Limit for Bank C, THEN emergency boration SHALL be initiated AND all Control Banks fully inserted.
- IF ICRR plot predicts a critical rod height within the next 50 steps, THEN rod withdrawal rate SHALL be reduced to ensure Startup Rate remains less than 1.0 dpm. (Reference 2.5)
- Approach to Criticality - IF Tavg lowers to ≤ 561 °F, THEN manually trip the reactor and GO TO 0POP05-EO-EO00, REACTOR TRIP OR SAFETY INJECTION.

6.10 INITIATE ICRR per Addendum 1. ______

6.11 ENSURE “ROD BANK SEL” switch in MANUAL position. (Reference 2.9) ______

6.12 VERIFY RCS Temperature is greater than 561°F, and Shutdown Banks fully withdrawn prior to any Control Bank withdrawal AND again within 15 minutes prior to achieving criticality using Form 1. (Technical Specification 4.1.1.4.a, 4.1.3.5.a) ______

6.13 ENSURE Shutdown Margin is adequate by verifying ECP rod position greater than Control Bank C at 65 steps AND RECORD time and Date verified. (Technical Specification 4.1.3.6) ______

TIME: ______________ DATE:______________

Initials

6.20 WITHDRAW Control Banks in 50 step increments OR as determined by the Reactor Engineer OR STA OR when reactor achieves criticality AND PERFORM the following:

6.20.1 VERIFY Reactor Condition, (i.e. Subcritical or Critical). _____

6.20.2 IF the Reactor is **NOT** critical, THEN ALLOW count rate to stabilize. _____

6.20.2.1 WHEN count rate has stabilized, THEN RECORD count rate on ICRR Data Sheet. _____

6.20.2.2 PLOT 1/M calculated data point on ICRR Data Sheet. _____

NOTE

IF the Reactor is **NOT** critical, AND P6 Bistable Status Monitoring lights are illuminated, THEN Step 6.21 may be deferred until criticality is achieved.

6.21 WHEN Reactor is critical, THEN PERFORM the following:

6.21.1 IF Reactor is critical with Control Bank C less than 65 steps, THEN INITIATE Emergency Boration AND immediately INSERT all Control Banks fully into core. _____

6.21.1.1 NOTIFY Reactor Engineering to evaluate the unexpected situation. _____

6.21.1.2 NOTIFY Unit Operations Manager. _____

6.21.1.3 WHEN unexpected response is evaluated AND with the Unit Operations Manager approval, THEN GO TO this procedure Step 1.0 to re-perform Reactor Startup. _____

6.21.2 ESTABLISH a steady startup rate of approximately 0.5 dpm. _____

6.21.3 ANNOUNCE twice over Plant Paging System, “Attention all plant personnel, Unit 1 (Unit 2) Reactor is critical”. _____

6.21.4 RECORD time and date of criticality. _____

TIME: ____________ DATE: _____________

6.21.5 LOG in the Control Room Logbook the time of criticality. _____

Exam Bank No.: 3166 **Last used on an NRC exam:** Never

RO Sequence Number: 58

A Large Break LOCA has occurred in Unit 2.

- The crew is performing 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant.
- Reactor Containment Building (RCB) hydrogen concentration indicates 3.4% and rising slowly on the H_2 monitors.
- Hydrogen Recombiner 2A is placed in service.
- Hydrogen Recombiner 2B's breaker trips when placing in service.

Reactor Containment Building hydrogen concentration will _____(1)_____ 4% by volume.

If Hydrogen Recombiner 2A subsequently trips, the ______(2)_____ Containment Purge System can be used to control RCB hydrogen concentration.

A. (1) remain below
(2) Normal

B. (1) remain below
(2) Supplementary

C. (1) exceed
(2) Normal

D. (1) exceed
(2) Supplementary

Answer: B (1) remain below (2) Supplementary

Exam Bank No.: 3166 **Source:** New **Modified from** N/A

K/A Catalog Number: 028 K3.01 Knowledge of the effect that a loss or malfunction of the HRPS will have on the following: Hydrogen concentration in containment

RO Importance: 3.3 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 201.27 **Objective Number:** 97855

Given a plant or system condition, PREDICT the operation of the Containment Combustible Gas Control System.

Reference: LOT 201.27 Power Point slides 10,12 and 13; 0POP02-CG-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as the Normal Containment Purge System is one of two methods used to prurge the RCB atmospherre but the FSAR specifically credits the Supplementary Containment Purge System for Hydrogen control.. First part is correct.

B: CORRECT. Although the H2 Recombiners are no longer part of our design bases, the original design of the CCGC System was that only one of two H2 Recombiners was needed post-LOCA to keep H2 concentration below 4% by volume to satisfy single failure criteria. The Supplemental Containment Purge System is credited in the FSAR for hydrogen control post accident.

C: INCORRECT. Plausible if student is unaware of the design capacity of a Hydrogen Recombiner and the Normal Containment Purge System is one of two methods used to prurge the RCB atmospherre but the FSAR specifically credits the Supplementary Containment Purge System for Hydrogen control

D: INCORRECT. Plausible if student is unaware of the design capacity of a Hydrogen Recombiner. The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions and determine the effect on hydrogen recombiner operation.

1.0 Purpose

Establish the guidelines for operation of the Electric Hydrogen Recombiner System.

2.0 References

2.1 ETB-CG00-9001, Westinghouse Electric Hydrogen Recombiner Model B Technical Manual

2.2 Elementary Diagram 9-E-CG10-01 NSSS Elect Hydrogen Recombiner

2.3 Single Line Diagrams

2.3.1 9-E-PMAL-01 480V Class 1E MCC E1B4(E2B4), EAB

2.3.2 9-E-PMAH-01 480V Class 1E MCC E1C2(E2C2), EAB

2.4 Technical Requirements Manual 3.6.4.

2.5 SER ST-ST-AE-NOC-04001311 (Implements Elimination of the requirements for the Hydrogen Recombiners and moves the Hydrogen Monitor Technical Specification to the Technical Requirements Manual.) (TS Amendments 165 and 155)

2.6 MATS Item 8501727-866, FSAR Section 6.2.5.2.1

2.7 MATS Item, 8700438-860

2.8 0POP02-HC-0001, Containment HVAC

3.0 Prerequisites

3.1 ENSURE Reactor Containment Fan Coolers in operation per 0POP02-HC-0001, Containment HVAC. _____

3.2 PERFORM Lineup 1, Electrical Lineup. _____

4.0 Notes and Precautions

4.1 Heater temperature **SHALL NOT** exceed 1450°F as determined by any one thermocouple.

4.2 DO NOT exceed an output power of 75 KW.

4.3 IF containment hydrogen concentration reaches 3.5% by volume, THEN a Hydrogen Recombiner SHALL be placed in service

4.4 Only one of the two Hydrogen Recombiners is required for hydrogen removal in a Post-LOCA condition, the second Hydrogen Recombiner SHOULD be kept in standby during Post-LOCA.

This procedure, when complete, SHALL be retained for five years.

Obj. 2

Design Basis

- Capability is provided to control the hydrogen concentration in the Containment below LCL following LOCA by purging the Containment atmosphere through the Supplementary Containment Purge Subsystem, which is not safety-related.
  - Available in the latter stages of an accident when the Containment pressure is nearly atmospheric.
  - Meets the requirements of RG 1.7.
- 1.5 percent metal/water reaction is assumed.

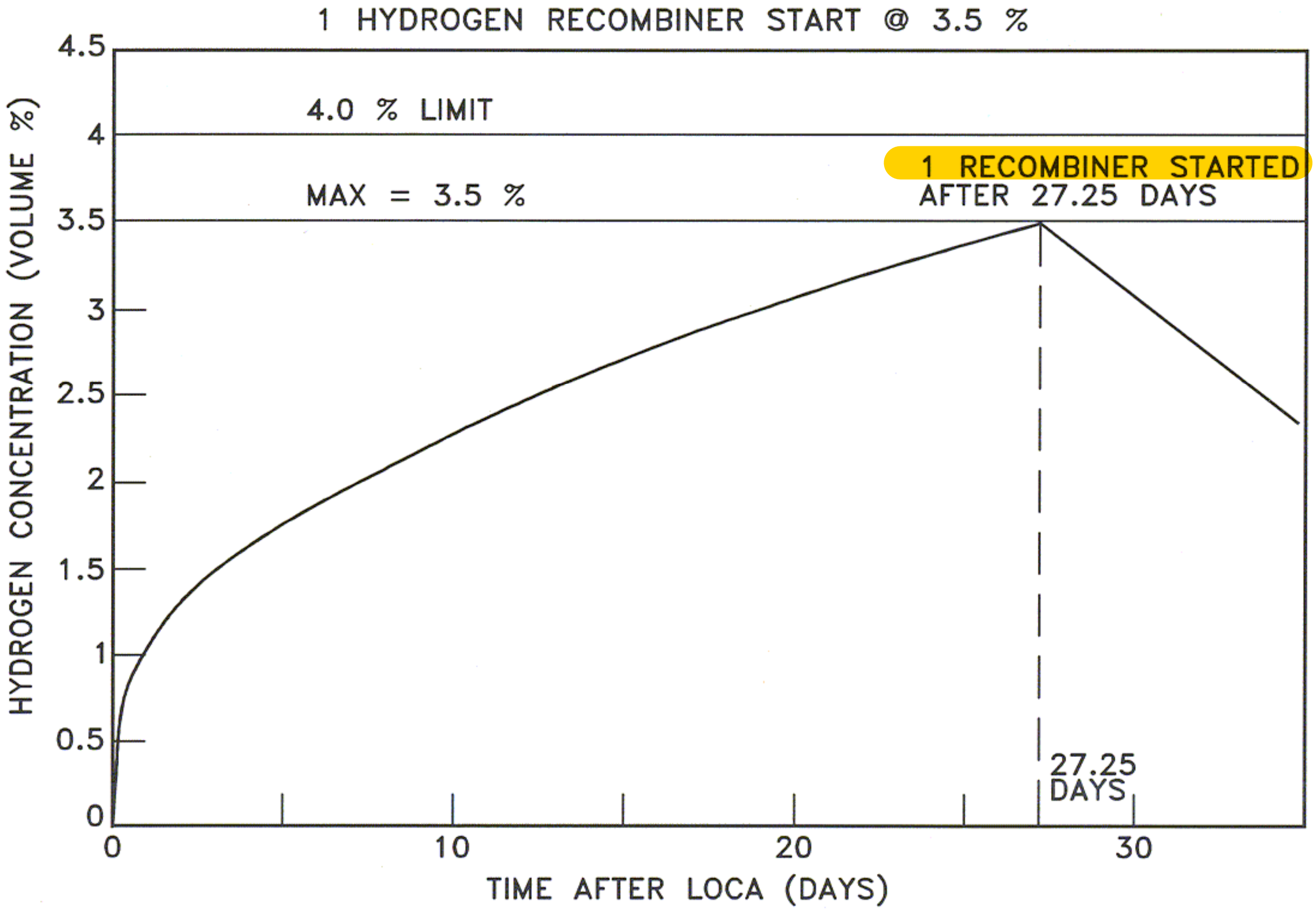
Obj. 2

Design Basis

- Per 10CFR50.44 and the UFSAR, hydrogen recombiners are no longer required for design basis accidents.
- However:
  - One Recombiner is designed to keep cont. H_2 concentration below 4.0% by volume w/o purging or release of radioactive material to environment.
  - Meets requirements of a safety class 2 system.
  - Contribution of combustible gases from secondary sources included in sizing.
  - Designed to meet seismic Category I requirements.

HYDROGEN CONCENTRATION IN CONTAINMENT

Obj. 2



HYDROGEN CONCENTRATION (VOLUME %)

SOUTH TEXAS PROJECT

UNITS 1 & 2

HYDROGEN V/O VS. TIME
AFTER LOCA (DAYS)
(1 RECOMBINER)

Exam Bank No.: 3167 **Last used on an NRC exam:** Never

RO Sequence Number: 59

Unit 1 is in Mode 3.

- During a repair activity inside containment, noble gas activity has increased and stabilized.
- A grab sample of RCB atmosphere confirms a noble gas activity level of 6.50E-4 μCi/cc.
- A RCB Purge Notification Level of 9.75E-4 μCi/cc has been issued by Chemistry.

Based on these conditions the operators should initiate a RCB purge in accordance with ___(1)___ .

To prevent an ESF Containment Ventilation Isolation (CVI) from occurring during the purge, the operators should ___(2)___.

A. (1) 0POP02-HC-0002, Normal Containment Purge
   (2) verify CVI actuation is BLOCKED

B. (1) 0POP02-HC-0002, Normal Containment Purge
   (2) raise the HIGH ALARM setpoint on RT-8012 and RT-8013, RCB Purge Exhaust Monitor

C. (1) 0POP02-HC-0003, Supplementary Containment Purge
   (2) verify CVI actuation is BLOCKED

D. (1) 0POP02-HC-0003, Supplementary Containment Purge
   (2) raise the HIGH ALARM setpoint on RT-8012 and RT-8013, RCB Purge Exhaust Monitor

Answer: D (1) 0POP02-HC-0003, Supplementary Containment Purge (2) raise the HIGH ALARM setpoint on RT-8012 and RT-8013, RCB Purge Exhaust Monitors

Exam Bank No.: 3167 **Source:** Modified **Modified from** 2484

K/A Catalog Number: 029 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Maintenance or other activity taking place inside containment

RO Importance: 2.6 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(12)

STP Lesson: LOT 202.33 **Objective Number:** 33166

DISCUSS 0POP02-HC-002 and 0POP02-HC-0003 including:
B. Precautions
C. Evolutions
D. Notes

Reference: 0POP02-HC-0003; 0POP02-HC-0002

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because 0POP02-HC-0002 is used for RCB purge and blocking CVI actuation but only in Modes 5 - 6 and Defueled. There is one section that can be done in any mode and that is only the section to set the system upp or verify it in the MODE 1-4 condition.

B: INCORRECT. Plausible because 0POP02-HC-0002 is used for RCB purge but only in Modes 5 - 6 and Defueled. There is one section that can be done in any mode and that is only the section to set the system upp or verify it in the MODE 1-4 condition.The second part is correct.

C: INCORRECT. Plausible because blocking CVI actuation is performed in 0POP02-HC-0002 for RCB purges but 0POP02-HC-0002 is only used in Modes 5 - 6 and Defueled. There is one section that can be done in any mode and that is only the section to set the system upp or verify it in the MODE 1-4 condition. The first part is correct.

D: CORRECT. 0POP02-HC-0003 is used to purge the RCB in Modes 1 - 4. With the RCB noble gas activity level above the high alarm setpoint for RT-8012 and RT-8013 (5.00E-4), the procedure directs raising the high alarm setpoint to prevent a Containment Ventilation Isolation actuation prior to performing the purge.

Question Level: H **Question Difficulty** 3

Justification:

The student must be able to analyze the given condition to determine the correct procedure and actions

Source For RO #59

Exam Bank No.: 2484 Last used on an NRC exam: 2016

The Unit is at 100% power with the following condition:

- A recent Chemistry sample of RCB Atmosphere Noble Gas is reading 6.50E-4 µCi/cc resulting in a purge permit notification level of 9.75E-4 µCi/cc.

(1) What procedural actions should be performed prior to starting the purge?

AND

(2) If a purge was started WITHOUT performing this action what would be the consequences?

A. (1) Raise the isolation setpoint on RT-8012 and RT-8013.
(2) An ALERT alarm ONLY will come in on RT-8012 and RT-8013 RCB Purge Exhaust Monitors.

B. (1) Raise the isolation setpoint on RT-8012 and RT-8013.
(2) An ALERT and HIGH alarm will come in on RT-8012 and RT-8013 RCB Purge Exhaust Monitors.

C. (1) Verify Containment Ventilation Actuation is BLOCKED.
(2) An ALERT alarm ONLY will come in on RT-8012 and RT-8013 RCB Purge Exhaust Monitors.

D. (1) Verify Containment Ventilation Actuation is BLOCKED.
(2) An ALERT and HIGH alarm will come in on RT-8012 and RT-8013 RCB Purge Exhaust Monitors.

Answer: B (1) Raise the isolation setpoint on RT-8012 and RT-8013.
(2) An ALERT and HIGH alarm will come in on RT-8012 and RT-8013 RCB Purge Exhaust Monitors.

Exam Bank No.: 2484 **Source:** New **Modified from:**

K/A Catalog Number: 029 A2.04 Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System, and (b) based on those predictions, use procedures to correct, or mitigate the consequences of those malfunctions or operations:
Health physics sampling of containment atmosphere.

RO/SRO Importance: 2.5 / 3.2 **Tier:** 2 **Group/Category:** 2

RO-10CFR55.41 # 10 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 202.33 **Objective Number:** 97097

Discuss 0POP02-HC-0003 including: A. Purpose and scope B. Precautions C. Notes.

Reference: 0POP02-HC-0003, Supplemental Containment Purge Rev 25 page 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because the student has to be aware of the isolation alarm (High Alarm) set point that would cause an ESF Containment Ventalation Isolation (CVI).

B: CORRECT: RT-8011 reading is above the level which will cause an isolation alarm (High Alarm) on RT-8012 and RT-8013 (5.00E-4). The procedural action is to raise the RT-8012 and RT-8013 isolation alarm set point (High Alarm) to prevent an ESF actuation prior to performing the purge.

C: INCORRECT: Plausible because the student has to be aware of the isolation alarm (High Alarm) set point that would cause an ESF Containment Ventalation Isolation (CVI). Also, you can block CVI but the procedural action to verify that CVI is blocked is for Mode 5 and 6 and Defueled only.

D: INCORRECT: Plausible because you can block CVI but the procedural action to verify that CVI is blocked is for Mode 5 and 6 and Defueled only.

Level H **Difficulty** 3

Justification:

The student must be able to analyze the given condition to determine the correct procedural actions and have knowledge of the ESF actuation setpoint for CVI. NOTE: Chemistry performs samples of containment at STP. NOT Health Physics.

Lineup 4	Mode 4 Lineup	Page 2 of 3

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY
HC-VFN007	SPLY FAN 11A(21A)	Control Room CP022	NORMAL		
HC-VFN008	SPLY FAN 11B(21B)	Control Room CP022	NORMAL		
HC-VFN009	EXH FAN 11A(21A)	Control Room CP022	NORMAL		
HC-VFN010	EXH FAN 11B(21B)	Control Room CP022	NORMAL		
HC-CRHS-0007	SPLY OCIV MOV-0007	Control Room CP022	CLOSED/AUTO		
HC-CRHS-0009	EXH ICIV MOV-0009	Control Room CP022	CLOSED/AUTO		
HC-CRHS-0008	SPLY ICIV MOV-0008	Control Room CP022	CLOSED/AUTO		
HC-CRHS-0010	EXH OCIV MOV-0010	Control Room CP022	CLOSED/AUTO		
E1A2(E2A2)/J3	NORMAL CNTMT PURGE SUPPLY OCIV 1(2)-HC-MOV-0007	EAB 10' Train A Switchgear Room 010 MCC E1A2(E2A2)	LOCKED OFF		
E1A1(E2A1)/J3	NORMAL CNTMT PURGE EXHAUST ICIV 1(2)-HC-MOV-0009	EAB 10' Train A Switchgear Room 010 MCC E1A1(E2A1)	LOCKED OFF		
E1B1(E2B1)/J1	NORMAL CNTMT PURGE SUPPLY ICIV 1(2)-HC-MOV-0008	EAB 35' Train B Switchgear Room 212 MCC E1B1(E2B1)	LOCKED OFF		
E1B2(E2B2)/J1	NORMAL CNTMT PURGE EXHAUST OCIV 1(2)-HC-MOV-0010	EAB 35' Train B Switchgear Room 212 MCC E1B2(E2B2)	LOCKED OFF		

Lineup 4	Mode 4 Lineup	Page 3 of 3

DEVICE NUMBER	COMPONENT NOUN DESCRIPTION	LOCATION	POSITION REQUIRED	ALIGNED BY	VERIFIED BY
HC-MOV-0007	REACTOR CONTAINMENT BUILDING NORMAL PURGE SUPPLY (ORC) ISOLATION VALVE	FHB EL. 93’ 112˚ Room 304	LOCKED CLOSED		
HC-MOV-0008	NORMAL CNTMT PURGE SPLY ICIV	RCB EL. 83’ Room 501D	LOCKED CLOSED		
HC-MOV-0009	NORMAL CNTMT PURGE EXH ICIV	RCB EL. 83’ Room 501D	LOCKED CLOSED		
HC-MOV-0010	REACTOR CONTAINMENT BUILDING NORMAL PURGE EXHAUST (ORC) ISOLATION VALVE	MAB EL. 68’ Room 326	LOCKED CLOSED		

4.0 Notes and Precautions

4.1 IF in Modes 1 through 4, THEN containment pressure SHALL be maintained between -0.1 and +0.3 psig. (Technical Specification 3.6.1.4)

4.2 IF in Modes 1 through 4, THEN supplementary purge SHALL be terminated as soon as the reason for purge has been satisfied.

4.3 Changing conditions, such as RCFC configuration and outside weather conditions, will have an effect on the rate of Containment pressure change once the purge is commenced. Operators should be prepared to secure the purge evolution promptly to avoid entering the one hour TS LCO action statement.

4.4 For RCB Supplementary Purge System, opening purge valves without use of a fan constitutes a containment purge.

4.5 The Normal Containment Purge System and the Supplementary Containment Purge System SHALL NOT be operated simultaneously with the exception of testing per 0PSP03-SP-0016, Containment Ventilation Isolation Actuation and Response Time Test.

4.6 When Containment Vent Isolation (CVI) is OPERABLE, a High Radiation Alarm on RT-8012 or RT-8013 will automatically close the supplementary containment purge isolation valves and isolate RT-8011.

4.7 Calculation ZC 7015, using the Offsite Dose Calculation Manual methodology, determined that the setpoint for RT-8012 and RT-8013 may be increased to 6.69E-03 µCi/cc and the allowable value for RT-8012 and RT-8013 may be increased to 8.36E-03 µCi/cc, respectively, as allowed by Technical Specifications. However, this procedure uses the lower value of 3.00E-03 µCi/cc as the increased setpoint for conservatism.

4.8 IF an RMA is in effect, THEN 0POP01-ZO-0006 may require Containment Purges be limited to those necessary to satisfy Technical Specifications. (Reference 2.13)

4.9 Steps which require Dual Verification are preceded by **DV** before the first word of the step.

4.10 This procedure will refer to “A1(2)RA-RI-8012B REACTOR CONT BLDG PURGE ISOLATION” as RT-8012.

4.11 This procedure will refer to “C1(2)RA-RI-8013B REACTOR CONT BLDG PURGE ISOLATION” as RT-8013.

4.12 A BYPASS/INOP alarm will occur on Panel 26M023 when the valves to the purge radiation monitors are not aligned properly with supplementary purge "EXH OCIV FV-9777" not closed. (Reference 2.2.4)

6.0 Supplementary Purge System Operation or Containment Pressure Control in Modes 1-4

NOTE

- Chemistry will provide a copy of 0PCP09-HC-0001 Form 1, RCB Purge Notification Levels, which contains the activity readings at which Chemistry SHALL be notified that RCB purge criteria may require reevaluation.
- WHEN RT-8011 is inoperable, THEN RT-8012 AND RT-8013 readings are compared to their notification levels on the RCB Purge Notification Levels form, after the RCB purge has commenced. (see Note prior to Step 6.15) Flow is required for accurate readings on RT-8012 AND RT-8013.

6.1 ENSURE Section 5.0 has been performed. _____

6.2 VERIFY both RT-8012 AND RT-8013 OPERABLE. _____

6.3 IF RT-8011 is OPERABLE, THEN PERFORM the following:

6.3.1 COMPARE the RT-8011 Particulate, Iodine, and Noble Gas Channel activity readings indicated on the RM-11 or RM-23A 10-minute trends to the Notification Levels on the RCB Purge Notification Levels form. _____

6.3.1.1 IF the RT-8011 Particulate, Iodine, or Noble Gas Channel activity reading is **greater than or equal to** the notification level listed on the RCB Purge Notification Levels form, THEN INFORM Chemistry that RCB purge criteria may require reevaluation. _____

CAUTION

IF noble gas concentration is greater than 5.00E-04 µCi/cc AND purge is initiated, THEN an ESF Containment Ventilation Isolation will occur. To prevent the ESF actuation, RT-8012 and RT-8013 isolation setpoints may be raised for the supplementary purge provided the setpoints are returned to 5.00 E-04 µCi/cc following the purge. (Reference 2.11)

6.4 IF RT-8012 AND RT-8013 require a setpoint increase, THEN GO TO Section 10.0. _____

6.5 IF Section 10.0 was NOT performed, THEN PERFORM the following:

- VERIFY RT-8012 High Alarm setpoint is 5.00E-04 µCi/cc on RM-11 or RM-23A. (Reference 2.10) ______
- VERIFY RT-8013 High Alarm setpoint is 5.00E-04 µCi/cc on RM-11 or RM-23A. (Reference 2.10) ______

6.6 OBSERVE ventilation stack flow prior to initiating purge. (Plant Computer Point HMFA9308, RM-11 or CP023) ______

CAUTION

IF an RMA is in effect, THEN 0POP01-ZO-0006 may require Containment Purges be limited to those necessary to satisfy Technical Specifications. (Reference 2.13)

6.7 OPEN the following valves: (CP022) (Reference 2.7)

- "SPLY OCIV FV-9776" ______
- "SPLY ICIV MOV-0003" ______
- "EXH ICIV MOV-0005" ______
- "EXH OCIV FV-9777" ______

6.8 VERIFY the following valves OPEN: (Reference 2.7)

- "SPLY OCIV FV-9776" ______
- "SPLY ICIV MOV-0003" ______
- "EXH ICIV MOV-0005" ______
- "EXH OCIV FV-9777" ______

6.9 RECORD Date/Time purge is initiated below: (valves opened at Step 6.7) ______

Date: ________ Time: ________Hrs.

Exam Bank No.: 3228 **Last used on an NRC exam:** Never

RO Sequence Number: 60

Per the UFSAR and Technical Specifications, primary to secondary leakage will be limited to _______________ gallons/day through any one steam generator.

A. 30

B. 75

C. 150

D. 500

Answer: C 150

Exam Bank No.: 3228 **Source:** New **Modified from** N/A

K/A Catalog Number: 035 G2.2.38 Knowledge of conditions and limitations in the facility license: Steam Generators

RO Importance: 3.6 **Tier:** **Group/Category:** **10CFR Reference:** 55.41(b)(4)

STP Lesson: LOT 503.01 **Objective Number:** 31089

DETERMINE the applicable Technical Specification and/or the Technical Requirements Manual (TRM)Limiting Conditions for Operation (LCOs) and the required actions to be taken.

Reference: UFSAR Chapter 5 (5.2.5.1.2.c) TS 3.4.6.1 bases

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because this number is used in Addendum 6 of 0POP04-RC-0004 and 0PGP03-ZO-0041 for determining actions to take for low level primary-to-secondary leakage.

B: INCORRECT. Plausible because this number is used in Addendum 6 of 0POP04-RC-0004 and 0PGP03-ZO-0041 for determining actions to take for low level primary-to-secondary leakage.

C: CORRECT. Per USFAR paragraph 5.2.5.1.2 and Tech Spec 3.4.6.2 the limit for primary to secondary leakage is 150 gpd in any steam generator

D: INCORRECT. Plausible because this number is used in the 3.4.6.2 Tech Spec bases for primary to secondary leakage as the number used in the safety analysis for the faulted SG primary to secondary leakage in the steam line break safety analysis.

Question Level: F **Question Difficulty** 2

Justification:

Student must recall primary to secondary leakage limits for Steam Generators.

Section XI. System leak tests are conducted prior to startup following each reactor refueling outage in accordance with IWB-5000.

Examinations performed during these tests may be conducted without the removal of insulation. The system leakage test program is consistent with the operating limitations during heatup and cooldown, as provided in Chapter 16.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection System

The RCPB Leakage Detection System provides a means of detecting significant leakage from the RCS. The system was designed to conform with NRC General Design Criterion (GDC) 30 and RG 1.45. identifiable leakage from the reactor pressure vessel (RPV) flange leakoff, valve leakoffs, RCP leakoffs, and drain line leakage is collected and measured in the reactor coolant drain tank (RCDT). Unidentified leakage is determined by measuring the increase in Containment sump level, Containment air particulate radioactivity, and Containment humidity. Leakage from the RCPB to auxiliary systems is monitored by the measurement of radioactivity and water inventory balances. Measurement of RCPB leakages is sufficiently sensitive to assure that small increases in leakage can be detected while total leakage remains below a value consistent with safe plant operation.

5.2.5.1 Reactor Coolant Pressure Boundary Leakage.

5.2.5.1.1 Normal Expected Leakage: Unidentifiable leakage from the RCPB into the Containment is expected to be less than 0.02 gal/min. Identifiable leakage into the RCDT is expected to be less than 0.3 gal/min, primarily from the RCPs seals.

5.2.5.1.2 Limits for Reactor Coolant Leakage: RCS leakage will be limited to the following by the STPEGS Technical Specifications:

a. No pressure boundary leakage

b. One gal/min unidentified leakage

c. 150 gal/day primary-to-secondary leakage through any one steam generator

d. Ten gal/min identified leakage from the RCS

e. 0.5 gal/min leakage per nominal inch of valve size up to a maximum of 5 gal/min at an RCS pressure of 2,235 ± 20 psig from any RCS Pressure Isolation Valve specified in Table 5.2-7. Test pressures less than 2,235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2,235 psig, assuming the leakage to be directly proportional to pressure differential to the one-half power.

5.2.5.1.3 Collection of Identified Leakage: Leakage from the RPV flange, valves, reactor coolant pump seals, and equipment drains is collected in the RCDT. The collected fluid is recirculated and cooled. As the level in the tank reaches a preset level, the discharge valve is opened, allowing the excess fluid to be discharged to the Boron Recycle System (BRS) holdup tanks. The discharge flow is measured by an integrating flow meter. Increased identified leakage from the

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The UFSAR analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the main steam safety valves and the majority is steamed to the condenser. The 1 gpm primary-to-secondary leakage safety analysis assumption is relatively inconsequential.

The SLB is more limiting for primary-to-secondary leakage. The safety analysis for the SLB assumes 500 gpd and 936 gpd primary-to-secondary leakage in the faulted and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was 1.0 μCi/gm DOSE EQUIVALENT 1-131 at a conservatively high letdown flow of 250 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

No PRESSURE BOUNDARY LEAKAGE is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.

b. UNIDENTIFIED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

Leakage from systems connected to the Reactor Coolant System will initially manifest itself as UNIDENTIFIED LEAKAGE until the source of the leak is identified. If the leakage exceeds the 1 gpm limit for UNIDENTIFIED LEAKAGE, then Action b is entered. When the source of the leakage is identified and UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are verified within limits, Action b can be exited.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpd per each steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 1). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, “The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

e. Reactor Coolant System Pressure Isolation Valve Leakage

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

Exam Bank No.: 3227 **Last used on an NRC exam:** Never

RO Sequence Number: 61

Unit 1 is in MODE 3.

- Tave is 567°F.
- Pressurizer pressure is 2235 psig.
- Steam Dumps are in the STEAM PRESSURE mode.

The "STM DUMP UNBLK AVAIL" light is ___(1)___.

If Steam Header Pressure Instrument PT-0557 fails HIGH, Steam Dump valves will **INITIALLY** ____(2)____.

A. (1) LIT
(2) CLOSE

B. (1) LIT
(2) OPEN

C. (1) OFF
(2) CLOSE

D. (1) OFF
(2) OPEN

Answer: B (1) LIT; (2) OPEN

Exam Bank No.: 3227 **Source:** New **Modified from** N/A

K/A Catalog Number: 041 K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS

RO Importance: 2.7 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(7)

STP Lesson: LOT 202.09 **Objective Number:** 93002

Given plant conditions, DETERMINE their effects on the Steam Dump System.

Reference: LOT 202.09

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if student inverts the effect of setpoint vs. process. The first part is correct.

B: CORRECT. If PT-557 fails high it is now higher than setpoint and controller setpoint will go to 100% and the steam dumps will initially open. The RCS will then cooldown and the valves will reclose when Tavg reaches 563 degrees F on the P-12 interlock. The valves then cycle around the P-12 setpoint as the RCS heats up, reopening the valves.

C: INCORRECT. Plausible because Steam Dump indications for the two operating modes are frequently misunderstood and student can invert effect of setpoint vs. process.

D: INCORRECT. Plausible because Steam Dump indications for the two operating modes are frequently misunderstood. The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

Student must evaluate conditions given in the stem and determine system response.

5. Instrumentation Display

5.1 The following instrumentation is located on the operator console panels to assist the operator in determining steam dump system status: (CP007)

5.1.1 Main steam bypass header pressure

5.1.2 Total steam dump demand

5.1.3 Steam dump valve position status lights. (Both on in mid-position)

5.1.4 Steam dump drain valve position status lights

5.1.5 On CP007 there are two round white status lights

Upper - "STM DUMP VLVS TRP"; on when in Tavg Mode and any Trip Open BS actuates.

Lower- "STM DUMP UNBLK AVAIL", lit if you have C9 coincident with a C7, C8, or selected to Steam Pressure Mode.

5.2 Status and permissive lamp boxes

5.2.1 The following indications are located on CP005 status and permissive lamp boxes:

LO-LO TAVG COND CLDN VLVS BYPASSED

1. Illuminates if P-12 is bypassed allowing operation of the bank 1 valves.

UNIT 1: C7 STEAM INLET PRESS STM DUMP PERMISSIVE

UNIT 2: C7 TURB IMP PRESS STM DUMP PERMISSIVE

2. Illuminates if C-7 B/S actuates.

P-12 MAN BYPASS CLDN VALVES PERM DUMP VLVS BLOCKED

3. Illuminates if P-12 actuates (2/4 Tavg ≤ 563°F)

COND AVAIL FOR STM DUMP

4. Normally illuminated, extinguishes if C-9 is lost

Scenario #2

The Turbine is at 30% power following a Load Rejection. C-7 has not reset. PT-505 fails low.

Describe the response of the Steam Dump System.

What would be the response if PT-505 failed high?

Scenario #3

Turbine load is being rapidly reduced due to degrading vacuum.

The STM DMP UNBLK AVAIL lamp lights on CP007. The turbine trips at 40% reactor power due to low vacuum.

Describe the response of the Steam Dumps during this transient.

Scenario #4

The Tavg input to the Steam Dumps becomes inoperable. The system is transferred to Steam Pressure Mode.

The plant trips from 100% power.

How will the Steam Dumps respond?

Scenario #5

The plant is in Mode 2 pulling rods toward criticality. Steam Dumps are in Steam Pressure Mode controlling steam pressure at 1185 psig.

PT-557 fails high.

Describe the response of the Steam Dump System.

PLANT SCENARIO ANSWERS

Scenario #1

Bank 1 and Bank 2 Steam Dumps will be "tripped" open (7.5ºF delta T). Bank 3 will be slightly modulated open. Steam Dumps will modulate to reduce temperature to 587.5°F (584.5 °F + 3 °F deadband). With rods in Automatic, initially they will step in at 72 steps per minute.

Scenario #2

With Steam Dumps already armed, once PT-505 input for Tref, fails low (567°F), the Steam dumps will control as in scenario #1 with temperature controlled at 570°F.

With PT-505 failed high (clipped to 592.5°F), Steam dumps would close.

Scenario #3

Lo Vacuum trips the turbine and also removes C-9 (Condenser Available). Steam Dumps will not open. With Tavg at 575.8°F, the Steam Generator PORV will open to lower temperature to 572°F (saturation for 1225 psig).

Scenario #4

The Steam Header Pressure Controller is normally set for approximately 1185 psig (8.46 on the potentiometer). Following the trip, steam pressure will increase from approximately 1066 psig to 1435 psig, causing the Steam Dumps to open after pressure increases above 1185 psig.

Scenario #5

The Steam Pressure Header Controller is a 1% / psig proportional gain controller. With PT-0557 failed high (1400 psig), there is a 217 psig difference, i.e. a 217% demand to open all steam dump valves. Since header pressure is failed, temperature will drop to 563°F and cycle around this temperature.

Exam Bank No.: 3169 **Last used on an NRC exam:** Never

RO Sequence Number: 62

Unit 2 is at 55% power.

- Annunciator 07M3 Window A-2 “EHC SUPPLY PUMP TRIP” is in alarm.
- The standby EHC pump failed to start due to a breaker trip.
- EHC discharge pressure indicates 1310 psig and is lowering at 10 psi/minute.

The operators should…

A. ensure the turbine tripped and GO TO 0POP04-TM-0003, Turbine Trip Below P-9.

B. enter 0POP04-TM-0005, Fast Load Reduction and reduce turbine load at 5% per minute.

C. ensure the reactor is tripped and GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.

D. dispatch an operator to reset the standby EHC pump breaker to restore EHC discharge pressure to normal.

Answer: C ensure the reactor tripped and enter 0POP05-EO-EO00, Reactor Trip Or Safety Injection.

Exam Bank No.: 3169 **Source:** New **Modified from** N/A

K/A Catalog Number: 045 K1.18 Knowledge of the physical connections and/or cause-effect relationships between the MT/G system and the following systems: RPS

RO Importance: 3.6 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(4)

STP Lesson: LOT 202.03 **Objective Number:** 101677

GIVEN a plant or system condition, PREDICT the operation of the Main Turbine System.

Reference: LOT 202.06 PPT slide; 0POP09-AN-07M3 Window A-2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student thinks that the conditions warrant a turbine trip and implementation of 0POP04-TM-0003.

B: INCORRECT. Plausible because other conditions such as high turbine vibration would warrant a rapid load reduction as a means to stabilize the degrading condition. Additionally a 5-7% power reduction would prevent a reactor trip on turbine trip.

C: CORRECT. If reactor power is above P-9 (50%) and EHC discharge pressure is approaching 1300 psig then the ARP directs the operator to ensure reactor trip and enter 0POP05-EO-EO00, Reactor Trip Or Safety Injection. An automatic reactor trip will not occur until 1245.8 psig so the operators have time to perform a manual trip per the ARP.

D: INCORRECT. Plausible as the action to reset the breaker is directed by the ARP if prior steps to trip the reactor are not applicable.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions and determine the correct action and procedure.

EHC SPLY PUMP TRIP

Automatic Actions: Standby EHC pump starts at 1600 psig lowering.

Immediate Actions: None

Subsequent Actions:

______ 1) ENSURE standby EHC pump running.

______ 2) VERIFY EHC discharge pressure is being maintained at approximately 2000 psig on PI-6308 OR Plant Computer Point P6304.

______ 3) IF Reactor Power is greater than P-9 (50%) AND EHC pressure is approaching 1300 psig, THEN PERFORM the following:

______ a) ENSURE the Reactor tripped.

______ b) ENSURE the Turbine and Generator tripped.

______ c) ENSURE all SGFPTs tripped

______ d) GO TO 0POP05-EO-EO00, Reactor Trip or Safety Injection.

______ 4) IF Reactor Power is less than P-9 (50%) AND EHC pressure is approaching 1300 psig, THEN PERFORM the following:

______ a) ENSURE the Turbine and Generator tripped.

______ b) ENSURE all SGFPTs tripped

______ c) ENSURE SU SGFP is running

______ d) GO TO 0POP04-TM-0003, Turbine Trip Below P-9.

______ 5) IF the alarm is the result of an EHC reservoir extreme low level, THEN TAKE appropriate action per Annunciator Response 07M3 Window B-2 "EHC RSVR EXTREME LO-LO LVL".

______ 6) DISPATCH an Operator to check the following applicable breaker position:

______ a) "EHC OIL PUMP 11(21) 1G3(2G3)/C4".

______ b) "EHC OIL PUMP 12(22) 1G6(2G6)/C4".

Obj 12

EHC Fluid Pressure Low Reactor Trip

- 2/3 pressure channels less than 1245.8 psig,

  IF

- Reactor Power is above P-9 (50%)

Exam Bank No.: 3170 **Last used on an NRC exam:** Never

RO Sequence Number: 63

Unit 2 is in Mode 5 during a plant shutdown for upcoming maintenance on the Circulating Water (CW) system.

- A liquid waste release of a Waste Monitor Tank (WMT) is in progress in accordance with 0POP02-WL-0100, Liquid Waste Release.
- CW Pumps 21, 22 and 23 are in operation to support the liquid waste discharge.

CW Pump 21 Trips.

The liquid waste release will need to be stopped ___(1)___.

After the release is secured, the CW pumps shall remain in service for a minimum of ___(2)___ hours.

A. (1) locally from the Radwaste Control Room
(2) 1.5

B. (1) remotely from Control Room Panel CP-018
(2) 1.5

C. (1) locally from the Radwaste Control Room
(2) 24

D. (1) remotely from Control Room Panel CP-018
(2) 24

Answer: A (1) locally from the Radwaste Control Room (2) 1.5

Exam Bank No.: 3170 **Source:** Modified **Modified from** 2266

K/A Catalog Number: 068 K4.01 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids

RO Importance: 3.4 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(13)

STP Lesson: LOT 203.11 **Objective Number:** 101637

GIVEN an abnormal plant condition concerning equipment associated with the Liquid Waste Processing System (LWPS), DETERMINE the probable cause of the condition and any corrective actions if necessary.
LOT 202.22 (29966) - DISCUSS the proper use of the circulating water system operating procedure for normal and abnormal operating conditions. Include the system operating parameters to be observed at different power levels.

Reference: 0POP02-WL-0100, Rev. 25; 0POP02-CW-0001, Rev. 93

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Per 0POP02-WL-0100, all liquid waste releases are controlled from the Radwaste Control Room under direction of Control Room personnel, the CW pump trip requires that the release be stopped. In the event all CW pumps are to be secured, the procedure requires that the CW system remain in service for a minimum of 1.5 hours to prevent potential contamination of the system.

B: INCORRECT. Plausible because a gaseous radwaste release can be controlled from CP-018 in the Control Room. The second part is correct.

C: INCORRECT. Plausible as 24 hours is used as the time frame for conducting the radiation monitor channel check and to secure a CW pump on a loss of seal water flow.

D: INCORRECT. Plausible because a gaseous radwaste release can be controlled from CP-018 in the Control Room and 24 hours is used as the time frame for conducting the radiation monitor channel check and to secure a CW pump on a loss of seal water flow.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of controls for the liquid radwaste system and procedures affected by the discharge and CW Pump trip.

Source For RO #63

Exam Bank No.: 2266 **Last used on an NRC exam:** 2014

RO Sequence Number: 62

Unit 2 is in Mode 5 and preparing to shut down all of the Circulating Water (CW) System for maintenance.

A release of Waste Monitor Tank (WMT) 2C is currently in progress with 3 CW Pumps in operation.

Which of the following describes where the liquid radwaste release will be secured and the EARLIEST time that all CW Pumps can be stopped if the liquid radwaste release is secured at 1400 hours?

	Where Liquid Radwaste will be Secured	**EARLIEST time to Stop All CW Pumps**
A.	Remotely in the Main Control Room at CP-022.	1530 hours
B.	Locally at the Mechanical Auxiliary Building Control Panel.	1400 hours
C.	Remotely in the Main Control Room at CP-022.	1400 hours
D.	Locally at the Mechanical Auxiliary Building Control Panel.	1530 hours

Answer: D Locally at the Mechanical Auxiliary Building Control Panel. - 1530 hours

Exam Bank No.: 2266 **Source:** Bank **Modified from**

K/A Catalog Number: 068 G2.3.11 Liquid Radwaste,
Ability to control radiation releases.

RO Importance: 3.8 **Tier:** 2 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(12)

STP Lesson: LOT 202.22 **Objective Number:** 23901

DISCUSS the proper use of the circulating water system operating procedure for normal and abnormal operating conditions. Include the system operating parameters to be observed at different power levels.

Reference: 0POP02-CW-0001, Circulating Water System Pump Operation, Notes and Precautions 4.21 & 4.22

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: This distractor is credible because a gaseous radwaste release can be controlled from CP-018 in the Control Room (Normal and Supplemental Purge) but not any liquid radwaste releases which are controlled locally from the LWPS Control panel in the MAB.

B: INCORRECT: This distractor is credible because the time listed would be correct if CW Pump configuration was being reduced and not fully secured.

C: INCORRECT: This distractor is credible because a gaseous radwaste release can be controlled from CP-018 in the Control Room (Normal and Supplemental Purge) but not any liquid radwaste releases which are controlled locally from the LWPS Control panel in the MAB. Also, the time listed would be correct if CW Pump configuration was being reduced and not fully secured.

D: CORRECT: The liquid radwaste release is controlled form the LWPS control panel in the MAB and the CW procedure requires that CW remain in service for a minimum of 1.5 hours after a liquid radwaste release is secured in order to prevent contamination of the CW system.

Question Level: F **Question Difficulty** 2

Justification:

The student must have fundamental of controls for the liquid radwaste system and procedural knowledge of how liquid radwaste and CW system interface.

4.21 WHEN an LWPS release is in progress, THEN the release SHALL be secured prior to reducing the number of running CW Pumps to LESS THAN the number assumed to be running per the current effluent release permit.

4.22 WHEN LWPS release is in progress, THEN the release SHALL be secured prior to securing ALL CW Pump to prevent potential contamination of CW system. CW Pumps SHALL remain in service for a minimum of 1.5 hours after LWPS release is secured. (References 2.11 and 2.12)

4.23 IF a CW Pump is started, THEN its vacuum breaker isolation valve SHALL be opened after a 5-minute stabilization period.

4.24 IF a CW Pump is operating, THEN its vacuum breaker isolation valve SHALL be open.

4.25 IF a CW Pump is secured, THEN its vacuum breaker isolation valve SHALL be closed after a 5-minute stabilization period.

4.26 IF **ONLY Two** CW Pumps are operating, THEN it is preferred to run both CW Pumps in one leg (i.e. CW Pump 11(21) with 12(22) or CW Pump 13(23) with 14(24)). Running both pumps in the same leg will help to maintain discharge pressure of the pumps GREATER THAN 3 psig (2 psig), which will prevent air from being drawn into the CW System through the vacuum breakers.

4.27 **(UNIT 1 ONLY)** IF **ONLY Three** CW Pumps will be operating AND CW Pump 12 will be one of the operating pumps, THEN CW Pump 11 SHOULD also be operated. This is to prevent CW Pump 12 runout, which lowers system pressure below 3 psig at the vacuum breaker that MAY cause air entrainment in the CW system. IF CW Pump 11 is unavailable, THEN three pump-CW System operation is permissible as long as the CW Pump 12 operating parameters are satisfactory.

4.28 IF performing a total pump down of the Circulating Water System, THEN the four top manway covers on the discharge piping **AND** the two top manway covers at the CW Outfall structure SHALL be removed **AND** the CW Pump discharge MOVs OPEN prior to permitting personnel entry into the pipeline.

4.29 WHEN starting or stopping a CW Pump, THEN ALL non-essential personnel SHALL be clear of the CWIS pit area. The Unit Supervisor/Shift Manager MAY waive this requirement as long as those personnel remaining are aware of the evolution.

4.30 Addendum 1 (CW Pump Start Permissive Restoration) MAY be performed at any time to restore the pump start interlocks for an available CW Pump.

4.31 The purpose of Addendum 2, (Waterbox Priming Air Trap Operation), is to place the Waterbox Priming Air Traps in service, as directed by Section 6.0 or Section 19.0, **OR** to return a leaking Waterbox Priming Air Trap to service (i.e., isolated after being identified as the source of inleakage into the Vacuum Control Tank).

4.32 Waterbox Priming Air Traps are designed to function **ONLY** with the Waterbox Priming System in operation.

4.0 Notes and Precautions

4.1 Any activity that requires entry or access to a Radiologically Controlled Area (RCA) requires a Radiation Work Permit.

4.2 Improper performance of steps in this procedure could result in contamination of non-contaminated systems. (Reference 2.2.3)

4.3 LWPS discharge flow rate shall be limited to less than 280 gpm.

4.4 Radiation Monitor RT-8038 flow switch should be given time to reset before closing sample inlet and outlet valves (1(2)-RA-3267 & 1(2)-RA-3268, respectively) to avoid RM-11 alarms when monitor secured.

4.5 The Channel Check of RT-8038 SHALL be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made. Since TDS Tank releases are "batch" releases which take less than 24 hours to complete, performance of the channel check prior to commencing the release satisfies this requirement. Re-performance of the channel check is NOT required if the release spans 2 calendar days.

4.6 Strainer baskets upstream of RT-8038 low flow switch have been removed. (Ref. 2.3.10)

4.7 1(2)WL-FQI-4078 LWPS Discharge Header Flow Totalizer Indicator is calibrated once every fuel cycle per 0PSP05-WL-4078 (Plant Liquid Waste Discharge Flow Calibration (F-4078). Consult the Operability Assessment System (OAS) as necessary to determine indicator operability.

4.8 IF the number of running CW pumps is decreased below the number used to generate the Pre-Release Permit, THEN the release SHALL be terminated immediately.

4.9 Steps or sections of this procedure which are NOT used may be marked N/A if not applicable for the evolution in progress. This is indicated by being routed around a step or section or by a conditional step or section where the condition is NOT met (IF - THEN, etc.).

4.10 Individuals performing portions of this procedure SHALL be identified by signing the "Performers and Verifiers" block on the Procedure Performance Data Sheet.

4.11 Tolerance of WL-FT-4078 is + or - 21 gpm (9Z019ZS1038 states "+ or - 0.5% of [300 psid] span") which is the tolerance in 0PSP05-WL-4078, Plant Liquid Waste Discharge Flow Calibration (F-4078) of + or - 0.5 VDC.

Initials

5.53 IF the running number of CW pumps becomes less than that assumed on the Effluent Release Permit, OR Radiation Monitor RT-8038 trips for any reason, THEN immediately PERFORM the following:

5.53.1 PLACE “1(2)WL-FV-4077 WASTE MONITOR TANK PUMPS 1A, B & C(2A, B & C) DISCHARGE DIVERT VALVE” in RECIRC position. (Radwaste Control Rm on ZLP-189) ______

5.53.2 NOTIFY the Unit Supervisor/Shift Manager the release is secured. ______

5.53.3 NOTIFY Chemistry the release is secured. ______

5.53.4 GO TO Step 5.55. ______

5.54 WHEN the release is finished, THEN PLACE “1(2)WL-FV-4077 WASTE MONITOR TANK PUMPS 1A, B & C(2A, B & C) DISCHARGE DIVERT VALVE” in RECIRC position. (Radwaste Control Rm on ZLP-189) ______

5.55 RECORD date and time the release was terminated on the Procedure Performance Data Sheet. ______

5.56 SECURE recirc of WMT by performing the applicable step below:

5.56.1 WMT 1A(2A):

5.56.1.1 PLACE “1(2)WL-FV-4073 WASTE MONITOR TANK PUMP 1A(2A) DISCHARGE/RECIRC FLOW CONTROL” in RECIRC.
(Radwaste Control Rm on ZLP-189) ______

5.56.1.2 PLACE “1(2)-WL-HS-4070 WMT Pump 1A(2A) HANDSWITCH” in STOP.
(Radwaste Control Rm on ZLP-189) ______

5.56.2 WMT 1B(2B):

5.56.2.1 PLACE “1(2)WL-FV-4065 WASTE MONITOR TANK PUMP 1B(2B) DISCHARGE/RECIRC FLOW CONTROL VALVE” in RECIRC
(Radwaste Control Rm on ZLP-189) ______

5.56.2.2 PLACE “1(2)-WL-HS-4062 WMT Pump 1B(2B) HANDSWITCH” in STOP.
(Radwaste Control Rm on ZLP-189) ______

Exam Bank No.: 3171 **Last used on an NRC exam:** Never

RO Sequence Number: 64

Which of the following are tasks assigned to the Reactor Operators during refueling operations in accordance with 0POP08-FH-0009, Core Refueling?

1. Evaluate Inverse Count Rate Ratio (ICRR) data on loaded fuel assemblies.
2. Monitor the Core Monitoring NI channels during and following insertion of each fuel assembly.
3. Record the operating RHR loop inlet temperature every 12 hours.
4. Monitor the remote camera monitors used to observe refueling activities.

A. 1 and 4

B. 1 and 2

C. 2 and 3

D. 3 and 4

Answer: C 2 and 3

Exam Bank No.: 3171 **Source:** Modified **Modified from** 1101

K/A Catalog Number: G2.1.2 Knowledge of operator responsibilities during all modes of plant operation.

RO Importance: 4.1 **Tier:** 3 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 201.43 **Objective Number:** 32271

DESCRIBE the procedural requirements of the fuel handling equipment operating procedure(s) to include purpose, scope, precautions and limitations.

Reference: 0POP08-FH-0009

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as item #1 ICRR data is collected and evaluated but by Reactor Engineering and item #4 the camera monitors are available to the operator but it is not an assigned task to continuously monitor. Note that the RO is trained to perform ICRRs and evaluated during Startup Certification.

B: INCORRECT. Plausible as Item #2 is an assigned task and item #1 ICRR data is collected and evaluated but by Reactor Engineering. but Item #1 is not as noted in the explanation for Choice ‘A’ above

C: CORRECT. Per 0POP08-FH-0009, Core Refueling two of the tasks assigned to Control Room personnel include: 1) Monitoring NI channels during and following insertion of each fuel assembly (Step 6.2.10.3) and 2) Record the operating RHR loop inlet temperature and boron concentration every 12 hours (Steps 6.1.2 and 6.2.2).

D: INCORRECT. Plausible as Items #3 is an assigned task and and item #4 the camera monitors are available to the operator but it is not an assigned task to continuously monitor.

Question Level: F **Question Difficulty** 2

Justification:

The student must be able to recall the administrative responsibilities for Control Room personnel during Refueling operations.

Source For RO #64

STP LOT-24 NRC EXAM Print Date 7/28/2022

Exam Bank No.: 1101 Last used on an NRC exam: 2020

In accordance with 0POP08-FH-0009, Core Refueling, which of the following is a task for a Licensed Control Room Operator during refueling operations?

A. Inform the Core Load Supervisor of the next core location to have a fuel assembly loaded.

B. Operate the remote television monitoring equipment used to observe refueling activities.

C. Monitor the Core Monitoring NI channels during and following insertion of each fuel assembly.

D. Evaluate Inverse Count Rate Ratio (ICRR) data on loaded fuel assemblies.

Answer: C Monitor the Core Monitoring NI channels during and following insertion of each fuel assembly.

Exam Bank No.: 1101 **Source:** Bank **Modified from:**

K/A Catalog Number: G2.1.40 Knowledge of refueling administrative reqirements.

RO/SRO Importance: 2.8 / **Tier:** 3 **Group/Category:**

RO-10CFR55.41 # 10 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 201.43 **Objective Number:** 66407

DESCRIBE the procedural requirements of the fuel handling equipment operating procedure(s) to include purpose, scope, precautions and limitations

Reference: 0POP08-FH-0009, Core Refueling, Step 6.2.10.3.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because move sheets are available to and monitored by the RO during fuel movement, but not an assigned responsibility.

B: INCORRECT: Plausible because the equipment is available to the operator, but not an assigned responsibility

C: CORRECT: per 0POP08-FH-0009, Core Refueling

D: INCORRECT: Plausible because ICRR data is collected, but by Reactor Engineering in Step 6.2.10.4. NOTE: Reactor Operator is required to evaluate/perform an ICRR per the Task Analysis of LOT Training during their reactor startup certification..

Level F **Difficulty** 2

Justification:

The student must be able to recall the administrative responsibilities for Licensed Operators during refueling.

6.0 Procedure

NOTE

- All fuel handling machinery (cranes, tools, etc.) SHALL be operated per the appropriate procedure.
- Fuel Transfer Forms (FTFs) SHALL be In Hand at the Controlling Station AND copies of FTFs SHALL be available at each refueling station.
- The Core Loading Supervisor may allow Fuel movement to continue while revised FTFs are being transferred between refueling stations.

6.1 Core Offload

6.1.1 VERIFY Prerequisites Checklist (Form 1) has been completed for core offload.

6.1.2 NOTIFY Control Room personnel to RECORD the operating RHR Loop inlet temperature and boron concentration every 12 hours on Form 2.

6.1.3 NOTIFY Control Room personnel or Chemistry to RECORD the Refueling Canal boron concentration on Form 3. Surveillance sample frequency is 72 hours. An administrative sample frequency of 48 hrs. + 6 hours (from 48 to 54 hrs.) has been established to ensure Technical Specification 4.9.1.2 requirements are not exceeded.

6.1.4 ENSURE an approved Fuel Transfer Form (FTF) prepared per 0PEP02-ZM-0005, “Fuel Transfer Form,” is followed during core offload.

6.1.5 NOTIFY Control Room personnel to track the temporary and final locations of ALL fuel assemblies and inserts for each step of the core offload.

6.1.6 NOTIFY the Health Physics Supervisor prior to transferring the first irradiated fuel assembly through the transfer tube and ENSURE Health Physics concurs that fuel transfer may occur without radiological incident.

6.2 Core Reload

6.2.1 VERIFY Prerequisite Checklist (Form 1) has been completed for core reload.

6.2.2 NOTIFY Control Room personnel to record the operating RHR Loop inlet temperature and boron concentration every 12 hours on Form 2.

6.2.3 NOTIFY Control Room personnel or Chemistry to RECORD the Refueling Canal boron concentration on Form 3. Surveillance sample frequency is 72 hours. An administrative sample frequency of 48 hrs. + 6 hours (from 48 to 54 hrs.) has been established to ensure Technical Specification 4.9.1.2 requirements are not exceeded.

6.2.4 ENSURE an approved Fuel Transfer Form (FTF) prepared per 0PEP02-ZM-0005, “Fuel Transfer Form,” is followed during core reload.

6.2.5 NOTIFY the Health Physics Supervisor prior to transferring the first irradiated fuel assembly through the transfer tube and ENSURE Health Physics concurs that fuel transfer may occur without radiological incident.

6.2.6 NOTIFY Control Room personnel to track the temporary and final locations of ALL fuel assemblies and inserts for each step of the core reload.

6.2.7 ENSURE fuel assemblies are transferred from their specified locations on the FTF to the final locations specified on the FTF as directed by the Core Loading Supervisor.

6.2.8 RECORD the date and time the transfer was completed and INITIAL the Fuel Transfer Form (FTF).

6.2.9 <u>WHEN</u> moving a fuel assembly from the FHB to the RCB, <u>THEN</u> PERFORM the following:

6.2.9.1 NOTIFY personnel in the FHB to verify the fuel assembly number prior to transferring the fuel assembly to the RCB. (Reference 7.29)

6.2.9.2 NOTIFY personnel in the FHB to verify the insert type (T-Bar, Hub Mounted, or None).

6.2.9.3 <u>IF</u> the insert is a BPRA, <u>THEN</u> NOTIFY personnel in the FHB to verify the insert number.

6.2.9.4 INITIAL the “Numbers Verified” block on the FTF.

6.2.9.5 NOTIFY personnel in the FHB to initiate the transfer of the assembly to the RCB.

6.2.10 WHEN moving a fuel assembly in the RCB to a core location, THEN PERFORM the following:

6.2.10.1 Prior to lowering a fuel assembly over the core, VERIFY that the Reactor Engineer has completed all required count data from the previously loaded fuel assembly.

6.2.10.2 VERIFY fuel assembly will not become entangled with the shoehorn cabling prior to lowering the fuel assembly in the core.

CAUTION

The Refueling Machine Operator SHALL NOT disengage from the fuel assembly until nuclear instrumentation indicates stable conditions and approval to disengage is received from the Core Loading Supervisor.

6.2.10.3 NOTIFY a Licensed Control Room Operator to monitor the Core Monitoring NI channels during and following the insertion of each fuel assembly into the core AND the Core Loading Supervisor SHALL be notified of the stability or instability of the count rate.

a. IF the count rate is unstable, THEN DIRECT the Refueling Machine Operator to WITHDRAW the fuel assembly AND suspend core alterations.

b. IF the count rate is stable, THEN DIRECT the Refueling Machine Operator to unlatch the fuel assembly.

6.2.10.4 NOTIFY the Reactor Engineer to obtain count rate data from the Core Monitoring NI channels by following the instructions in Addendum 2.

6.2.11 When core loading is complete, logging on Forms 2 and 3 should be suspended.

6.2.12 WHEN refueling is complete, THEN PERFORM a Level 1 Inventory of the Reactor Vessel and core alignment check per 0PEP02-ZM-0004, “Physical Inventory of Fuel Assemblies”.

Exam Bank No.: 3172

Last used on an NRC exam: Never

RO Sequence Number: 65

Which of the following would be a violation of work hour limits in accordance with 0PGP03-ZA-0114, Fatigue Rule Program? (Consider each case separately)

A. Working from 0600 to 2200 in a single shift.

B. Working from 0600 to 2000 on two consecutive days.

C. Working six 12 hour shifts in a 7 day period.

D. Working 1800 to 1000 and then 2000 to 0600 the following shift.

Answer: B Working from 0600 to 2000 on two consecutive days.

Exam Bank No.: 3172 **Source:** New **Modified from** N/A

K/A Catalog Number: G2.1.5 Ability to use procedures related to shift staffing, such as minimum shift complement, overtime limitations, etc.

RO Importance: 2.9 **Tier:** 3 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrative procedure, DISCUSS the requirements associated with the referenced procedure.

Reference: 0PGP03-ZA-0114, pg .17

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because this is right at the actual work hour limit of 16 hours but would not be a violation.

B: CORRECT. Each day would be 14 hours resulting in 28 hours worked in a 48 hour period exceeding the limit of 26 hours.

C: INCORRECT. Plausible because this would be 72 hours but this would still meet the 7 day limit..

D: INCORRECT. Plausible because this is right at the limit for both time in 48 hours (26 hrs) and break between shifts (10 hrs).

Question Level: H **Question Difficulty** 3

Justification:

The student evaluate each condition and compare to the working hour limits.

4.3 Managing Hours Worked

4.3.1 Work Hour Limits

4.3.1.1 The following limits apply to covered workers regardless of unit status:

a. Work Hour Ceilings:

1. No more than 16 work hours in any 24-hour period.
2. No more than 26 work hours in any 48-hour period.
3. No more than 72 work hours in any 7-day period.
4. Work Hour Breaks:
5. At least a 10-hour break between successive work periods, or an 8-hour break when a break of less than 10 hours is necessary to accommodate a crew's scheduled transition between work schedules or shifts.
6. A 34-hour break in any 9-calendar day period.

4.3.1.2 Average work week scheduling:

NOTE

Exceeding the 54 hour per week average is NOT a violation of the Fatigue Rule. A waiver is NOT required to exceed 54 hour threshold.

a. The Fatigue Rule establishes an average work week length of 54 hours as a threshold for potential cumulative fatigue.

b. Covered Worker on-line scheduling should limit work hours to 54 hours per week averaged over the established shift cycle.

c. IF a Covered Worker exceeds the 54 hour criteria, THEN a Condition Report SHALL be generated with actions to prevent reoccurrence.

d. IF schedule additions cause a covered worker to exceed the 54 hour criteria, THEN the Supervisor should evaluate opportunities to reduce work hours elsewhere in the schedule to keep from exceeding the 54 hour average work week length.

Exam Bank No.: 3210 **Last used on an NRC exam:** Never

RO Sequence Number: 66

In accordance with 0PGP03-ZO-0055, Protected Components, which entry into posted Protected Equipment areas is allowed WITHOUT the permission of the Shift Manager/Unit Supervisor on duty?

A. A Performance Technician performing non-intrusive thermography.

B. A Plant Operator performing normal operator rounds.

C. A Chemistry Technician collecting non-protected system samples.

D. The Plant General Manager performing an inspection tour.

Answer: B A Plant Operator performing normal operator rounds.

Exam Bank No.: 3210 **Source:** New **Modified from** N/A

K/A Catalog Number: G2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordinator with the transmission operator.

RO Importance: 2.6 **Tier:** 3 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 507.01 **Objective Number:** 32815

Given the title of an administrative procedure, IDENTIFY the objectives, functions, purpose, scope, and definitions of that procedure.

Reference: 0PGP03-ZO-0055 pgs 13,14

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the Performance Technicians are part of the Operations Department but are not watchstanders. Step 7.4.2 applies.

B: CORRECT. Per 0PGP03-ZO-0055 step 7.5 implementation of Emergency or Abnormal Response procedures shall not be impeded by postings directed by the procedure.

C: INCORRECT. Plausible because samples on non-protected systems should not affect a Protected system, however step 7.4.3 applies and a discussion must be had with the SM/US and a determination made that that the work is non-intrusive and does not impact the availability of Protected Equipment.

D: INCORRECT. Plausible because the Plant General Manager outranks the SM/US but step 7.4.5 applies and they are not reponding to an emergency or meet the the requirements of step 7.4.6.

Question Level: F **Question Difficulty** 2

Justification:

Student must know protected equipment responsibilities.

7.0 Requirements for Entering Areas with Protected Component Posting

7.1 Access to protected equipment or areas that contain protected equipment is tightly controlled.

7.2 Work on protected equipment or behind protected component postings is prohibited(8.25).

7.3 Limit the transport of tools, equipment, and materials near protected equipment to minimize the potential for equipment to be actuated or disabled because of inadvertent bumping.

7.4 The following criteria shall be applied to protected equipment area access:

7.4.1 Operations Department personnel may enter protected equipment areas as part of normal operator rounds. Operators on rounds in these areas shall ensure inappropriate or unauthorized work is NOT being performed.

7.4.2 Operations department personnel may also enter Protected Equipment areas as part of other normal operator duties after discussion of the applicable Protected Equipment scheme with Shift Manager/ Unit Supervisor and it is determined that the duties are non-intrusive and do not impact the availability of the Protected Equipment.

7.4.3 Chemistry department personnel may also enter Protected Equipment areas for ***non-protected*** system sampling and meter reading only following discussion of the applicable Protected Equipment scheme with Shift Manager/ Unit Supervisor and it is determined that the work is non-intrusive and does not impact the availability of the Protected Equipment.

7.4.4 Security personnel on normal rounds and roving fire watches on duty may enter protected equipment areas to conduct normal assigned rounds once they have been trained to recognize protected equipment in the field, understand the vulnerabilities inadvertent operation presents to the plant, and understand the need to avoid bumps and radio transmissions in sensitive areas

7.4.5 Other personnel are NOT to enter posted protected equipment areas without specific permission from the Shift Manager/Unit Supervisor on duty unless they are responding to an emergency, meet the requirements of 7.4.6 or are transiting the area via a Safe Transit Path that is clearly identified.

7.4.6 General access to posted protected areas shall be deferred until a time when the area is no longer protected. Exceptions to this rule are:

7.4.6.1 Emergency Medical Response Teams

7.4.6.2 Fire Brigade response

7.4.6.3 NRC inspections

7.4.6.4 Switchyard Coordinator rounds in the switchyard

7.5 Implementation of Emergency or Abnormal response procedures shall not be impeded by postings directed by this procedure.

Exam Bank No.: 2848 **Last used on an NRC exam:** 2019

RO Sequence Number: 67

Unit 1 is at 100% power.

- A material deficiency is causing an annunciator to repeatedly alarm.
- The RO requests to remove the ICS Alarm from ALARM CHECKING.

Per 0PGP03-ZO-0039, Operations Configuration Management, to accomplish this task the crew should enter the associated ICS point in the _____(1)_____ and direct _____(2)_____ to review the status change within 36 hours.

A. (1) OAS Log
(2) I&C Maintenance

B. (1) Points Off-Scan Log
(2) I&C Maintenance

C. (1) OAS Log
(2) Engineering

D. (1) Points Off-Scan Log
(2) Engineering

Answer: D (1) Points Off-Scan Log (2) Engineering

Exam Bank No.: 2848 **Source:** Bank **Modified from** N/A

K/A Catalog Number: G2.2.43 Knowledge of the process used to track inoperable alarms.

RO Importance: 3.0 **Tier:** 3 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 507.01 **Objective Number:** 92183

Given the title of an administrative procedure, IDENTIFY the individuals (by job title) with specific responsibilities in the procedure.

Reference: 0PGP03-ZO-0039, Operations Configuration Mangement, Section 4.5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because the OAS tracks taking equipment out of and returning to service but it is not used to track non-TS annunciators and I&C does perform maintenance on faulty alarms but is not the correct group for review.

B: INCORRECT: Plausible as this is the correct log and I&C does perform maintenance on faulty alarms but is not the correct group for review.

C: INCORRECT: Plausible because the OAS tracks taking equipment out of and returning to service but it is not used to track non-TS annunciators. This is the correct group.

D: CORRECT: The Points Off-Scan Log is used and Engineering must review within 36 hours, in accordance with 0PGP03-ZO-0039, Operations Configuration Management..

Question Level: F **Question Difficulty** 2

Justification:

The student must have knowledge of the process to remove an annunciator from service.

NOTE

Section 4.5 does **NOT** apply to chemistry related Plant Computer points controlled by 0PCP01-ZA-0040, Process Instrumentation Setpoint Control.

4.5 Plant Computer Point Program

4.5.1 Shift Manager/Unit Supervisor permission is required to place a computer point in an off-scan condition, remove from alarm checking, or install an entered value.

4.5.2 In Modes 1, 2 and 3, all Plant Computer points being tracked in the Points Off-Scan Log SHALL be identified on a Condition Report. The CR number is required to be entered on the Points Off-Scan Log for the associated Plant Computer point (See Step 4.5.6.).

NOTE

0POP03-ZG-0001, Plant Heatup, ensures Plant Computer Alarm pages are satisfactory for Mode 3 operation prior to the transition from Mode 4.

4.5.3 In Modes 4, 5, 6 or Defueled, **ONLY** Plant Computer points placed in an off-scan condition, removed from alarm checking, or with an entered value installed due to a material deficiency shall be tracked in the Points Off-Scan Log. These Points Off-Scan Log entries due to material deficiencies also require Condition Report initiation and CR number entry in the Points Off-Scan Log (see Step 4.5.6). Non-material deficiency related Plant Computer points in an off-scan condition shall be tracked by printing out a Plant Computer Point Off-Scan Report (see Addendum 1) at least once per day. IF any of the non-material deficiency related Plant Computer points are still in an off-scan condition when entry into Mode 3 is made, THEN the CR tracking requirements of Step 4.5.2 will apply.

4.5.4 The condition report tag for any Plant Computer point in an off-scan condition, removed from alarm checking, or has an entered value SHALL be placed in the Points Off-Scan Log.

4.5.5 ALL Plant Computer points being tracked due to material deficiency in the Points Off-Scan Log SHALL be reviewed within 36 hours from the time the Computer Point was taken off-scan, removed from alarm checking, or had substitute value entered. IF another source for this computer point is available, THEN the review period may be extended to 72 hours.

4.5.5.1 Engineering Duty Manager SHALL coordinate the Computer Point Off-Scan review when notified of the following by Operations:

- CR number documenting material deficiency
- Off-Scan Computer Point requiring review
- Time Computer Point taken off-scan
- Off-Scan review due date and time

NOTE

Once a computer point is taken off scan, Engineering has 36 hours from that time to review the point(s).

4.5.5.2 CR number and Engineering contact SHALL be recorded in Station Log Book. Include in the log entry the Engineer who was contacted concerning the off scan computer point.

4.5.5.3 Reviewer SHALL perform the following:

- Document review in CR action.
- Ensure review results include the following:
    - What Computer Point monitors
    - Risk of Computer Point being off-scan
    - Compensatory measures, if required
- Notify Control Room(s) of results when review is complete.

Exam Bank No.: 3226 **Last used on an NRC exam:** Never

RO Sequence Number: 68

Per 0PGP03-ZR-0050, Radiation Protection Program, the Administrative Action Level (AAL) for TEDE is ___(1)___ Rem.

At a MINIMUM, _______(2)______ approval is required to extend this administrative limit by an additional 500 mRem.

A. (1) 4
(2) Radiation Protection Manager

B. (1) 4
(2) Site Vice President

C. (1) 2
(2) Radiation Protection Manager

D. (1) 2
(2) Site Vice President

Answer: C (1) 2; (2) Radiation Protection Manager

Exam Bank No.: 3226 **Source:** New **Modified from** N/A

K/A Catalog Number: G2.3.4 Knowledge of radiation exposure limits under normal or abnormal condtions.

RO Importance: 3.2 **Tier:** 3 **Group/Category:** 3 **10CFR Reference:** 55.41(b)(12)

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrtatiove procedure, DISCUSS the requirements sassociated with the referenced procedure.

Reference: 0PGP03-ZR-0050

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because 4 Rem is the third level of extensiion for the AALs at STP. The second part is correct.

B: INCORRECT. Plausible because 4 Rem is the third level of extensiion for the AALs at STP and Site VP approval is required for exceeding 3 or 4 Rem.

C: CORRECT. Per 0PGP03-ZR-0050, the admin limit for TEDE is 2 Rem and the minimum approval required is the Radiation Protection Manager.

D: INCORRECT. Plausible becauseite VP approval is required for exceeding 3 or 4 Rem.. The first part is correct.

Question Level: F **Question Difficulty** 2

Justification:

The student must recall administrative limits for dose and the requirements for extending the limits.

6.5 Administrative Action Levels (AAL) for Individual Radiation Dose (ANI EC8.4, INPO 05-008)

NOTE

Because minors are not employed by STP, the AALs discussed herein assume an adult worker and do not apply to minors.

6.5.1 The AAL on annual TEDE is 2 rem. Actions should be taken to ensure awareness of individuals approaching this AAL. (10CFR20.2104, INPO 05-008)

6.5.2 The AAL for Shallow Dose Equivalent (whole body skin dose) and extremity dose is 40 rem per year. Approval from the Radiation Protection Manager SHALL be required to exceed 40 rem per year, limited by the 50 rem per year limit of 10CFR20. (10CFR20.1201(a))

6.5.3 The AAL for lens of eye dose is 12 rem per year. Dose to lens of eye is limited to 15 rem per year by regulation. (10CFR20.1201(a))

6.5.4 The skin, extremity, and total organ AAL is 40 rem per year.

6.5.5 <u>IF</u> the current year (year-to-date) dose is greater than 2 rem or expected to exceed 2 rem, <u>THEN</u> a Personnel Dose Extension Authorization form (Form 2) should be completed prior to setting an AAL in the computer.

6.5.6 Year-to-date dose history, including eye dose, SHALL be determined prior to granting a larger AAL. (10CFR20.2104)

6.5.7 The following signatures are required as a minimum on Form 2.

6.5.7.1 The individual to whom the extension applies.

6.5.7.2 The responsible work group supervisor

6.5.7.3 The Radiation Protection Manager.

6.5.8 Additional management approvals MAY be required for dose extensions as indicated on Form 2 as necessary.

6.5.9 In order to limit the dose to the embryo/fetus of a declared pregnant woman as defined in 10CFR20.1003 the following actions SHALL be taken:

6.5.9.1 The declared pregnant woman SHALL complete section 1 of Form 1 and deliver Form 1 to Dosimetry.

6.5.9.2 Dosimetry SHALL collect the declared pregnant woman's current DLR AND restrict the declared pregnant woman's RCA access. Collected DLR SHOULD be processed IAW site procedures.

File:U28/D43 PDRP DTL 6935

Form 2 (Rev 4)	Personnel Dose Extension Authorization	Page 1 of 1

Section I: Extension Request (Completed by Work Group Supervisor)

Name: ______________________________
Last First MI

ID/SSN: ______________________________

I am requesting the individual named above, current year TEDE be extended to: ______________ mrem

Reason for Dose Extension______________________________

____

Section II: Current Year Dose Summary (Completed by RP)

TEDE (Record)	________	mrem	*Absent Record or Worker Dose Estimate	☐ No ☐ Yes	If yes, do not approve
TEDE (SRD Estimate only)	________	mrem	****TEDE - Extended to**	________	**mrem**
TEDE (Total)	________	mrem			

Section III: Acknowledgement of Dose Extension (Completed by worker being extended)

I have reviewed and agree with my current year dose summary in Section II.

Signature: ______________________________ Date: ______________

Section IV: Approvals for Dose Extension (IV (a) approvals required for ALL extensions).

IV (a) ☐ Exceed 2,000 mrem TEDE	IV (b) ☐ Exceed 3,000 mrem TEDE ☐ Exceed 4,000 mrem TEDE
Work Group Supervisor: ______________ Date: ________	Station/Plant Manager: ______________ Date: ________
RP Manager: ______________ Date: ________	Site Vice President: ______________ Date: ________

Section V: Data Entry (Completed by RP)

Data Entry by: ______________________________ / Date: ______________
Verified By: ______________________________ / Date: ______________

* Individuals with absent / unobtainable current year dose or worker dose estimates may not be extended beyond 2000 mrem.

** Extensions beyond 4,000 mrem, should not have significant SRD dose estimate

** Extensions are only applicable to this licensee or fleet licensees sharing a common database.

** Dose extensions expire at year end or when occupational monitoring is ended.

When completed, this form SHALL be retained in accordance with the Document Type List (DTL).

Exam Bank No.: 2946 **Last used on an NRC exam:** 2020

RO Sequence Number: 69

To verify an alarm is consistent with plant conditions requires that the RO validate the alarm by ________(1)________.

This requirement is in accordance with ___________(2)___________.

A. (1) verifying the ICS point that drives the alarm is in the proper state
(2) Conduct of Operations Chapter 2, Shift Operating Practices

B. (1) verifying the ICS point that drives the alarm is in the proper state
(2) 0PGP03-ZA-0010, Performing and Verifying Station Activities

C. (1) comparing it with other indications
(2) Conduct of Operations Chapter 2, Shift Operating Practices

D. (1) comparing it with other indications
(2) 0PGP03-ZA-0010, Performing and Verifying Station Activities

Answer: C (1) comparing it with other indications (2) Conduct of Operations Chapter 2, Shift Operating Practices

Exam Bank No.: 2946 **Source:** Bank **Modified from** N/A

K/A Catalog Number: G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

RO Importance: 4.2 **Tier:** 3 **Group/Category:** 4 **10CFR Reference:** 55.41(b)(10)

STP Lesson: LOT 505.01 **Objective Number:** 92107

Discuss automatic actions expected to occur on entry conditions for the reference procedure.

Reference: Conduct of Operations, Chapter 2, Step 13.1.7

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as the Conduct of Operations is used for annunciator response and many alarms are driven from ICS but it is not the required actions to validate the alarm.

B: INCORRECT: Plausible as 0PGP03-ZA-0010 does contain requirements for operator activities and many alarms are driven from ICS but it is not the required actions to validate the alarm.

C: CORRECT: When an alarm is received, the Conduct of Operations requires the RO to validate the alarm by comparing it with other indications.

D: INCORRECT: Plausible as 0PGP03-ZA-0010 does contain requirements for operator activities. This is the correct method to validate an alarm.

Question Level: F **Question Difficulty** 2

Justification:

The student must recall information from the Conduct of Operations.

additional alarms when acknowledging an alarm.

- A Reactor Operator will address another member of the Control Room Staff, THEN announce the alarm title.
- IF the alarm is expected due to current operations, surveillance testing, alarm testing, material condition or Maintenance Activities, THEN "EXPECTED" will be announced after the alarm title and the reason for the alarm.
- PRIOR TO conducting an evolution or direct operator action, expected alarms should be identified and discussed. IF the action results in alarms that were not identified, THEN they should be responded to as unexpected alarms. (Expectations during tasks and evolutions)
- For all expected alarms, a member of the control room staff SHALL review or have previously reviewed during the shift the associated annunciator response procedure to verify alarm input validity and perform required alarm response actions. IF the Control Staff is given an "Unreliable Sheet" during maintenance/surveillance testing that states what the unreliable annunciators are, THEN the Unit Supervisor/Shift Manager will be briefed to ensure the appropriate alarm response actions are being taken.

13.1.2 IF the alarm condition requires entry into an off normal or emergency procedure THEN the review of the annunciator response procedure may be delayed or waived at the discretion of the Unit Supervisor.

13.1.3 For Plant Computer Alarms, the Operator SHALL determine the cause of the alarm and take appropriate corrective action.

13.1.4 The annunciator response procedures SHALL be utilized for local alarm panels for all alarm conditions by the plant operator responding to the alarm condition.

13.1.5 Referring to annunciator response procedures is not required for alarms caused by testing of alarm panels.

13.1.6 IF annunciator response procedures do NOT provide guidance to address an alarm condition, THEN the operator SHALL notify the Unit Supervisor/Shift Manager. The Unit Supervisor/Shift Manager will determine the specific course of action to address the alarm condition. The Unit Supervisor/Shift Manager SHALL ensure such actions do NOT adversely impact the safety of personnel or equipment. This condition and course of action SHALL be documented in the Control Room Logbook and condition report. (Refer to 0PGP03-ZA-0010, Verifying Station Activities for specific details.)

13.1.7 Alarms should be validated by comparison with other indications. IF an alarm is determined to be invalid, THEN the actions directed by an annunciator response procedure or an abnormal operating procedure need not be performed as determined by the Unit Supervisor/Shift Manager.

Exam Bank No.: 3174 **Last used on an NRC exam:** Never

RO Sequence Number: 70

Unit 1 is in Mode 3.

- RCS Average Temperature = 450ºF
- NO inoperable control rods
- RCS boron concentration = 1410 ppm
- Cycle Burnup is 8,000 MWD/MTU

Using the 0PSP10-ZG-0003, Shutdown Margin Verification - Modes 3, 4 and 5, Form 2 and Figure 5.5, Shutdown Margin Limit Curve provided answer the following:

C_B SDM Limit = ___(1)___ ppm.

Shutdown Margin ___(2)___ met.

(References provided)

A. (1) 1396
(2) is

B. (1) 1396
(2) is NOT

C. (1) 1529
(2) is

D. (1) 1529
(2) is NOT

Answer: A (1) 1396 (2) is

Exam Bank No.: 3174 **Source:** New **Modified from** N/A

K/A Catalog Number: 192002 K1.13 Calculate shutdown margin using procedures and given plant parameters.

RO Importance: 3.5 **Tier:** 4 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(1)

STP Lesson: LOT 110.01 **Objective Number:** 50209

Given data and procedure(s) DETERMINE if reactor shutdown margin is within procedural acceptance criteria.

Reference: 0PSP10-ZG-0003 pgs, 6, 7, 9, and 16, Plant Curve Book (PCB) Figure 5.5

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☑ **Attachment:** 0PSP10-ZG-0003 Form 2, Figure 5.5, U1C24 Shutdown Margin Limit Curve

Distractor Justification

A: CORRECT. Based on Figure 5.5 Cb SDM Limit is 1396 ppm. For the given conditions Cb Effective at 1410 ppm is greater than Cb SDM Limit and SDM is met.

B: INCORRECT. Plausible because the Cb SDM Limit is correct but the student could invert the Cb Effective and Cb SDM Limit numbers to determine that SDM is not met.

C: INCORRECT. Plausible because student could use the wrong curve for Cb SDM Limit and invert the Cb Effective and Cb SDM Limit numbers to determine that SDM is met.

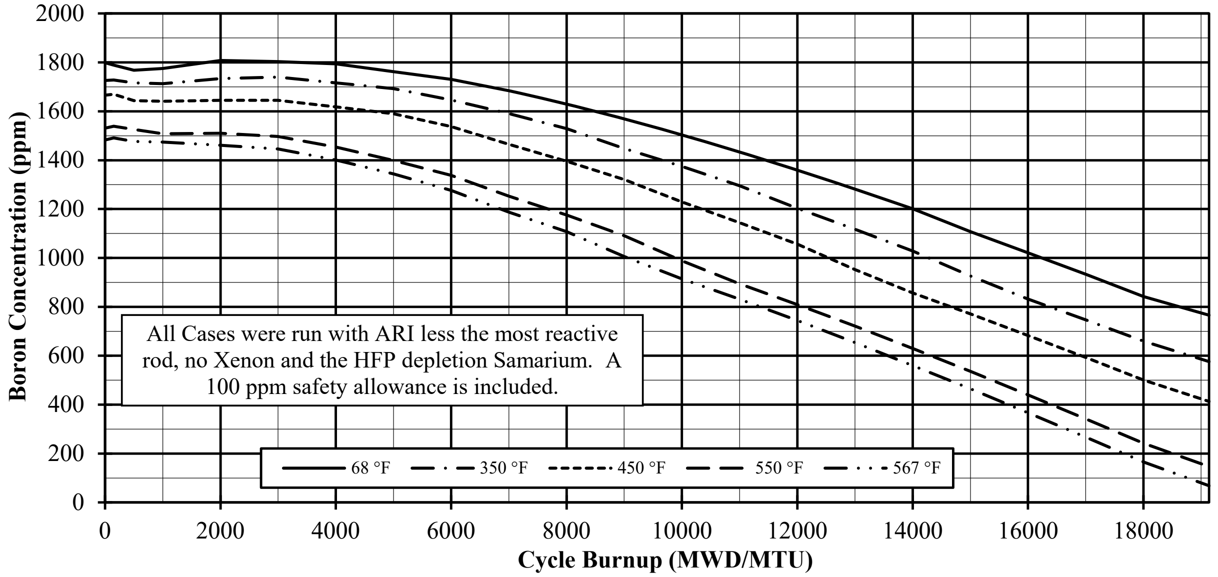
D: INCORRECT. Plausible because student could use the wrong curve for Cb SDM Limit and determine that SDM is not met

Question Level: H **Question Difficulty** 2

Justification:

The student must assess given conditions, interpret curves, and determine if it meets Technical Specification limits.

Figure 5.5
SHUTDOWN MARGIN LIMIT CURVE
Unit 1 Cycle 24 (Modes 3, 4, and 5)
(Source: A41009--00656UB, Rev 0)



All Cases were run with ARI less the most reactive rod, no Xenon and the HFP depletion Samarium. A 100 ppm safety allowance is included.

Burnup (MWD/MTU)	Boron Concentration (ppm)										
	68 °F	140 °F	200 °F	250 °F	300 °F	350 °F	400 °F	450 °F	500 °F	550 °F	567 °F
0	1799	1799	1799	1775	1751	1726	1696	1666	1599	1531	1483
150	1789	1789	1789	1769	1748	1728	1699	1670	1604	1539	1491
500	1768	1766	1766	1749	1733	1716	1680	1644	1585	1526	1477
1000	1775	1775	1775	1754	1734	1713	1677	1641	1575	1508	1474
2000	1808	1808	1808	1783	1759	1734	1689	1645	1577	1510	1461
3000	1803	1803	1803	1782	1761	1740	1692	1645	1571	1497	1446
4000	1794	1794	1794	1768	1742	1716	1667	1618	1536	1454	1401
5000	1762	1762	1762	1739	1716	1693	1641	1590	1495	1399	1344
6000	1731	1731	1731	1703	1674	1646	1592	1538	1438	1338	1277
7000	1684	1681	1681	1651	1621	1591	1528	1465	1359	1253	1187
8000	1629	1624	1624	1592	1560	1529	1462	1396	1286	1175	1107
9000	1569	1561	1561	1523	1486	1448	1385	1321	1206	1091	1006
10000	1504	1482	1482	1446	1411	1375	1302	1229	1108	988	914
11000	1434	1409	1409	1372	1334	1296	1220	1144	1019	894	831
12000	1359	1334	1333	1289	1246	1203	1129	1056	932	809	744
13000	1282	1254	1252	1207	1162	1117	1034	952	837	721	654
14000	1201	1172	1158	1115	1071	1028	942	857	743	630	560
15000	1107	1077	1072	1023	974	926	848	771	654	536	465
16000	1021	990	984	933	883	832	758	683	562	440	367
17000	933	901	884	838	792	747	670	593	468	342	267
18007	842	814	791	747	703	659	580	500	371	242	165
19139	766	737	713	667	622	576	494	413	280	147	68

Hot Rod Test C_B, ARO, 400°F to 567°F, K <= 0.98: **2257 ppm**	Mode 5 C_B, ARO, 68°F to 200°F, K <= 0.95: **2604 ppm**	Refueling C_B, ARO, 68°F to 140°F, K <= 0.95: **2582 ppm**

Prepared By: ______________________ Date: 10/24/21

Reviewed By: ______________________ Date: 10/24/21

Approved By: ______________________ Date: 10/25/21

Reactor & Fuel Engineering Supervisor

NOTE

An RCS boron concentration greater than 12 hours old SHALL not be used to verify SDM at current conditions.

5.3.4 RCS boron concentration (C_B RCS), sample date and sample time. A boron sample up to 12 hours old may be used provided the RCS was not diluted since the sample time. Initiate RCS boron sampling if required. _____

5.4 IF SDM is to be verified using the "Short Form SDM Limit Curve" method THEN proceed to Step 5.5. IF SDM is to be verified using the "Long Form SDM Limit Curve" method THEN proceed to Step 5.6. IF SDM is to be verified using the "0.95 Keff" method THEN proceed to Step 5.7. _____

5.5 "Short Form SDM Limit Curve" Method.

5.5.1 Verify the plant is in Mode 3, 4 or 5 and initial Form 2. __*__

5.5.2 IF the number of inoperable rods recorded per Step 5.3 is NOT zero (0), THEN determine the RCS boron concentration inoperable rod correction (C_B Rod) by completing Form 5. Record the value of C_B Rod on Form 2. _____

5.5.3 IF the number of inoperable rods recorded per Step 5.3 is zero (0), THEN record a value of zero (0) on Form 2 for the RCS boron concentration inoperable rod correction (C_B Rod). _____

5.5.4 Determine the effective boron concentration (C_B Effective) by the following expression. Record C_B Effective on Form 2. _____

C_B Effective = C_B RCS - C_B Rod

NOTE

- **DO NOT** interpolate Plant Curve Book Figure 5.5 for RCS average temperature. IF the RCS average temperature is between two of the supplied curves, THEN use the curve which provides the greater required boron concentration.
- A more conservative (more positive value) SDM Limit boron concentration (C_B SDM Limit) may be recorded on Form 2.
- Reference Precaution 3.4 for an example of incorporating a conservative philosophy to select RCS Average Temperature and determine the bounding SDM Limit.
- When recording the C_B SDM Limit; the RCS Average Temperature and the Cycle Burnup values recorded on Form 2 in Step 5.5.5.2 should be the actual values from the axes of the table for Plant Curve Book Figure 5.5 that correspond to that C_B SDM Limit.

5.5.5 Obtain the SHUTDOWN MARGIN Limit boron concentration (C_B SDM Limit) from Plant Curve Book Figure 5.5 at the conservative AND non-interpolated RCS Average Temperature and the conservative OR interpolated Cycle Burnup based on the data recorded in Step 5.3. _____

5.5.5.1 Record the SHUTDOWN MARGIN Limit boron concentration (C_B SDM Limit) on Form 2. _____

5.5.5.2 Record the RCS Average Temperature and Cycle Burnup used to determine the SHUTDOWN MARGIN Limit boron concentration (C_B SDM Limit) on Form 2. _____

5.5.6 Continue with Step 5.8. _____

5.6 Long Form SDM Limit Curve" Method.

5.6.1 Verify the plant is in Mode 3, 4 or 5 and initial Form 3. __*__

5.6.2 IF the number of inoperable rods recorded per Step 5.3 is NOT zero (0), THEN determine the RCS boron concentration inoperable rod correction (C_B Rod) by completing Form 5. Record the value of C_B Rod on Form 3. _____

5.6.3 IF the number of inoperable rods recorded per Step 5.3 is zero (0), THEN record a value of zero (0) on Form 3 for the RCS boron concentration inoperable rod correction (C_B Rod). _____

5.6.4 Determine the RCS boron concentration Xenon correction (C_B Xenon) per Addendum 1 and record the value on Form 3. _____

5.7.3 Obtain the **Mode 5 C_B, ARO, K ≤ 0.95** from the applicable unit and cycle Plant Curve Book Figure 5.5. Record the value on Form 4 as the SHUTDOWN MARGIN Limit boron concentration (C_B SDM Limit). _____

5.8 Compare the Effective boron concentration (C_B Effective) with the SHUTDOWN MARGIN Limit boron concentration (C_B SDM Limit). Record the results of the comparison on Form 2, Form 3 or Form 4. _____

5.9 Evaluate the test results against the acceptance criteria.

5.9.1 <u>IF</u> C_B Effective is greater than or equal to C_B SDM Limit, <u>THEN</u> adequate SHUTDOWN MARGIN is present. _____

5.9.2 <u>IF</u> C_B Effective is less than C_B SDM Limit, <u>THEN</u> adequate SHUTDOWN MARGIN is NOT present. _____

5.10 <u>IF</u> SDM was verified using the "Long Form SDM Limit Curve" method, complete Addendum 2 to determine the length of time that the SDM verification remains valid. Record the SDM Validity Time on Form 3. _____

5.11 The test coordinator SHALL sign completed Form 2, Form 3, Form 4, Form 5, or Form 6. *

5.12 Forward the test package to a verifier. _____

5.13 The verifier SHALL perform the following:

5.13.1 Independently verify all entries on completed Form 2, Form 3, Form 4, Form 5 and Form 6. _____

5.13.2 Verify any assumed conservative values on Form 2 or Form 3 are truly conservative. _____

5.13.3 Sign completed Form 2, Form 3, Form 4, Form 5 and Form 6 when the verification is complete. *

5.13.4 Forward the test package to the Test Coordinator. _____

5.14 The Test Coordinator SHALL ensure the names of all data takers and procedure performers are entered on Form 1. _____

5.15 Indicate the reason for performing this test on Form 1. _____

5.16 Evaluate the test results against the acceptance criteria. _____

5.17 Indicate the test results, sign, <u>AND</u> enter the date and time on Form 1. *

Form 2	Short Form SDM Limit Curve Method	Page 1 of 1

Unit ________ Cycle _________

5.5.1 Plant is in Mode 3, 4 or 5: ____________
Initial

5.5.2 C_B Rod = (+) ____________ ppm
5.5.3

5.5.4 C_B Effective = ____________ - _____________
C_B RCS C_B Rod

C_B Effective = _____________ ppm

5.5.5.1 C_B SDM Limit = _____________ ppm

5.5.5.2 RCS Average Temperature ___________ °F

Cycle Burnup ______________ MWD/MTU

5.8 Is C_B Effective $\geq$ C_B SDM Limit? ________ Yes _________ No

Completed By: ____ Date: _____________________

Verified By: ____ Date: _____________________

This Form, when complete, SHALL be retained for the life of the plant.

Exam Bank No.: 3209 **Last used on an NRC exam:** Never

RO Sequence Number: 71

Unit 1 is at 50%, EOL steady state.

- Rod Control is in MAN
- Turbine control is in INLET IN
- The RO withdraws rods for 5 seconds.

Actual reactor power will stabilize _____(1)_____ the initial power level and reactor coolant temperature will stabilize _____(2)_____ the initial temperature.

(Assume main turbine load remains constant and the reactor does not trip.)

A. (1) at
(2) at

B. (1) at
(2) above

C. (1) above
(2) at

D. (1) above
(2) above

Answer: B (1) at (2) above

Exam Bank No.: 3209 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 192005 K1.03 Predict direction of change in reactor power for a change in control rod position.

RO Importance: 3.5 **Tier:** 4 **Group/Category:** **10CFR Reference:** 55.41(b)(1)

STP Lesson: LOT 102.22 **Objective Number:** N99695

Predict direction of change in reactor power for a change in control rod position.

Reference: GP Reactor Theory Chapter 5 Control Rods NRC Exam Bank Questions Study Guide Pgs. 7,8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible that temperature ends up at the same poiint because the steam demand will change. First part is correct.

B: CORRECT. The rod withdrawal will add positive reactivity to the critical core causing a positive SUR and reactor power to rise. The subsequent increase in moderator anf fuel temperatures will add negative reactivity equal to the positive reactivity added by the rosd withdrawal. Final condition of the core will be reactor power at the same power, and RCS average temperature will be higher.

C: INCORRECT. Plausible if student thinks that the reactivity feedback effects of Doppler and moderator temperature turn the power but it stabilizes at the higher level and that temperature ends up at the same poiint because the steam demand will change.

D: INCORRECT. Plausible if student thinks that the reactivity feedback effects of Doppler and moderator temperature turn the power but it stabilizes at the higher level . The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

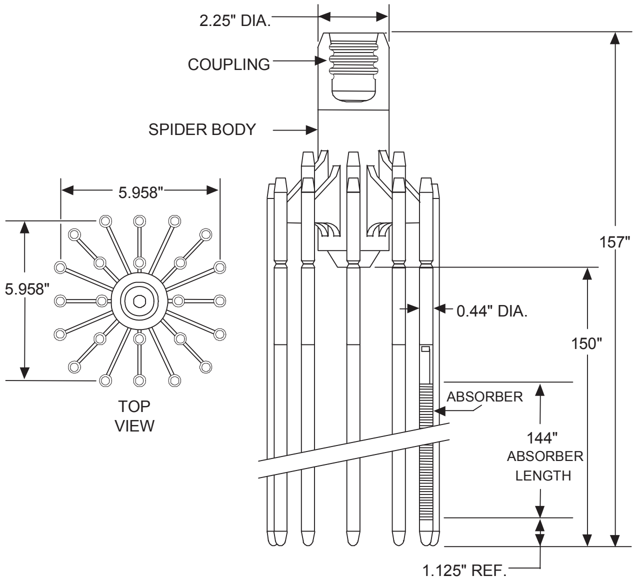
Student must assess the given conditions and determine the final status of the reactor.

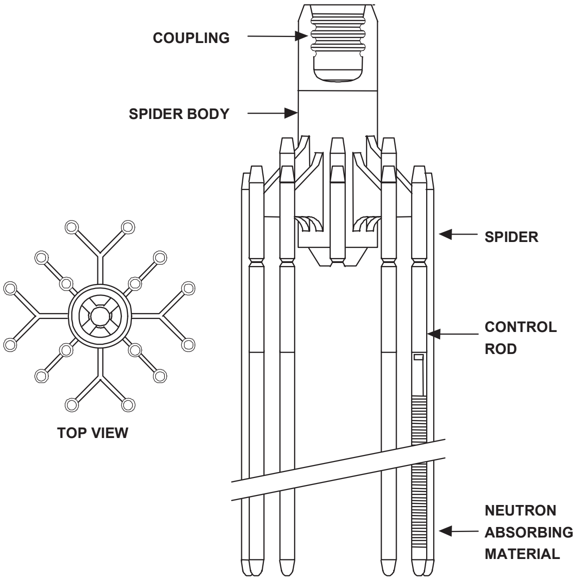
PWR Generic Fundamentals

REACTOR THEORY

CHAPTER 5

CONTROL RODS





NRC Exam Bank Questions Study Guide (thru Jun 2011)

Rev 4 (revised Jan 2012)

0 / 2 QID: P1054 Reactor power during change in control rod position	
BWR TOPIC: 292005	PWR TOPIC: 192005
KNOWLEDGE:	KNOWLEDGE: PK1.03 [3.5/3.6]
Similar questions	
None currently identified	
Recent BWR exams	Recent PWR exams
Not used in 2001 through 2011Jun	Sep2007, Mar2005

A reactor is operating at end of core life with a steady state 50% power level when the operator withdraws a group of control rods for 5 seconds. (Assume turbine load remains constant and the reactor does not scram/trip.)

Actual reactor power will stabilize ______________ the initial power and coolant temperature will stabilize ______________ the initial temperature.

A. at; at

B. at; above

C. above; at

D. above; above

0 / 2 QID: P1054 Reactor power during change in control rod position

Explanation

A reactor is operating at end of core life with a steady state 50% power level when the operator withdraws a group of control rods for 5 seconds. (Assume turbine load remains constant and the reactor does not scram/trip.) Actual reactor power will stabilize _AT_ the initial power and coolant temperature will stabilize _ABOVE_ the initial temperature.

The rod withdrawal will add positive reactivity to the critical core, causing a positive SUR (startup rate) to be developed and reactor power to rise. The subsequent increase in moderator and fuel temperatures will add negative reactivity equal to the positive reactivity added by the rod withdrawal. Final condition of the core will be reactor power at the same power level, and RCS (reactor coolant system) average temperature will be higher.

A. At (TRUE); at (FALSE)
   Incorrect - Even though actual reactor power does not change because steam demand did not change, temperature will go up to add the negative reactivity to compensate for the positive reactivity added by the initial rod withdrawal.

B. At (TRUE); above (TRUE)
   CORRECT - Temperature will go up to add the negative reactivity to compensate for the positive reactivity added by the initial rod withdrawal, but power will not change since steam demand has not changed.

C. Above (FALSE); at (FALSE)
   Incorrect - Reactor power is a function of steam demand. If it does not change, all that will change due to a reactivity change will be temperature. In this case since rods were withdrawn, temperature will increase, so both parts of this choice are incorrect.

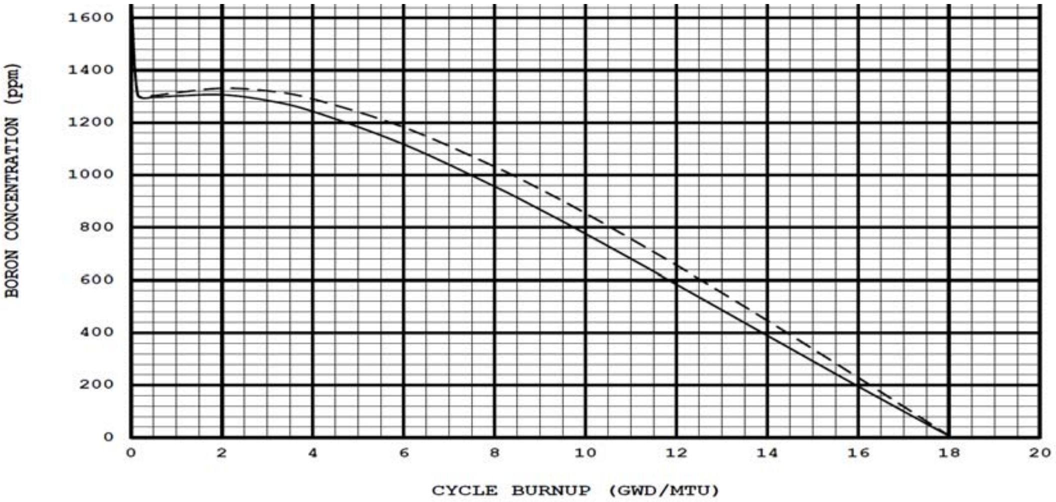
D. Above (FALSE); above (TRUE)
   Incorrect - Even though temperature will go up to add the negative reactivity to compensate for the positive reactivity added by the initial rod withdrawal, reactor power will not be higher (ABOVE) the original power level since steam demand has not changed.

The correct answer is ANSWER B. at; above

Exam Bank No.: 3176 **Last used on an NRC exam:** Never

RO Sequence Number: 72

Refer to the Critical Boron Concentration Curve below:



Between 0 and 0.1 GWD/MTU, ___(1)___ is responsible for the majority of the change in critical boron concentration.

Between 5 GWD/MTU and 10 GWD/MTU burnup, ___(2)___ will offset fuel depletion to maintain the reactor critical.

A. (1) fuel depletion
   (2) a reduction in boron concentration

B. (1) fuel depletion
   (2) burnable poison burnout

C. (1) fission product poison buildup
   (2) a reduction in boron concentration

D. (1) fission product poison buildup
   (2) burnable poison burnout

Answer: C (1) fission product poison buildup (2) a reduction in boron concentration

Exam Bank No.: 3176 **Source:** New **Modified from** N/A

K/A Catalog Number: 192007 K1.04 Describe how and why boron concentration changes over core life

RO Importance: 3.1 **Tier:** 4 **Group/Category:** 1 **10CFR Reference:** 55.41(b)(1)

STP Lesson: LOT 101.24 **Objective Number:** N99726

Describe how and why boron concentration changes over core life.

Reference: LOT 101.24 Power Point slides 26 & 30

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible if the student thinks the reason for the change in critical boron concentration is due to fuel depletion. The second part is correct.

B: INCORRECT. Plausible if the student thinks the reason for the change in critical boron concentration is due to fuel depletion and that burnable poison burnout compensates for fuel depletion to maintain the reactor critical as the core ages.

C: CORRECT. The initial rapid drop in critical boron concentration at the beginning of a core cycle is due to the buildup of fission product poisons. As the core ages to offset fuel depletion, boron concentration is reduced to maintain the reactor critical.

D: INCORRECT. Plausible as fission product buildup is the reason for the rapid initial drop in critical boron concentration and if the student thinks burnable poison burnout offsets fuel depletion as the core ages.

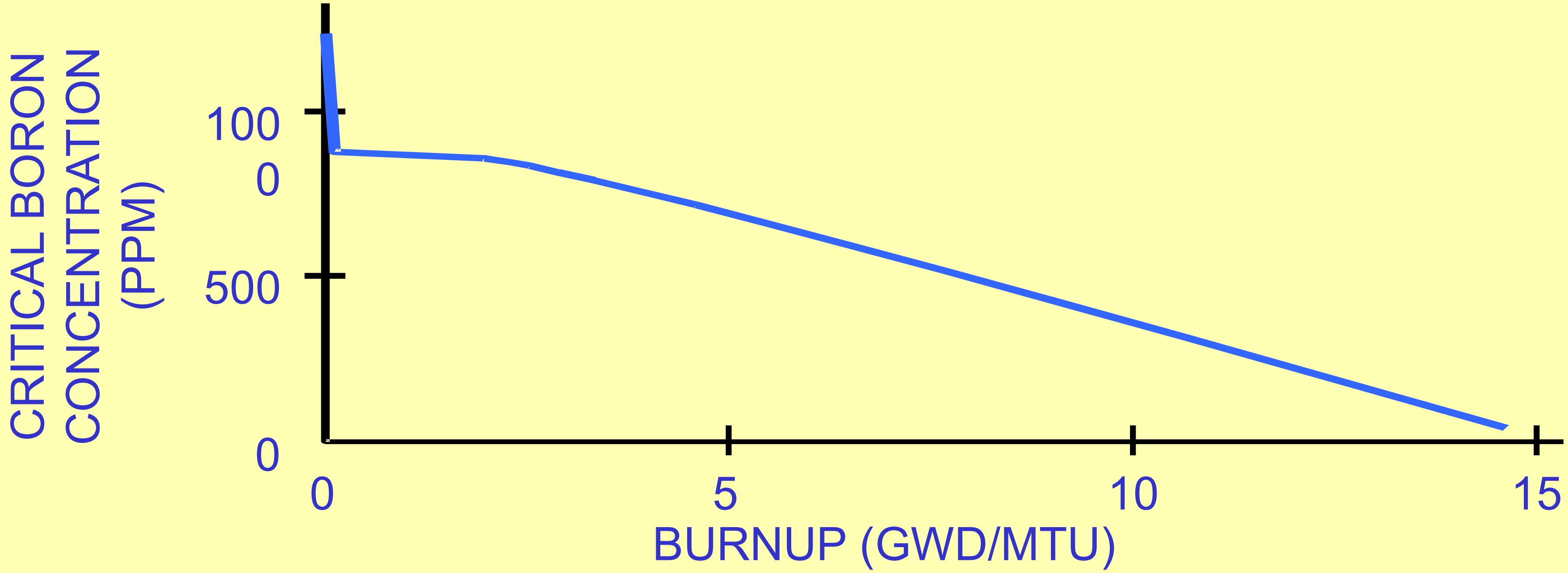
Question Level: H **Question Difficulty** 3

Justification:

The student must analyze the critical boron concentration curve based on given conditions to determine the correct reason for the boron concentration response.

EXAMPLE

Which one of the following is responsible for the majority of the rapid initial decrease in critical boron concentration?



a. Fuel depletion

b. Fission product buildup

c. Burnable poison burnout

d. Conversion of U-238 to Pu-239

Ex 7-8

EXAMPLE

As the core ages, the excess reactivity decreases due to fuel depletion. Over the long term, how is the reactor maintained critical as the fuel is depleted? Disregard coastdown conditions.

a. Withdrawing control rods

b. Reducing coolant temperatures

c. Reducing reactor power

d. Reducing boron concentration

Ex 7-12

Exam Bank No.: 3177 **Last used on an NRC exam:** Never

RO Sequence Number: 73

Unit 1 is at 100% power.

- Pressurizer level is on program.
- A leak occurs on the reference leg line of Pressurizer Level Transmitter LT-466.

The differential pressure sensed by LT-466 will **INITIALLY** ___(1)___ and LT-466 indicated level will ___(2)___.

A. (1) lower
   (2) lower

B. (1) lower
   (2) rise

C. (1) rise
   (2) lower

D. (1) rise
   (2) rise

Answer: B (1) lower (2) rise

Exam Bank No.: 3177 **Source:** New **Modified from** N/A

K/A Catalog Number: 193001 K1.03 Describe how pressure and level sensing instruments work

RO Importance: 2.6 **Tier:** 4 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(14)

STP Lesson: LOT 102.52 **Objective Number:** N99763

Describe how the following level sensing instruments operate. c. Closed vessel differential pressure level detectors

Reference: LOT 102.52 Student HO pgs. 19, 20

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because indicated level response for a wet reference leg vs. dry reference leg are commonly confused. The first part is correct, the differential pressure would lower as the reference leg drains.

B: CORRECT. The differential pressure detected by LT-466 would lower as the reference leg drains. In a Wet reference leg level system indicated level is inversely proportionally to D/P, so as the D/P lowers indicated level for LT-466 will rise.

C: NCORRECT. Plausible if the student confuses the high and low pressure sides for leak location and because indicated level response for a wet reference leg vs. dry reference leg are commonly confused.

D: INCORRECT. Plausible if the student confuses the high and low pressure sides for leak location. The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the given conditions and determine the failure effects on the wet reference leg level instrument.

Figure 1-14 illustrates the effects of a temperature change on the density of water. Density changes of the fluid in the vessel affects the accuracy of the hydrostatic level detectors. The pressure gauge and indication circuitry must be properly calibrated to compensate for the changes in fluid density, and obtain accurate level indication.

Assuming the pressure detector used to determine level is at the surface of the earth and is not changing, the gravitational acceleration remains constant (g↔). And if the density of the fluid remains constant (ρ↔) then the changes in pressure in the hydrostatic detector will be solely and directly due to the change in level. $P = \overset{\leftrightarrow}{\rho}\,\overset{\leftrightarrow}{g}\,z$. Thus if level increases $(z\uparrow)$, then pressure will increase proportionally $(P\uparrow)$ $\overset{\leftrightarrow}{\rho}\,\overset{\leftrightarrow}{g}\,\overset{\uparrow}{z} = P\uparrow$.

OPEN VESSEL DIFFERENTIAL PRESSURE (D/P) LEVEL DETECTORS

Open vessel differential pressure (D/P) level detectors also operate on the hydrostatic head pressure principle. A differential pressure transmitter (Figure 1-15) measures vessel hydrostatic pressure in reference to atmospheric pressure. As vessel level rises, hydrostatic pressure increases. The differential pressure felt by the transmitter increases since atmospheric pressure remains relatively constant. This increase in differential pressure is processed by the level indication circuitry.

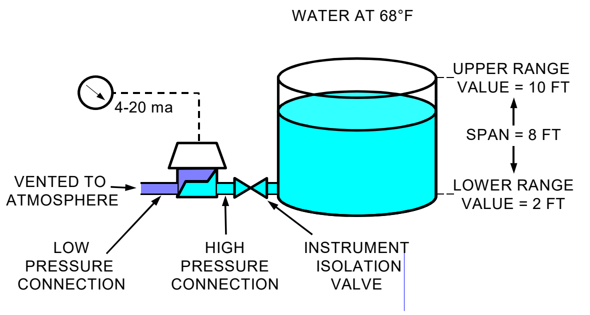
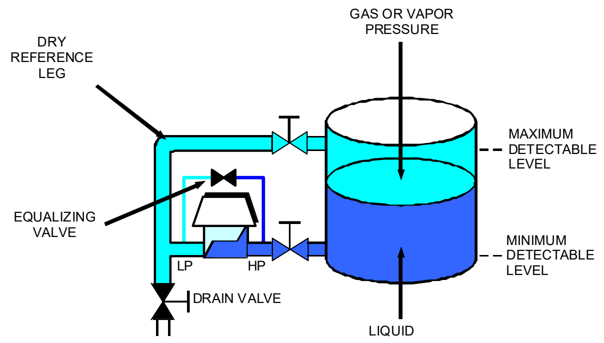


Figure 1-15 Open Vessel D/P Level Detector

CLOSED VESSEL DIFFERENTIAL PRESSURE (D/P) LEVEL DETECTORS

Some vessels contain hazardous materials and cannot be left open to the atmosphere. Other vessels must be pressurized. Closed vessel differential pressure level detectors can be used to measure level in these vessels.

Closed vessel level detectors operate very similarly to open vessel level detectors. However, the effect of static pressure in the closed vessel, due to vapor pressure above the fluid level, must be taken into account. Since the high pressure (HP) side of the detector senses the pressure exerted by the column of fluid in the vessel and the static vapor or gas pressure exerted on top of the fluid, indicated level would be inaccurate if the static pressure was not accounted for. The static pressure element must be accounted for, for the detector to indicate the correct level. To compensate for the static pressure above the fluid volume, a "dry" reference leg is used (Figure 1-16).



Dry Reference Leg

Figure 1-16 Dry Reference Leg Closed Vessel D/P Level Detectors with Dry Reference Leg

In this application, instead of referencing the hydrostatic head developed by the fluid to a known reference such as atmospheric, the static pressure above the fluid is applied to the reference (low pressure, LP) side of the detector by a pipe and isolation valve. The high pressure side of the detector senses both hydrostatic pressure due to fluid level and static pressure due to the volume of vapor or gas.

In this manner, the static pressure in the tank is sensed on both sides of the detector, and therefore cancelled. Level indication, then, is a result only of variations in the column of fluid above the high pressure tap of the detector.

In cases where there are two phases of a fluid in a closed vessel, a "wet" reference leg can be used (as shown in Figure 1-17). The vapors at the top of the vessel are condensed and used to form a second liquid level (reference leg). This second liquid level is maintained constant by either a condensation or fill connection.

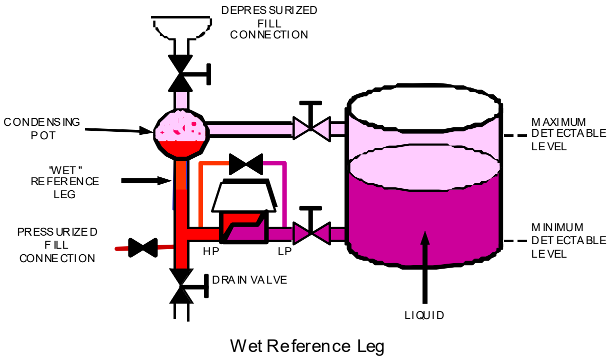


Figure 1-17 Closed Vessel D/P Level Detectors with Constant Filled Wet Reference Leg

In the two-phase, closed-vessel level detector, the reference leg is wet and is used as the high pressure end. Due to the relative constant temperature (density) and level in the reference leg, the high pressure side provides a constant reference point for varying tank levels. As vessel level rises, the hydrostatic head pressure in the vessel increases. Since the hydrostatic head pressure in the reference leg remains constant, the differential pressure felt by the transmitter decreases. This change in differential pressure is processed by the level indication circuitry. When actual vessel level equals reference leg level, the differential pressure is "zero," and the level indication is at its maximum.

Exam Bank No.: 3178 **Last used on an NRC exam:** Never

RO Sequence Number: 74

Unit 1 is at 100% power.

Which of the following will **LOWER** overall thermal efficiency?

A. Reducing turbine inlet steam moisture content.

B. Reducing condensate depression.

C. Raising turbine exhaust pressure.

D. Raising temperature of feedwater entering the steam generators.

Answer: C Raising turbine exhaust pressure

Exam Bank No.: 3178 **Source:** New **Modified from** N/A

K/A Catalog Number: 193005 K1.03 Describe how changes in secondary system parameter affect thermodynamic efficiency.

RO Importance: 2.5 **Tier:** 4 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(14)

STP Lesson: LOT 102.56 **Objective Number:** N99821

Explain how changes in secondary system parameters affect plant efficiency.

Reference: LOT 102.56 PPT slides 47-50

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because removing the moisture results in a higher overall enthalpy of the fluid that is sent through the turbine and this could be confused with adding more heat rather than removing lower enthalpy fluid. One formula for efficiency is Work output/Heat input.

B: INCORRECT. Plausible because reducing condensate depression affects pump NPSH and the student may think that this may cause more pump work but reducing condensate depression reduces the amount of heat that has to be reduces the amount of heat that has to be added to the process in the steam generators, which increases efficiency.

C: CORRECT. Increasing the exhaust pressure at the turbine exhaust reduces turbine work and increases heat rejected in the condenser. This reduces overall cycle efficiency.

D: INCORRECT. Plausible because this could also be confused with reducing pump efficiency requiring more work by the system pumps but this reduces the amount of heat that has to be added in the steam generator which is the reason we use feedwater heaters.

Question Level: H **Question Difficulty** 2

Justification:

The student must assess the affect of the given conditions and comprehend the thermodynamic effect on the plant efficiency.

FEEDWATER HEATING

Feedwater preheating is the process in which a fraction of the steam/water is removed from the turbine and used to preheat the condensate and feedwater.

The heat extracted is not lost to the cooling water in the condenser but is used to preheat the feedwater in the feedwater preheaters.

Therefore, the total heat rejected from the cycle is reduced, and plant efficiency increases.

PLANT PARAMETER CHANGES AFFECTING EFFICIENCY

Increasing condenser vacuum – the lower the pressure in the main condenser, the greater the work done by the turbine and the greater the overall plant efficiency.

Increasing circulating water system flow rate – a higher circulating water system flow rate reduces condenser temperature, lowering the pressure at the exhaust of the turbine. Lower pressure at the exhaust of the turbine increases plant efficiency.

PLANT PARAMETER CHANGES AFFECTING EFFICIENCY

Lowering circulating water system inlet temperature – lower circulating water system inlet temperature increases the heat transfer rate in the condenser, resulting in lower condenser temperature and pressure. Lower pressure at the exhaust of the turbine increases plant efficiency.

Reducing condensate depression – the less heat energy rejected to the circulating water system by excessively subcooling the condensate, the more efficient the plant.

PLANT PARAMETER CHANGES AFFECTING EFFICIENCY

Increasing feedwater heating – for greatest efficiency, heat transfer should occur at a constant temperature. By raising feedwater temperature closer to T_{sat} for the existing pressure, less energy from the fission process is required (the temperature at which the heat is transferred rises). This raises plant efficiency.

Increasing steam temperature at the turbine entrance – higher inlet steam temperature increases the available work that can be extracted from the turbine. More work results in an increase in plant efficiency.

Exam Bank No.: 3179 **Last used on an NRC exam:** Never

RO Sequence Number: 75

Steam Flow channel FT-512 is density compensated because, for a steam pressure increase at a constant volumetric flow rate, steam density will _____(1)_____ and the actual mass flow rate will _____(2)_____.

A. (1) increase
(2) decrease

B. (1) increase
(2) increase

C. (1) decrease
(2) decrease

D. (1) decrease
(2) increase

Answer: B (1) increase (2) increase

Exam Bank No.: 3179 **Source:** Bank **Modified from** N/A

K/A Catalog Number: 193006 K1.12 Explain why flow measurements must be corrected for density changes.

RO Importance: 2.5 **Tier:** 4 **Group/Category:** 2 **10CFR Reference:** 55.41(b)(14)

STP Lesson: LOT 102.57 **Objective Number:** N99848

Explain why flow measurements must be density compensated.

Reference: LOT 102.57 Student HO pgs 61-63, LOT 202.15 PPT slide 40

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as density does increase and if the student confuses how density compensation is used to correct mass flow rate.

B: CORRECT. If you increase steam pressure the steam is compressed raising density. For a constant volumetric flow rate if the density rises, the actual mass flow rate rises.

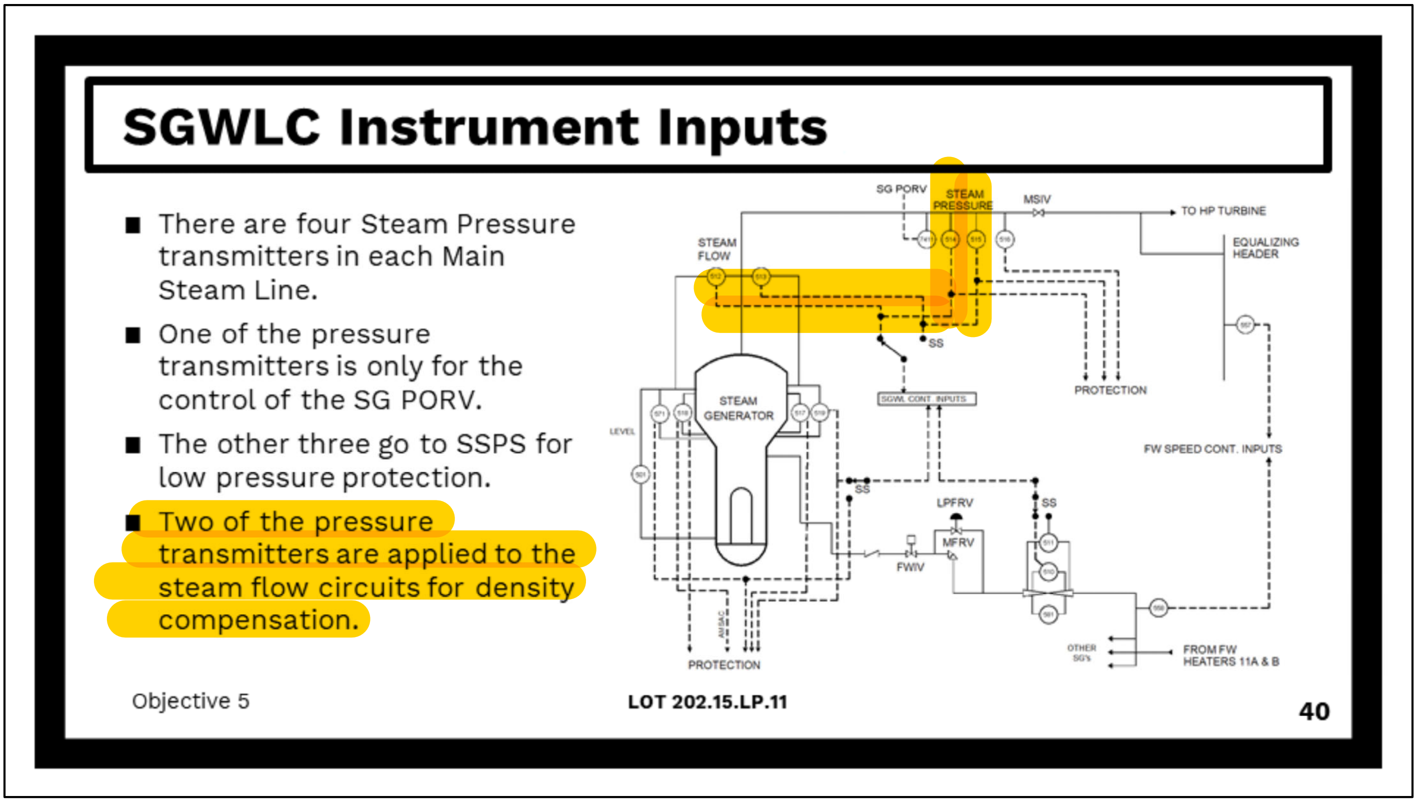
C: CORRECT. Plausible because specific volume decreases and since it is the inverse of density this can be easily confused and if the student confuses how density compensation is used to correct mass flow rate.

D: INCORRECT. Plausible because specific volume decreases and since it is the inverse of density this can be easily confused. The second part is correct.

Question Level: H **Question Difficulty** 2

Justification:

The student must comprehend how a change in density affects flow instrumentation and mass flow vs. volumetric flow.



SGWLC Instrument Inputs

- There are four Steam Pressure transmitters in each Main Steam Line.
- One of the pressure transmitters is only for the control of the SG PORV.
- The other three go to SSPS for low pressure protection.
- Two of the pressure transmitters are applied to the steam flow circuits for density compensation.

Objective 5

LOT 202.15.LP.11

- **INSTR. NOTE:** Objectives 7 & 8 (21412, 05686)
- Four steam pressure transmitters
  - One solely used for the power operated steam relief
  - Three used for protection
- Two PTs provide density compensation to steam flow.
  - For each SG loop
    - Loop 1 PT514 for FT512, PT515 for FT513
    - Loop 2 PT524 for FT522, PT525 for FT523
    - Loop 3 PT534 for FT532, PT535 for FT533
    - Loop 4 PT544 for FT542, PT545 for FT543
- **INSTR. NOTE:** Objective 5 (21430)
  - Protective outputs for:
    - Safety injection, and steam line isolation on low (735 2/3) steamline pressure trip in effect above P-11 (1985)
    - Steamline isolation on high steam pressure rate only when below P-11 and not blocked. This protects against excessive cooldown.
  - Indication is located on:
    - Two meters on CP-006: 0 -1400 psig
    - One meter on auxiliary shutdown panel. (ZLP-100**)**

E. Orifice Meter

1. An orifice meter is another device used to measure flow
    a. It operates in manner similar to that of venturi meter
    b. Rather than giving gradually converging and diverging sections, however, changes are sharp
    c. A typical orifice meter is shown in Figure 6-11
2. The relationships for average velocity in orifice ($\vec{v}_{av}$) and mass flow rate ($\dot{m}$) are same as for venturi meter
    a. However, orifice coefficient (C_o) is used in place of venturi factor (C_v)
    b. The orifice coefficient ranges from about 0.5 to 0.97, depending on Reynolds number of flow and size of orifice.

F. Density Effects on Flow Measurement

1. All of discussed methods of flow measurement require calculations using density
    a. Both pressure and temperature affect density of fluid
    b. Pressure increases tend to increase density of fluids (P↑ ρ↑)

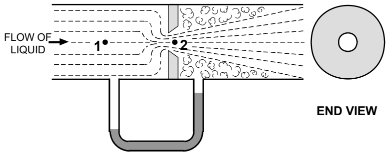
$$P_2 = \left(15{,}705\frac{lb_f}{ft^2}\right)\left(\frac{ft^2}{144\ in^2}\right)$$

$$P_2 = \left(109.06\frac{lb_f}{in^2}\right)$$

The pressure in the throat of the venturi is 109 psig

Objective 27

Figure 6-11 / TP 6-95



Objective 28

INSTRUCTOR GUIDE	KEY POINTS, AIDS, QUESTIONS/ANSWERS
c. Temperature increases cause density of most fluids to decrease (T↑ρ↓) 2. Since most liquids are noncompressible, pressure has insignificant effect on density a. When measuring flow rates of liquids where temperature changes of fluid are small, volumetric flow rate continuity equation is used in calculational circuitry b. The minor effects of density changes are uncompensated 3. Temperature has more pronounced effect on density of all fluids, especially gases and vapors a. Not only does density vary inversely with temperature, but magnitude of change in density increases as temperature increases b. Where temperature of measured fluid changes, flow measurement circuitry uses mass flow rate continuity equation which allows for density compensation 4. For example, steam flow is mass flow rate measurement, which must be compensated for steam pressure a. With constant volume flow rate, increase in steam pressure causes density and actual mass flow rate to increase b. If steam flow instrument were not pressure compensated, it would not accurately measure change in mass flow rate	

INSTRUCTOR GUIDE	KEY POINTS, AIDS, QUESTIONS/ANSWERS
c. This effect can be seen in Equation 6-1, Equation 6-2 and Equation 6-3	**Example 6-11 / TP 6-96 and TP 6-97** Flow in system is monitored by uncompensated volumetric flow measurement system. Temperature in system rises. A constant mass flow rate is maintained. What effect does temperature change have on indicated volumetric system flow? From Equation 6-1 and Equation 6-2: Mass flow rate is given by $\dot{m} = v_{av}A\rho$ Volumetric flow rate is given by $\dot{V} = v_{av}A$ Therefore, mass flow rate and volumetric flow rate are related by Equation 6-3: $\dot{m} = \dot{V}\rho$ The increase in system fluid temperature causes density of fluid to decrease. Since mass flow rate is held constant, volume flow rate must increase to offset decrease in density. The velocity of fluid increases to maintain mass flow rate constant. Indicated volumetric flow will increase
G. Two-Phase Fluid Flow 1. All of fluid flow relationships discussed previously are for flow of single-phase of fluid, whether liquid or vapor 2. There are several techniques to predict head loss due to fluid friction for two-phase flow a. One accepted technique involves two-phase flow friction multiplier (R) b. The two-phase flow friction multiplier is defined as ratio of two-phase head loss divided by head loss evaluated using single-phase saturated liquid properties	**Objective 29**

Exam Bank No.: 3218 **Last used on an NRC exam:** Never

SRO Sequence Number: 76

Unit 1 is at 100% power when the following occurs:

- RCP 1A Thermal Barrier Isolation, MOV-0339 closes due to a failure of RCP 1A THERM BAR CCW DISCH TEMP switch 1-CC-TSH-4620.
- The crew enters 0POP04-RC-0002, Reactor Coolant Pump Off Normal.
- 5 minutes later the running Charging Pump trips.

The Unit Supervisor should direct the RO to trip the reactor AND ensure the main turbine is tripped. The specific mitigative strategy to protect RCP 1A is to stop RCP 1A _____(1)______ AND to establish seal injection with the PDP if all RCP Number 1 Seal Inlet Temperatures are less than a MAXIMUM of __(2)__ ºF.

A. (1) within 1 minute
(2) 310

B. (1) within 1 minute
(2) 230

C. (1) between 3 and 5 minutes after tripping the reactor
(2) 310

D. (1) between 3 and 5 minutes after tripping the reactor
(2) 230

Answer: B (1) within 1 minute (2) 230

Exam Bank No.: 3218 **Source:** New **Modified From**

K/A Catalog Number: APE015 AA2.10 Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection

SRO Importance: 3.7 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The SRO must display knowledge of when to implement a specific mitigative strategy within an abnormal operating procedure.

STP Lesson: LOT 505.01 **Objective Number:** 92109

Given a plant condition, describe and/or interpret the requirements and/or limits of a precaution or step of a referenced procedure.

Reference: 0POP04-RC-0002, CIP

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as this is the correct time frame and 310F is another procedurally directed temperature to trip the RCP.
B: CORRECT: With the given conditions, the RCP must be tripped within 1 minute. Seal injection should be established from the PDP if seal inlet temperatures have remained below 230F
C: INCORRECT: Plausible as between 3 and 5 minutes is a time frame in the procedure for excessive seal leakoff flow and 310F is another procedurally directed temperature to trip the RCP.
D: INCORRECT: Plausible as between 3 and 5 minutes is a time frame in the procedure for excessive seal leakoff flow and this is the correct temperature.

Question Level: H **Question Difficulty** 3

Justification:

The student must recall a specific mitigative strategy to protect the RCP under these conditions.

Conditional Information Page

RCP TRIP CRITERIA

IF ANY VALID condition listed below occurs, THEN PERFORM the following:

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP affected RCP(s).
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection
4. *CONTINUE* at Step 1.0 of procedure as resources permit

RCP TRIP conditions:

Motor Upper or Lower Radial Bearing Temp - GREATER THAN OR EQUAL TO 195°F	Lower Seal Water Bearing Temp - GREATER THAN OR EQUAL TO 230°F
Case Vibration (Mtr_Accel-Vert OR Mtr_Accel-Horiz) – a. GREATER THAN OR EQUAL TO 5 MILS b. GREATER THAN OR EQUAL TO 3 MILS AND RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 0.2 MIL PER HOUR	Shaft Vibration (Brg2-Vert OR Brg2-Horiz) – a. GREATER THAN OR EQUAL TO 20 MILS b. GREATER THAN OR EQUAL TO 15 MILS AND RATE OF VIBRATION INCREASE IS GREATER THAN OR EQUAL TO 1.0 MIL PER HOUR
Seal 1 Water Inlet Temp - GREATER THAN OR EQUAL TO 230°F	Motor Stator Winding Temp - GREATER THAN OR EQUAL TO 310°F
Number 1 Seal DP - LESS THAN 220 PSID	

LOSS OF ALL RCP SEAL COOLING

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP affected RCP(s) within 1 minute.
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection **WHILE CONTINUING WITH** the following actions below:
   IF all RCP seal 1 inlet temperatures are LESS THAN 230°F, THEN PERFORM the following:
   a. ENSURE “RECIRC HCV-0285” PDP recirculation valve is 100% OPEN
   b. START the PDP.
   c. Slowly CLOSE “RECIRC HCV-0285”, to establish seal injection flow to lower seal water inlet and lower seal water bearing temperatures at LESS THAN 1°F PER MINUTE.

RCP TRIP CRITERIA FOR HIGH NUMBER 1 SEAL LEAKOFF FLOW

IF RCP Number 1 Seal leakoff flow increases to GREATER THAN 6 gpm **OR** pegged high**,** THEN PERFORM the following:

1. IF the Reactor is critical, THEN PERFORM the following:
   a. TRIP the Reactor.
   b. ENSURE Main Turbine tripped.
2. STOP the affected RCP.
3. IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection.
4. *CONTINUE* actions of this procedure as resources permit.
5. CLOSE affected RCP Number 1 Seal leakoff isolation valve between 3 to 5 minutes after stopping RCP.

• RCP 1A(2A) "SEAL NO 1 LKF ISOL FV-3154"	• RCP 1B(2B) "SEAL NO 1 LKF ISOL FV-3155"
• RCP 1C(2C) "SEAL NO 1 LKF ISOL FV-3156"	• RCP 1D(2D) "SEAL NO 1 LKF ISOL FV-3157"

6. *MONITOR* CCW flow - ADEQUATE.

RCP MOTOR THRUST BEARING TEMPERATURE HIGH

IF Motor Upper or Lower Thrust Bearing Temp - GREATER THAN OR EQUAL TO 195°F, THEN, PERFORM Step 2 of the procedure body (page 7).

Exam Bank No.: 3219 **Last used on an NRC exam:** Never

SRO Sequence Number: 77

The unit is at 100% power.

- CCW Pump 1A is in RUN.
- CCW Pump 1B is in OFF.
- CCW Pump 1C is in STANDBY

A total Loss of Offsite Power occurs.

With NO operator action…

5 minutes after the Loss of Offsite Power, CCW Pump(s)_____(1)_____ are running.

Per Technical Specifications, the crew will be required to restore at least ONE offsite source within ___(2)__ hours. **(Assume CRMP will not be applied)**

A. (1) 1A and 1C ONLY
(2) 8

B. (1) 1A and 1C ONLY
(2) 24

C. (1) 1A, 1B, and 1C
(2) 8

D. (1) 1A, 1B, and 1C
(2) 24

Answer: D (1) 1A, 1B, and 1C (2) 24

Exam Bank No.: 3219 **Source:** New **Modified From**

K/A Catalog Number: APE 056 AA2.24 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: CCW pump ammeter, flowmeter and run indicator

SRO Importance: 3.1 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must display knowledge of Technical Specification actions and apply them to the given conditions.

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given a plant condition, state the actions required to be performed per the applicalbe Off-Normal procedure.

Reference: Technical Speicification 3.8.1.1, LOT 201.41 PPT

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as CCW 1B Mode Selector Switch is in OFF and with confusion over the TS action time. 8 hours is a valid action time mentioned in TS 3.8.1.1.

B: INCORRECT: Plausible as CCW 1B Mode Selector Switch is in OFF. This is the correct action time to restore 1 offsite source.

C: INCORRECT: Plausible as all CCW pumps will start and with confusion over the TS action time. 8 hours is a valid action time mentioned in TS 3.8.1.1.

D: CORRECT: With a Loss of Offsite Power, the 4160V Buses will strip, and the CCW pumps will all be sequenced on. Per TS 3.8.1.1, 1 offsite source must be restored within 24 hours.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess conditions and determine what happened and then apply Technical Specifications.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE.

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System (1), and

b. Three separate and independent standby diesel generators, each with a separate fuel tank containing a minimum volume of 60,500 gallons of fuel, and an automatic load sequencer.

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

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Within 72 hours restore the offsite circuit to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a standby diesel generator inoperable, demonstrate the OPERABILITY of the above-required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.5) for each such standby diesel generator separately within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generator(s). Within 14 days restore the inoperable standby diesel generator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. (12)

c. With one offsite circuit of the above-required A.C. electrical power sources and one standby diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Specification 4.8.1.1.1.a. within 1 hour and at least once per 8 hours thereafter; and if the standby diesel generator became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE standby diesel generator(s) by performing Surveillance Requirement 4.8.1.1.2.a.5) within 8 hours, unless it can be demonstrated there is no common mode failure for the remaining diesel generators; within 12 hours restore at least one of the inoperable sources to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (12)

d. With one standby diesel generator inoperable in addition to ACTION b. or c. above, verify that:

   1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generators as a source of emergency power are also OPERABLE, and

   2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

   If these conditions are not satisfied within 24 hours, apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and In COLD SHUTDOWN within the following 30 hours.

e. With two of the above required offsite A.C. circuits inoperable, within 24 hours restore at least one of the inoperable offsite sources to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours.

f. With two or three of the above required standby diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1.a. within 1 hour and at least once per 8 hours thereafter. With three of the above required standby diesel generators inoperable, within 2 hours restore at least one standby diesel generator to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. (12) With two of the above required standby diesel generators inoperable, within 24 hours restore at least two standby diesel generators to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Train A Sequence Times in Seconds

Function	Mode I	Mode II	Mode III
Incoming Breaker 480V Bus E1A1	N/A	1	1
Incoming Breaker 480V Bus E1A2	N/A	1	1
High Head Safety Injection Pump 1A	6	N/A	6
Low Head Safety Injection Pump 1A	10	N/A	10
Containment Spray Pump 1A	15	N/A	15
Reactor Containment Fan Cooler 11A	15	15	15
Reactor Containment Fan Cooler 12A	15	15	15
Component Cooling Water Pump 1A	20	20	20
Essential Cooling Water Pump 1A	25	25	25
Auxiliary Feedwater Pump 11	30	30	30
Control Room Makeup Fan 11A	35	N/A	35
CR/EAB Emergency HVAC	35	35	35
Standby Ess. Chiller and CHW pump 11A	35	35	35
Containment Spray Pump 1A	40	N/A	40
Essential Chiller 12A	240	240	240
Sequence Complete	280	280	280
Containment Spray Pump 1A Permissive	15	N/A	15
Containment Spray Pump 1A Timer 62	17	N/A	17
Containment Spray Pump 1A Permissive	40	N/A	40

Train B Sequence Times in Seconds

Function	Mode I	Mode II	Mode III
Incoming Breaker 480V Bus E1B1	N/A	1	1
Incoming Breaker 480V Bus E1B2	N/A	1	1
High Head Safety Injection Pump 1B	6	N/A	6
Low Head Safety Injection Pump 1B	10	N/A	10
Containment Spray Pump 1B	15	N/A	15
Reactor Containment Fan Cooler 11B	15	15	15
Reactor Containment Fan Cooler 12B	15	15	15
Component Cooling Water Pump 1B	20	20	20
Essential Cooling Water Pump 1B	25	25	25
Auxiliary Feedwater Pump 12	30	30	30
Control Room Makeup Fan 11B	35	N/A	35
CR/EAB Emergency HVAC	35	35	35
Standby Ess. Chiller and CHW pump 11B	35	35	35
Containment Spray Pump 1B	40	N/A	40
Essential Chiller 12B	240	240	240
Sequence Complete	280	280	280
Containment Spray Pump 1B Permissive	15	N/A	15
Containment Spray Pump 1B Timer 62	17	N/A	17
Containment Spray Pump 1B Permissive	40	N/A	40

Train C Sequence Times in Seconds

Function	**Mode I**	**Mode II**	**Mode III**
Incoming Breaker 480V Bus E1C1	N/A	1	1
Incoming Breaker 480V Bus E1C2	N/A	1	1
High Head Safety Injection Pump 1C	6	N/A	6
Low Head Safety Injection Pump 1C	10	N/A	10
Containment Spray Pump 1C	15	N/A	15
Reactor Containment Fan Cooler 11C	15	15	15
Reactor Containment Fan Cooler 12C	15	15	15
Component Cooling Water Pump 1C	20	20	20
Essential Cooling Water Pump 1C	25	25	25
Auxiliary Feedwater Pump 13	30	30	30
Control Room Makeup Fan 11C	35	N/A	35
CR/EAB Emergency HVAC	35	35	35
Standby Ess. Chiller and CHW pump 11C	35	35	35
Containment Spray Pump 1C	40	N/A	40
Essential Chiller 12C	240	240	240
Sequence Complete	280	280	280
Containment Spray Pump 1C Permissive	15	N/A	15
Containment Spray Pump 1C Timer 62	17	N/A	17
Containment Spray Pump 1C Permissive	40	N/A	40

Exam Bank No.: 3186

Last used on an NRC exam: Never

SRO Sequence Number: 78

Unit 2 is in MODE 4.

- Instrument Air (IA) header pressure is 70 psig and slowly lowering.
- The crew is implementing 0POP04-IA-0001, Loss of Instrument Air, Addendum 2, Loss of IA in Mode 4.

The failure position for the RHR HX Outlet Valves is _____(1)_____.

To control RCS temperature, the Unit Supervisor should implement Addendum 6, RCS Temperature Control on RHR and lower cooldown rate by OPENING the ______(2)_____ valve on the operating RHR train.

A. (1) closed
   (2) RHR MINI FLOW ISOL

B. (1) closed
   (2) SI HOT LEG INJECTION

C. (1) open
   (2) RHR MINI FLOW ISOL

D. (1) open
   (2) SI HOT LEG INJECTION

Answer: C (1) open: (2) RHR MINI FLOW ISOL

Exam Bank No.: 3186 **Source:** New **Modified From** N/A

K/A Catalog Number: APE 065 AA2.08 Ability to determine and interpret the following as they apply to the Loss of Instrument Air: Failure modes of air-operated equipment

SRO Importance: 3.3 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

Student must have knowledge of when to implement addendums within the procedure and have knowledge of content of the procedure vs. overall mitigative strategy.

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-IA-0001, Addendum 2, 5, and 6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible with a misunderstanding of the failure posItion, the second part is correct.

B: INCORRECT: Plausible with a misunderstanding of the failure postion and if the student believes that re-routing flow through the hot leg injection valves would result in a lower cooldown rate than the normally used cold leg injection valves by creating a recirculation path back to the suction of the RHR pumps, which take a suction off the associated RCS hot leg.

C: CORRECT: 0POP04-IA-0001, Addendum 5 states that the RHR HX Outlet Valves fail open on loss of air. Addendum 2 Step 1 requires implementation of Addendum 6 for RCS Temperature control. The RHR MINI FLOW ISOL valves are opened to reduce the cooldown rate.

D: INCORRECT: Plausible as the 1st part is correcct and if the student believes that re-routing flow through the hot leg injection valves would result in a lower cooldown rate than the normally used cold leg injection valves by creating a recirculation path back to the suction of the RHR pumps, which take a suction off the associated RCS hot leg..

Question Level: F **Question Difficulty** 3

Justification:

The student must know the failure position of IA system valves and recall a specific mitigative strategy within an AOP addendum.

NOTE

IF this Addendum is being performed due to a section of piping in the IA System being isolated, THEN the Unit Supervisor/Shift Manager should be consulted to evaluate the effects to individual components within the isolated section. The Unit Supervisor/Shift Manager may direct steps for unaffected components to be marked N/A at his discretion.

NOTE

Addendum 5, Safety Related Instrument Air Valves, provides a list of the fail position of all Safety-Related valves and dampers operated by IA.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

_____ 1.0	**VERIFY At Least Two SGs Available For Heat Sink:**
	● Steam Generator NR level within any of the following
	  ● Trending to between 68% and 74%
	  ● Between 68% and 74%
	● SG PORVs operable
	● AFW available to feed SGs with operable PORV
		PERFORM the following:
		a. OPERATE RHR to control RCS temperature per Addendum 6, RCS Temperature Control on RHR.
		b. GO TO Step 5.0 of this Addendum.

_____ 2.0	**CHECK All RHR trains – SECURED {CP001}**	SECURE all RHR trains except the train providing Low Pressure Letdown.

VALVE NUMBER	DESCRIPTION	FAILURE POSITION
RC-FV-3650	RMWS to PRT	CLOSED
RC-FV-3651	RMW to PRT OCIV	CLOSED
RC-FV-3652	Nitrogen to PRT OCIV	CLOSED
RC-FV-3400	Flange Leak-off Valve	OPEN
RC-LV-3655	PRT to LWPS RCDT Pump	CLOSED
RC-PCV-655B	RCS Loop 4 to PRZR Spray	CLOSED
RC-PCV-0655C	Loop 1 Pressurizer Spray	CLOSED
RH-HCV-864	RHR HX 2A Outlet	OPEN
RH-HCV-865	RHR HX 2B Outlet	OPEN
RH-HCV-866	RHR HX 2C Outlet	OPEN
RH-FCV-851	RHR HX 2A Bypass	CLOSED
RH-FCV-852	RHR HX 2B Bypass	CLOSED
RH-FCV-853	RHR HX 2C Bypass	CLOSED
RH-FV-3950	RHR to SIS Test Line	CLOSED
RH-FV-3951	RHR to SIS Test Line	CLOSED
RH-FV-3955	RHR to SIS Test Line	CLOSED
RH-FV-3956	RHR to SIS Test Line	CLOSED
RH-FV-3960	RHR to SIS Test Line	CLOSED
RH-FV-3961	RHR to SIS Test Line	CLOSED
RM-FV-7659	RMW Non Essential Service Isolation	CLOSED
RM-FV-7663	RMW Non Essential Service Isolation	CLOSED
RM-LV-7651	Reactor Makeup Water Storage Tank Fill Valve	CLOSED
SB-FV-4150	SG D Blowdown Isolation	CLOSED
SB-FV-4151	SG C Blowdown Isolation	CLOSED
SB-FV-4152	SG B Blowdown Isolation	CLOSED
SB-FV-4153	SG A Blowdown Isolation	CLOSED
SI-FV-3936	RWST to SFPCS Isolation	CLOSED
SI-FV-3937	RWST to SFPCS Isolation	CLOSED

Addendum 6	**RCS Temperature Control on RHR**	Addendum 6 Page 1 of 2

STEP		**ACTIONS/EXPECTED RESPONSE**	**RESPONSE NOT OBTAINED**
_____	**1.0**	**CHECK RCS Level NOT In Reduced Inventory - GREATER THAN 36 FEET 3 INCHES**	ENSURE RCS level greater than or equal to +11 inches.

NOTE

DO NOT exceed an RCS cooldown rate of 80°F per hour.

STEP		**ACTIONS/EXPECTED RESPONSE**	**RESPONSE NOT OBTAINED**
_____	**2.0**	**PERFORM The Following To Reduce RCS Cooldown**	
_____	**a.**	**ENSURE RHR “MINI FLOW ISOL” valve open on operating RHR Train(s) {CP001}** ● “MOV-0067A” ● “MOV-0067B” ● “MOV-0067C”	
_____	**b.**	**CHECK RCS Cooldown Rate – LESS THAN 80°F PER HOUR**	b. PERFORM any of the following to restore RCS Cooldown Rate to less than 80°F per hour: ● IF two RHR Trains are operating, THEN SECURE one RHR pump. {CP001} ● CLOSE Cold Leg Injection valve on operating RHR train(s): {CP001} ● “LOOP A Tc INJ” “MOV-0031A” ● “LOOP B Tc INJ” “MOV-0031B” ● “LOOP C Tc INJ” “MOV-0031C”

Exam Bank No.: 3199 **Last used on an NRC exam:** Never

SRO Sequence Number: 79

Both Units are at 100% power.

- Notification has been received from the STP Coordinator of a degraded voltage condition on the offsite electrical system.
- Both SWYD N VOLTS and SWYD S BUS VOLTS indicate 336 KV

All Offsite circuits are ___(1)___.

The minimum switchyard voltage ensures that ___(2)___ are able to perform their design function.

A. (1) OPERABLE
(2) Reactor Coolant Pumps

B. (1) OPERABLE
(2) ESF loads

C. (1) INOPERABLE
(2) Reactor Coolant Pumps

D. (1) INOPERABLE
(2) ESF loads

Answer: D (1) INOPERABLE (2) ESF loads

Exam Bank No.: 3199 **Source:** New **Modified From** N/A

K/A Catalog Number: APE 077 G2.1.32 Ability to explain and apply system limits and precautions.

SRO Importance: 4.0 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(2)

SRO Justification:

The students must know the contents of the procedure and not the overall mitigative strategy.

STP Lesson: LOT 501.01 **Objective Number:** 92109

Given a plant condition, describe and/or interpret the requirements and/or limits of a precaution or step of a referenced procedure.

Reference: 0POP04-AE-0005, EC 5000, 0PSP03-EA-0002

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because another number given in 0POP04-AE-0005 is 324 KV but this number applies to automatic Load Tap Changer capability as stated in the basis for step 10 of 0POP04-AE-0005 and RCPs are needed per Techncal Specifications to satisfy RCS Loop Operability requirements in MODES 1-4.

B: INCORRECT. Plausible because another number given in 0POP04-AE-0005 is 324 KV but this number applies to automatic Load Tap Changer capability as stated in the basis for step 10 of 0POP04-AE-0005. The second part is correct.

C: INCORRECT. Plausible because RCPs are needed per Techncal Specifications to satisfy RCS Loop Operability requirements in MODES 1-4. The first part is correct.

D: CORRECT. Per 0POP04-AE-0005, a degraded switchyard voltage condition is less than 339 KV as indicated in the Control Room and all offsite circuits are inoperable below that value. Per the step basis, EC 5000 concluded that the UAT and "B" Train ESF Transformer LTC operating band permit the ESF switchgear and loads to perform their design function during normal and worst case DBEs concurrent with a minimum switchyard voltage of 339 KV.

Question Level: H **Question Difficulty** 3

Justification:

The student must know system limits and apply to given conditions to determine operability of the offsite sources and bases for the limits.

NOTE

- Voltage indication on the 345 KV system may be in the normal operating range, while the potential to be in a degraded voltage condition exists in the event both STPEGS Units trip (i.e., loss of all STPEGS generation).
- Refer to Main Generator Capability Curve - Curve Book Figure 7.1 for operating limits of the Main Generator, as applicable.
- Preferred switchyard voltage readings are obtained by contacting the STP Coordinator and TDSP directly for voltage data obtained from available SCADA telemetry.
- Foldout CIP should be opened.
- Procedure Steps 4.0 through 17.0 are applicable in Modes 1-4.
- Procedure Steps 18.0 through 31.0 are applicable in Modes 5 and 6.

_____	**1.0**	**CHECK Entry Into This procedure Is Due To Notification From The STP Coordinator That ERCOT Has Lost The Ability To Assess The Operability Of The Electric System.**	GO TO Step 3.0.
_____	**2.0**	**GO TO Applicable Step:** • **Modes 1, 2, 3 or 4 – Step 8.0** • **Modes 5 or 6 – Step 22.0**	
_____	**3.0**	**CHECK Unit In Modes 1, 2, 3 Or 4**	GO TO Step 18.0.
_____	**4.0**	**CHECK Both 345 KV Switchyard "SWYD N BUS VOLTS" AND "SWYD S BUS VOLTS" (CP-010) Bus Voltage Indications– LESS THAN 339 KV.**	GO TO Step 7.0.

STEP DESCRIPTION FOR 0POP04-AE-0005 STEP 4.0

STEP: CHECK Both 345 KV Switchyard "SWYD N BUS VOLTS" AND "SWYD S BUS VOLTS" (CP-010) Bus Voltage Indications– LESS THAN 339 KV

PURPOSE: To determine the applicable entry condition and confirm the accuracy of the onsite indications according to current Transformer Breaker lineup.

BASIS: This step determines which entry condition to this procedure exists in Modes 1-4 and confirms the accuracy of the onsite voltage indications in the event that the indicated Switchyard voltage is low. This determination is made using voltage indications for the offsite power grid that are installed in the STPEGS Unit Control Rooms. *The stated minimum voltage level is based on load/voltage studies performed by the Transmission Operator and is then used as an input to Calculation EC 5000, Voltage Regulation Study and DCP 04-11502-20.* Contact is made with the proper authority to confirm that a true low voltage condition exists, or exit from the procedure is provided in the event the onsite indications are inaccurate or the procedure was entered in error.

ACTIONS: Determine if the indicated 345 KV Switchyard voltage is degraded.

INSTRUMENTATION:

- "SWYD N BUS VOLTS" (CP-010)
- "SWYD S BUS VOLTS" (CP-010)

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

In Calculation No. EC 5000 (Voltage Regulation Study) for both Unit 1 and Unit 2, we conclude that the current Unit Auxiliary Transformer (UAT) LTC operating band and "B" Train ESF Transformer LTC operating band permit the ESF switchgear and loads to perform their design function during normal operations and worst case loading design basis events concurrent with a minimum switchyard voltage of 339 KV (while preventing the degraded grid undervoltage relays from spuriously actuating).

STEP DESCRIPTION FOR 0POP04-AE-0005 STEP 10.0

STEP: PLACE "ESF E1B(E2B) LOAD TAP CHANGER - MODE SELECT" Switch
To "MANUAL".

PURPOSE: To place the ESF Train "B" LTC to manual.

BASIS:

EC 5000 proves the UAT is capable of supplying all the Aux and Standby busses as long as offsite power is greater than 324 KV with the LTC in automatic mode. Placing the ESF B Train LTC in manual will allow the UAT LTC to control ESF Train "B" Bus. Placing the ESF Train "B" Load Tap Changer in "MAN" precludes simultaneous operation of the Train "B" ESF and UAT Load Tap Changers in "AUTO".

ACTIONS: Place the ESF B Train LTC to manual at a Tap Position of "N".

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

ESF B Train LTC on CP003

KNOWLEDGE:

N/A

1.3.2 The ESF Transformer LTC - Future ETAP model is developed to perform future "study cases" of the plant configuration with the installation of Load Tap Changing (LTCs) transformers for the remaining AUX ESF Transformers (E1A, E1C, E2A and E2C). The ESF Transformer LTC - Future ETAP model will be used to perform analysis of the plant with LTCs installed on the remaining ESF transformers.

2.0 SUMMARY OF RESULTS

2.1 Minimum 4.16 kV System Voltage (Degraded Voltage) Analysis

The results of the degraded voltage analysis are tabulated in Section 9.1. The results demonstrate that the degraded voltage protection relay settings provided by Calculation EC 5052 [Ref. 7.1.2] will maintain adequate 4.16 kV and 480 V system voltages for continuous operation and that the terminal voltages for equipment required to operate in response to a LOCA are adequate with the 4.16 kV ESF bus voltages at the design basis minimum value.

2.2 Minimum 345 kV System (Grid) Voltage Requirements and ESF Motor Starting

The results of the minimum 345 kV system voltage requirements are tabulated in Section 9.2 and 9.9 for the load flow and static motor starting analyses, respectively.

2.2.1 Minimum Voltage for Normal Lineup

From the 345 kV system minimum voltage load flow and static motor starting analyses, the minimum allowable 345 kV system voltages required to ensure 4.16 kV ESF bus voltages recover above the maximum degraded voltage reset setting under normal lineup conditions (UAT available), are:

- 338.8 kV for the June 2017 Current Plant configuration: This limiting voltage is determined in Section 9.9.1 from static motor starting analysis results in Attachments C1, C2, C5, and C6.

2.2.2 Minimum 345 kV System Voltage for UAT Out of Service (OOS) and no Limiting Condition of Operation (LCO) in effect (with the restriction that bus alignments must match those developed in Calculation EC 5002 [Ref. 7.1.1]).

From the 345 kV system minimum voltage load flow and static motor starting analyses, the minimum allowable 345 kV system voltages required to ensure 4.16 kV ESF bus voltages recover above the maximum degraded voltage reset setting with a Unit's UAT Out-of-service are:

- 356.0 kV – Unit 1 UAT OOS for the June 2017 Current Plant configuration: This required voltage is determined in Section 9.9.1 from results in Attachment C3. Configuration V19-UAT1-OS6 restrictions apply [Ref. 7.1.1].
- 356.0 kV – Unit 2 UAT OOS for the June 2017 Current Plant configuration: This required voltage is determined in Section 9.9.1 from results in Attachment C4. Configuration V20-UAT2-OS6 restrictions apply [Ref. 7.1.1].

6.0 Acceptance Criteria

NOTE

- Addendum 2, Two Physically Independent Circuits, provides a drawing of rights of way and offsite circuits to aide in the definition of “two physically independent circuits”.
- Loss of one 13.8 KV Standby Bus to 4.16 KV ESF bus line constitutes loss of one required offsite source. (Reference 8.2)
- Loss of two 13.8 KV Standby busses to 4.16 KV ESF bus lines constitutes loss of two required offsite sources. (Reference 8.2)
- The preceding notes also apply when the 4.16 KV ESF bus is not energized by the 13.8 KV XFMR.
- Step 6.1 applies during standby diesel inoperability.
- Step 6.2 applies during offsite independent circuits inoperability.
- Note and Precaution 3.29 should be referred to for additional clarification regarding allowable indication to be utilized when obtaining 345 KV switchyard voltage.

6.1 Two physically independent circuits exist between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specifications 3.8.1.1.b, 3.8.1.1.f, and 4.8.1.1.1.a)

- North and South Bus in service with bus voltage ≥ 339 KV
- Two of the following Rights of Way with a 345 KV line are available:
    - NW Right of Way 1 (ANGSTR_STP1 / ANGSTR_WHITE_1)
    - NW Right of Way 2 (Elm Creek 27 OR WA Parish 39 OR Elm Creek 18)
    - Eastern Right of Way (Refuge 27 OR Jones Creek 18)
- Two of the following 13.8 KV XFMRs are available:
    - Unit Aux XFMR
    - Unit 1 Stby XFMR
    - Unit 2 Stby XFMR
- Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.

6.2 One circuit exists between the offsite transmission network and onsite Class 1E Distribution System as determined from Data Sheet 1, 2, 3, 4, and 9. (Technical Specification 3.8.1.1.a, 3.8.1.1.c, 4.8.1.1.1.a)

- North or South Bus in service with bus voltage ≥ 339 KV
- One of the following Rights of Way with a 345 KV line is available:
    - NW Right of Way 1 (ANGSTR_STP1 / ANGSTR_WHITE_1)
    - NW Right of Way 2 (Elm Creek 27 OR WA Parish 39 OR Elm Creek 18)
    - Eastern Right of Way (Refuge 27 OR Jones Creek 18)
- One of the following 13.8 KV XFMRs is available:
    - Unit Aux XFMR
    - Unit 1 Stby XFMR
    - Unit 2 Stby XFMR
- Three 13.8 KV Standby Buses energizing the 4.16 KV ESF bus lines.

6.3 Each ESF 4.16 KV bus E1A, E1B and E1C (E2A, E2B and E2C) is energized and supplying it's respective 480V Load Centers E1A1, E1A2, E1B1, E1B2, E1C1 and E1C2 (E2A1, E2A2, E2B1, E2B2, E2C1 and E2C2) via it's respective load center transformers as determined from Data Sheet 1, 2, 3, 4, and 9.
(Technical Specification 3.8.3.1a, b, and c, 4.8.3.1)

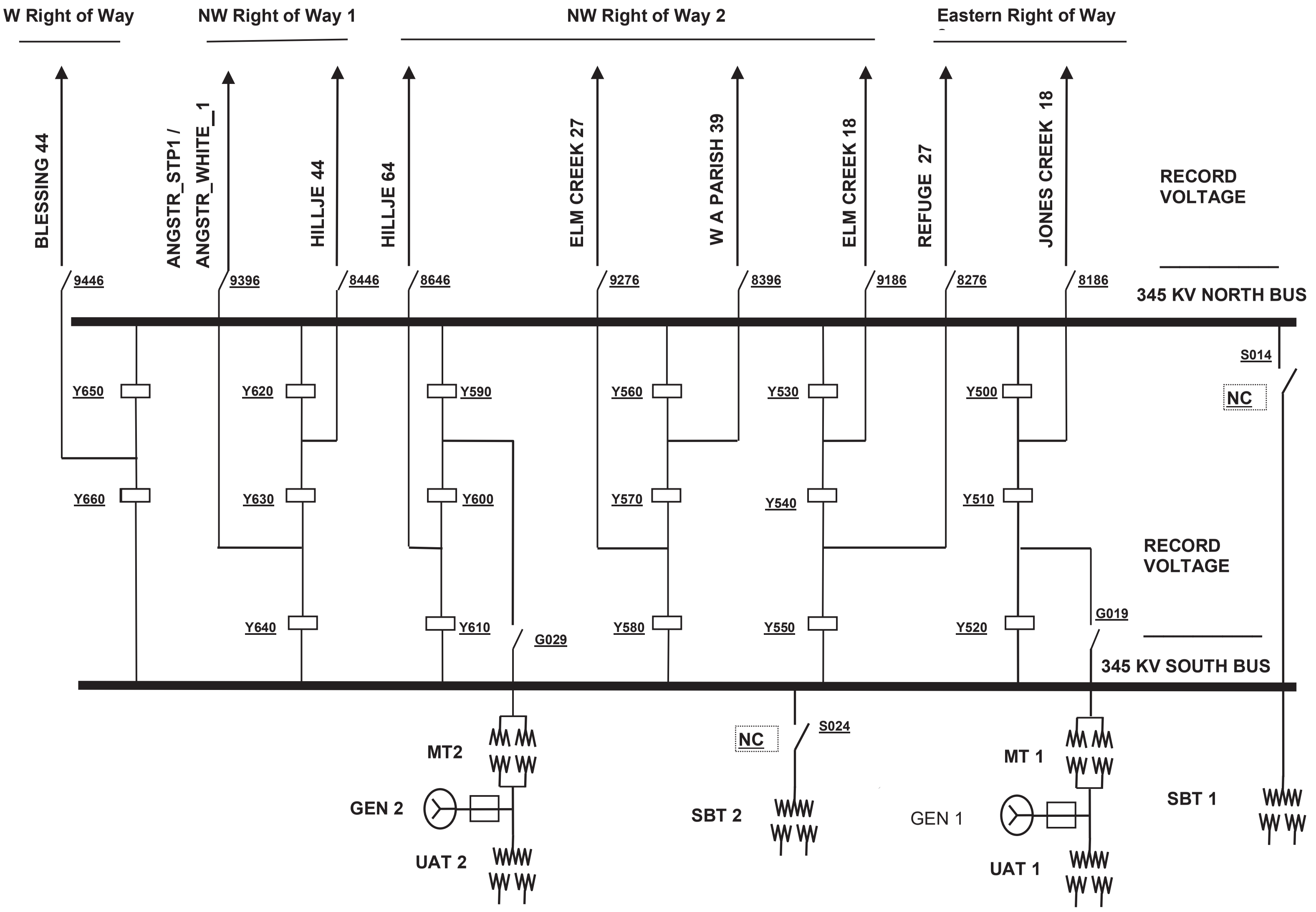
NOTE

IF notification has been received from the STP Coordinator or TDSP that a degraded voltage condition (less than 339 KV) could exist on the 345 KV System following a trip of both units, THEN Technical Specification LCO 3.8.1.1.e SHALL be entered per 0POP04-AE-0005, Offsite Power System Degraded Voltage. (References 3.15, 8.8, 8.17 and 8.18)

6.4 Notification has **NOT** been received from the STP Coordinator or TDSP that a degraded voltage condition (less than 339 KV) could exist on the 345 KV system following a trip of both units (i.e., loss of all STPEGS generation). (Technical Specification 3.8.1.1.e, References 3.15, 8.8, 8.17 and 8.18)

6.5 IF notified by the TDSP that switchyard battery chargers are not operating, THEN LCO 3.8.1.1.e has been entered within 20 hours. (Technical Specification 3.8.1.1.e, Reference 8.2, 8.18, 3.17)

UNIT 1 **(Circle Unit Performing Data Sheet)** **UNIT 2**



345 KV N BUS (CP010 OR SCADA 345A)		
345 KV S BUS (CP010 OR SCADA 345B)		
NORTH AND SOUTH BUS IN SERVICE WITH VOLTAGE ≥ 339 KV (REFERENCE AC STEPS 6.1, 6.2 AND 6.4)	YES	NO
APPROVED 345 KV OFFSITE CIRCUITS (GDC 17)		
NW RIGHT OF WAY 1		
ANGSTR_STP1 / ANGSTR_WHITE_1	YES	NO
NW RIGHT OF WAY 2		
ELM CREEK 27	YES	NO
OR		
W A PARISH 39	YES	NO
OR		
ELM CREEK 18	YES	NO
EASTERN RIGHT OF WAY		
REFUGE 27	YES	NO
OR		
JONES CREEK 18	YES	NO
AT LEAST 2 RIGHTS OF WAY AVAILABLE	YES	NO

This procedure, when complete, SHALL be retained for five years.

Exam Bank No.: 3200 **Last used on an NRC exam:** Never

SRO Sequence Number: 80

A Unit 2 shutdown for a refueling outage is in progress.

- RCS Tavg is 415ºF
- The Safety Injection Accumulators were just isolated per 0POP03-ZG-0007, Plant Cooldown.

Subsequently, the following conditions are observed:

- Mechanical Auxiliary Building sump and radiation levels are rising.
- Pressurizer level is 6% and lowering rapidly.

The first action the Unit Supervisor should take following entry into …

A. 0POP04-RC-0006, Shutdown LOCA is to secure and isolate running RHR trains.

B. 0POP04-RC-0006, Shutdown LOCA is to stop ALL RCPs to prevent a forced loss of RCS inventory.

C. 0POP05-EO-EC12, LOCA Outside Containment is to secure and isolate running RHR trains.

D. 0POP05-EO-EC12, LOCA Outside Containment is to stop ALL RCPs to prevent a forced loss of RCS inventory.

Answer: B 0POP04-RC-0006, Shutdown LOCA is to stop ALL RCPs to prevent a forced loss of RCS inventory.

Exam Bank No.: 3200 **Source:** New **Modified From** N/A

K/A Catalog Number: W/E04 G2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

SRO Importance: 4.2 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must know the applicability of the EOPs vs. the off-normal, select the correct procedure to implement, and know the content and bases of the procedure.

STP Lesson: LOT 501.01 **Objective Number:** 92016

GIVEN plant conditions/symptoms, EVALUATE the conditions/symptoms and STATE whether or not the procedure should be used.
LOT 501.01 92108 - Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-RC-0006

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as 0POP04-RC-0006 is the correct procedure to enter and if the student confuses the actions for accident mitigation with that of 0POP05-E0-EC12 for a LOCA in the MAB.

B: CORRECT. Based on the given conditions a leak outside containment is occurring in Mode 3 with RCS pressure less than 1000 psig (SI Accumulators Isolated) making 0POP04-RC-0006 the correct procedure. Per Step 1, if PRZR level is less than 8%, all RCPs are stopped to prevent a forced loss of RCS inventory.

C: INCORRECT. Plausible as 0POP05-E0-EC12 would be the correct procedure to transition to within the EOP network and the action noted is directed in the procedure for a leak in the MAB.

D: INCORRECT. Plausible as 0POP05-E0-EC12 would be the correct procedure to transition to within the EOP network and the action is the correct one per the shutdown LOCA procedure.

Question Level: H **Question Difficulty** 3

Justification:

The student must analyze the plant conditions and determine the appropriate procedure and actions to take.

PURPOSE

This procedure provides necessary operator actions for protecting the Reactor Core in the event of a Loss of Coolant Accident (LOCA) occurring during operation in Mode 3 with RCS pressure less than 1,000 psig or in Mode 4. Applicability is with RCS pressure less than 1,000 psig because this is when the accumulators are isolated.

This procedure will accomplish the following major actions:

- Isolate RHR and letdown
- Manually initiate charging and safety injection as necessary
- Prepare for and initiate RCS cooldown
- Depressurize RCS to refill Pressurizer
- Reduce injection flow
- Depressurize RCS to minimize subcooling

SYMPTOMS OR ENTRY CONDITIONS

The following are symptoms of a Shutdown LOCA:

- Uncontrolled drop in Pressurizer level
- Rising Containment sump levels
- Rising Containment radiation levels {RM11 or RM23}
- Rising MAB or FHB radiation levels {RM11 or RM23}
- RCS subcooling LESS THAN 35°F without a loss of secondary heat sink
- Rising Containment pressure
- Safety Injection actuation on Hi Containment pressure
- Rising Containment temperature
- Rising Containment dew point

NOTE

IF any of the following conditions are met, THEN USE Adverse Containment Values:

- Containment pressure greater than or equal to 5 PSIG
- Containment radiation levels greater than or equal to 10^5 R/HR
- Containment integrated radiation dose greater than or equal to 10^6 RADS

NOTE

- Any loss or any potential loss of fuel clad or RCS may require entry into the Emergency Plan. Refer to 0EPR04-UA-0001, EAL Wall Charts.
- Excessive RCS leakage may require entry into the Emergency Plan. Refer to 0EPR04-UA-0001, EAL Wall Charts.

_____ **1.0 CHECK if RCPs Must be Stopped**

a. Check any the following conditions met:

- RCS Subcooling based on CETs LESS THAN 35°F(45°F)

  OR

- PRZR Level – LESS THAN 8%(44%)

  OR

- Number 1 Seal DP – LESS THAN 220 PSID

  OR

- Number 1 Seal Leakoff Flow on each RCP – Below MINIMUM FLOW REQUIRED PER **ADDENDUM 5**

b. STOP ALL RCPs

Response not obtained:

a. GO TO Step 2.0

STEP DESCRIPTION FOR 0POP04-RC-0006 STEP 1.0

STEP: CHECK if RCPs Must be Stopped

PURPOSE: To secure the RCPs to prevent force loss of inventory

BASIS: Tripping the RCPs in shutdown LOCA conditions produces the same benefits as this action accomplishes for the at-power SBLOCA transient. These benefits include:

- Reduces mass loss throughout the transient. When the RCPs continue to operate, the RCS fluid conditions stay in quasi-homogenous state. As such, the break flow remains a low-quality two-phase mixture in lieu of a single-phase vapor even beyond the point when the collapsed liquid level in the system falls below the break elevation.
- Reduces heat addition to the RCS. When operating, each RCP can add several MW of heat energy to the RCS. This is additional heat energy that must dissipated by the break and/or steam generators. Higher ECCS flows are required to effectively deal with this additional energy.
- Protects RCPs and RCS from vibrational damage. Under high void fraction conditions, the pumps can become susceptible to vibration. While the RCPs are robustly designed, they have not been tested under these conditions. As such, it is unknown how long they will be able to operate under those conditions. In addition, the magnitude of the vibrations and their potential effect on other RCS components or piping is unknown.

Because the stored energy in the core is low in Modes 3 / 4, as is decay heat, the need for forced circulation is reduced. Natural circulation will be sufficient regardless if the RHR system is in service or not. As such there appears to be no significant negative consequences to trip the RCPs under Mode 3 / 4 LOCA conditions.

The current bases for RCP trip criteria for the at-power SBLOCA transient are presented in the RCP Trip Generic Issue in the ERG Executive Volume. The main focus in developing the criteria was to obtain parameters that directed RCP trip only when necessary (SBLOCA) but yet allowed continued operation in situations where it is advantageous to maintain forced circulation (SG tube rupture, steam line break, etc.). As such, three parameters were developed; any one of which could be utilized with the net result being that RCS mass is preserved in the SBLOCA transient. The three parameters were RCS pressure, hot leg subcooling and primary-to-secondary differential pressure. It is noted that the basis for criteria in the at-power SBLOCA transient does not apply to the shutdown situations. It is also noted for the at-power transient, an additional caveat to tripping the RCPs is that at least one HHSI pump must be running. High emphasis was placed on continued RCP operation when HHSI was not in service. This condition at least serves as a surrogate for core cooling until HHSI can be restored since this is outside the design basis of the plant. However, for the circumstances in the confines of shutdown LOCA space, it is possible that a HHSI pump may not be running due to reliance on manual SI. As such, the criterion of at least one HHSI pump running will not be included for RCP trip at shutdown conditions.

Exam Bank No.: 3203 **Last used on an NRC exam:** Never

SRO Sequence Number: 81

A loss of all Main and Auxiliary feedwater has occurred on Unit 2.

- The crew has entered 0POP05-EO-FRH1, Response to Loss of Heat Sink and have identified the need for RCS bleed and feed.

> **CAUTION**
>
> Steps 11 through 14 SHALL be performed quickly IN ORDER to establish RCS heat removal by RCS bleed and feed.

The operational implication of the caution above is that RCS bleed and feed must be established within several minutes to ______(1)______.

If both PORVs are not available for bleed and feed the Unit Supervisor should direct opening ____(2)____ head vent valve(s).

A. (1) minimize core uncovery
(2) only one

B. (1) prevent exceeding RCS pressure safety limit
(2) only one

C. (1) minimize core uncovery
(2) both

D. (1) prevent exceeding RCS pressure safety limit
(2) both

Answer: C (1) minimize core uncovery (2) both

Exam Bank No.: 3203 **Source:** New **Modified From** N/A

K/A Catalog Number: W/E05 G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes

SRO Importance: 4.3 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must know the content of the procedure to ensure adequate bleed and feed.

STP Lesson: LOT 504.33 **Objective Number:** 83013

Given a step, note or caution from 0POP05-EO-FRH1, STATE its basis.

Reference: LOT 504.33 Power Point slide 36; 0POP05-EO-FRH1; WOG ERG Background Document for FRH1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the student could believe that a head vent valve is a one for one replacement for a PZR PORV. The first part is correct.

B: INCORRECT. Plausible because the RCS pressure does rise as the RCS heats up but the PORVs and PZR safeties have the capacity to prevent exceeding RCS pressure safety limit and the student could believe that a head vent valve is a one for one replacement for a PZR PORV.

C: CORRECT. Per the 0POP05-EO-FRH1 basis, the caution prior to the steps to initiate bleed and feed alert the operator that the following steps must be completed within several minutes to minimize core uncovery and prevent an inadequate core cooling condition.

D: INCORRECT. Plausible because the RCS pressure does rise as the RCS heats up but the PORVs and PZR safeties have the capacity to prevent exceeding RCS pressure safety limit. The second part is correct.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of the operational significance of EOP steps, notes and cautions.

NOTE

- o Low pressure feedwater source should <u>NOT</u> be used unless other sources are unavailable.
- o Bleed and Feed should <u>NOT</u> be initiated due to low level in SGs being depressurized, unless CET are above 567 °F and rising.
- o Step 9 should be repeated if CETs rising.

____9 TRY TO ESTABLISH Feed Flow From Any Available Low Pressure Source Per ADDENDUM 8, ESTABLISHING LOW PRESSURE FEEDWATER SOURCE

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____10	CHECK For Loss Of Secondary Heat Sink: o SG wide range level in at least two SGs - LESS THAN 50% [73%] OR o Pressurizer pressure - GREATER THAN 2335 PSIG	RETURN TO Step 1. OBSERVE Caution and Notes Prior to Step 1.

CAUTION

Steps 11 through 14 SHALL be performed quickly IN ORDER to establish RCS heat removal by RCS bleed and feed.

____11 ACTUATE SI

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____14	**VERIFY Adequate RCS Bleed Path:** o Both pressurizer PORVs - OPEN o Both pressurizer PORV isolation valves - OPEN	PERFORM the following: a. OPEN reactor vessel head vent valves: o "ISOL HV-3658A" o "ISOL HV-3658B" b. OPEN reactor vessel head vent valves: o "ISOL HV-3657A" o "ISOL HV-3657B" c. OPEN reactor vessel head vent valves: o "HEAD VENT THROT VLV HCV-0601" o "HEAD VENT THROT VLV HCV-0602" -------------------------------------
____15	**DETERMINE If 0POP05-EO-EO00, REACTOR TRIP OR SAFETY INJECTION, Steps 1 Through 5 - HAVE BEEN COMPLETED**	PERFORM ADDENDUM 4, VERIFICATION OF SI EQUIPMENT OPERATION. -------------------------------------

CAUTION

The RCS bleed path SHALL be maintained even if RCS pressure remains GREATER THAN HHSI pump shutoff head.

____16	**MAINTAIN RCS Heat Removal:**	
	____a. MAINTAIN SI flow and charging flow	
	____b. MAINTAIN both pressurizer PORVs - OPEN	b. IF using reactor vessel head vents, THEN MAINTAIN head vents OPEN. -------------------------------------

STEP DESCRIPTION TABLE FOR FR-H.1 Step 11 - CAUTION

CAUTION: Steps 11 through 17 must be performed quickly in order to establish RCS heat removal by RCS bleed and feed.

PURPOSE: To alert the operator to complete the steps quickly for establishing RCS bleed and feed to ensure effectiveness of the heat removal method

BASIS:

Once the operator detects that secondary heat sink has degraded, RCS bleed and feed must be established within several minutes to prevent or minimize core uncovery.

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

CAUTION

Steps 11 through 14 SHALL be performed quickly in order to establish RCS heat removal by RCS bleed and feed

BASIS:

Once secondary heat sink has degraded, RCS bleed and feed must be established within several minutes to prevent or minimize core uncovery.

Exam Bank No.: 3220 **Last used on an NRC exam:** Never

SRO Sequence Number: 82

Unit 1 is at 100% power when the following occur:

- 5M03-A-3 PR UPPER DET FLUX DEV HI/AUTO DEF, alarms.
- 5M03-B-3 PR LOWER DET FLUX DEV HI/AUTO DEF, alarms.
- 5M03-C-3 PR CHANNEL DEV, alarms.
- 5M03-D-5 ROD SUPV MNTR ROD POSITION TRBL, alarms.
- DRPI indicates an abnormal rod configuration (Refer to Attachment)

The Unit Supervisor implements 0POP04-RS-0001, Control Rod Malfunction.

To respond to this event, the Unit Supervisor should direct the performance of _____(1)_____ .

If the rod can be recovered within ____(2)____ hour(s), then a power reduction is **NOT** required.

A. (1) Addendum 1, Recovery of a Dropped Rod
(2) 1

B. (1) Addendum 2, Recovery of Misaligned Rods
(2) 4

C. (1) Addendum 1, Recovery of a Dropped Rod
(2) 4

D. (1) Addendum 2, Recovery of Misaligned Rods
(2) 1

Answer: A (1) Addendum 1, Recovery of a Dropped Rod (2) 1

Exam Bank No.: 3220 **Source:** New **Modified From**

K/A Catalog Number: APE 003 AA2.03 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements .

SRO Importance: 3.8 **Tier:** 1 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must know the content of an addendum within 0POP04-RS-0001.

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-RS-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☒ **Attachment:** DRPI Picture

Distractor Justification

A: CORRECT: The symptoms point to a dropped rod and Addendum 1 would be chosen to proceed. If the recovery occurs within 1 hour, power does not need to be reduced.

B: INCORRECT: Plausible as some of these symptoms occur for a misaligned rod also and 4 hours is a common TS action time and is listed to prevent further power reduction in 0POP04-RS-0001.

C: INCORRECT: Plausible as this is a correct diagnosis and 4 hours is a common TS action time and is listed to prevent further power reduction in 0POP04-RS-0001.

D: INCORRECT: Plausible as some of these symptoms occur for a misaligned rod also and this is the correct action time.

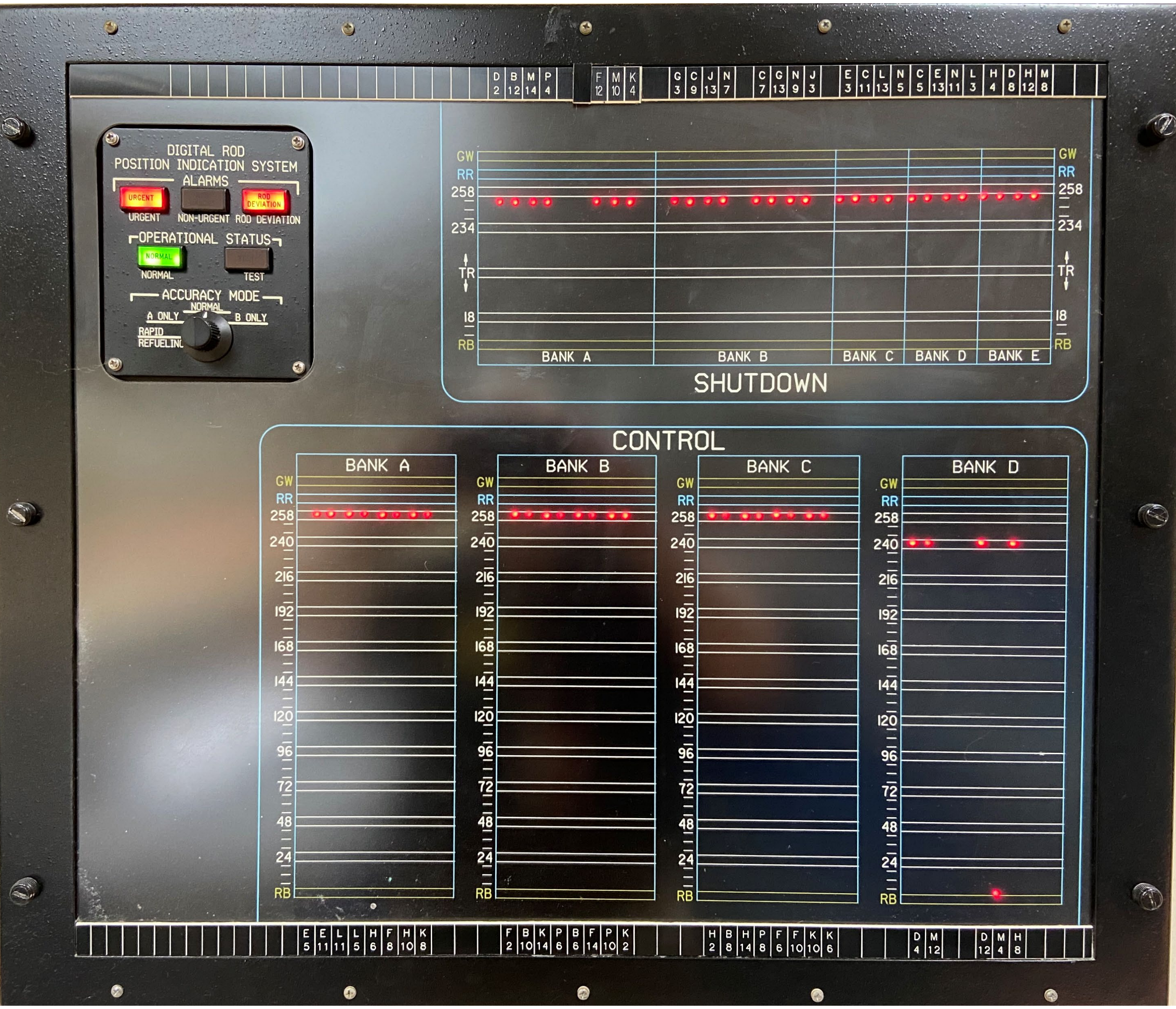
Question Level: H **Question Difficulty** 3

Justification:

The student must assess conditions and select an addendum with which to proceed.

Q82 Attachment

Clarification given that Rod M4 indicates < 6 steps.



NOTE

Steps 1.0 through 3.0 are IMMEDIATE ACTION Steps.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
______ **1.0**	**ENSURE “ROD BANK SEL” Switch In “MAN” {CP005}**	
______ **2.0**	**VERIFY All Rods – NO ROD MOTION**	PERFORM the following: **a.** TRIP the Reactor. **b.** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
______ **3.0**	**CHECK For Dropped Rods:**	
______	**a. CHECK All Rods – ANY RODS DROPPED**	**a.** GO TO Step 4.0.
______	**b. CHECK All Rods – ONLY ONE ROD DROPPED**	**b.** IF in Modes 1 OR 2, THEN PERFORM the following: **1)** TRIP the Reactor. **2)** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
______	**c. GO TO Addendum 1, Recovery of a Dropped Rod**	

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ **4.0**	**CHECK For Misaligned Rods:**	
_____ **a.**	**CHECK All Rods – ANY RODS MISALIGNED by GREATER THAN 12 INDICATED steps**	**a.** IF the only MISALIGNED Rods are LESS THAN OR EQUAL TO 12 INDICATED steps, THEN GO TO Addendum 2, **Recovery of Misaligned Rods**. **b.** IF No Rods are MISALIGNED, THEN GO TO Step 5.0.
_____ **b.**	**CHECK All Rods – ONLY ONE ROD MISALIGNED by GREATER THAN 12 INDICATED steps**	IF two or more rods are misaligned by GREATER THAN 12 INDICATED steps, THEN: **a.** REFER TO Technical Specification 3.1.3.1 Action d for appropriate action. **b.** COMMENCE load reduction per 0POP03-ZG-0006, Plant Shutdown From 100% To Hot Standby, to place the unit in Mode 3 within six hours of the time of misalignment. **c.** GO TO Step 5.0.
_____ **c.**	**GO TO Addendum 2, Recovery of Misaligned Rods**	
_____ **5.0**	**CHECK Reactor Trip Breakers – CLOSED**	IF all rods have **NOT** fully inserted following a Reactor Trip or shutdown, THEN GO TO Addendum 3, Insertion of Rods Which Fail to Fully Insert Following Reactor Trip or Shutdown.

NOTE

The following step is intended to assess the practical ability to recover the dropped rod in a timely manner. Unless required I&C personnel are already involved and timely rod recovery is considered highly likely, a power reduction should be commenced as soon as practical to achieve LESS THAN 75% within one hour of the Dropped rod.

- IF the rod can be re-aligned within 1 hour, THEN power reduction is not required.
- IF rod recovery will require more than one hour, THEN a power reduction to less than 75% is required.
- IF rod is NOT fully recovered (i.e. returned to its required position) within one hour above 75% power, THEN a power reduction to less than 40% is required.
- IF rod is NOT fully recovered (i.e. returned to its required position) within four hours, THEN a power reduction to less than 40% is required prior to rod recovery.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	6.0	**DETERMINE If Reactor Power Reduction Is Required**	GO TO Step 15.0 of this Addendum. --

NOTE

Following a control rod malfunction, I&C troubleshooting and diagnosis will be significantly enhanced if control rods are **NOT** moved. Therefore rods should **NOT** be moved unless rod motion is required to maintain Reactor control.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
			--
_____	7.0	**REDUCE Reactor Power To Less Than 75% Per 0POP03-ZG-0006, Plant Shutdown From 100% To Hot Standby Or 0POP03-ZG-0008, Power Operations**	
_____	8.0	**CHECK Reactor Power – LESS THAN 75%**	PERFORM the following: **a.** CONTINUE power reduction to less than 75%. **b.** WHEN Reactor Power is less than 75%, THEN GO TO Step 9.0 of this Addendum. --

Exam Bank No.: 3211 **Last used on an NRC exam:** Never

SRO Sequence Number: 83

Unit 1 is in MODE 6 and a core offload is in progress.

- SR Channel NI-31 is supplying Audio Count Rate in the Control Room and Containment.
- Power to DP-1202 swaps to the Voltage Regulating Transformer due to a failure of the associated inverter.

SR Channel NI-32 is ______(1)______.

If the Audio Count Rate amplifier were to lose power the action would be to _____(2)_____.

A. (1) operable
(2) suspend core alterations or operations that would introduce coolant to the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1

B. (1) operable
(2) suspend Core Alterations ONLY

C. (1) inoperable
(2) suspend core alterations or operations that would introduce coolant to the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1

D. (1) inoperable
(2) suspend Core Alterations ONLY

Answer: A (1) operable
(2) suspend core alterations or operations that would introduce coolant to the RCS with boron concentration less than required to meet the boron concentration of LCO 3.9.1

Exam Bank No.: 3211 **Source:** New **Modified From**

K/A Catalog Number: APE 032 G2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

SRO Importance: 4.2 **Tier:** 1 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(2)

SRO Justification:

Student must have knowledge of the bases of Technical Specifications.

STP Lesson: LOT 503.01 **Objective Number:** 92102

Given the topic or title of a specification in the Technical Specifications, describe the general requirements of the specification to include components or administrative requirements affected, limitations, major time frames involved, major surveillance in order to comply, and the bases for the specification/requirement.

Reference: Tech Spec 3.9.2 and bases, TS Bases Table 3.8-1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. NI-32 power is not required to be in its normal configuration per TS Bases Table 3.8-1, therefore NI-32 remains operable per TS 3.9.2. If the Audio Count Rate Amplifier were to lose power it would affect the ability of either channel of Source Range to provide audible count rate in the Control Room or Containment.

B: INCORRECT. Plausible because suspending core alterations is part of the the action however the spec also restricts the water sources that can be introduced to the RCS for these conditions. The first part is correct.

C: INCORRECT. Plausible if student is unaware of the reference that the Bases for TS 3.9.2 makes to the table in the bases of section 3.8. The second part is correct.

D: INCORRECT. Plausible if student is unaware of the reference that the Bases for TS 3.9.2 makes to the table in the bases of section 3.8 and because suspending core alterations is part of the the action however the spec also restricts the water sources that can be introduced to the RCS for these conditions

Question Level: H **Question Difficulty** 3

Justification:

Student must assess conditions and apply TS actions and apply TS Bases.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors* shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION, excluding the Neutron detectors, at a frequency in accordance with the Surveillance Frequency Control Program.

---

* An Extended Range Neutron Flux Monitor may be substituted for one of the Source Range Neutron Flux Monitors provided the OPERABLE Source Range Neutron Flux Monitor is capable of providing audible indication in the containment and control room.

TABLE 3.8-1

AC AND DC POWER REQUIREMENTS – SHUTDOWN

Supported Equipment	**Related Technical Specifications (TS)**	**Applicable Mode**	**Power Source Requirements** [Note 8]
Power Operated Relief Valves and Cold Overpressure Mitigation System	TS 3.4.9.3	5, 6 (with head on reactor vessel)	Operability not dependent on EDG operability. Channels I and II (Train A) & III (Train B) must be operable. One train must be powered by its associated operable Class 1E power system (batteries, inverters and chargers). [Note 2]
Source Range Neutron Monitors	TS 3.3.1, Table 3.3-1 Functional Unit 6.B	5	Operability not dependent on EDG operability. Channels I and II (Train A) must be operable. One Source Range channel must be powered by its associated operable Class 1E power system (batteries, inverters and chargers). [Note 3] [Note 4]
	TS 3.9.2	6	Operability not dependent on EDG operability. Channels I and II (Train A) must be operable. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]
Extended Range Neutron Monitors	TS 3.3.1, Table 3.3-1 Functional Unit 7	5	Operability not dependent on EDG operability. Channels I (Train A) and IV (Train C) must be operable. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]
	TS 3.9.2	6	Operability not dependent on EDG operability. Channel I (Train A) or IV (Train C) must be operable, if used to substitute for a Source Range NI in accordance with TS 3.9.2. Emergency power and associated normal Class 1E power are not required for operability. [Note 4]

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range and/or Extended Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core. AC and DC power requirements-shutdown for OPERABILITY of Source Range and Extended Range Neutron Flux instrumentation are given in Bases Table 3.8-1.

ACTION a. requires suspending the introduction into the RCS of coolant with boron concentration less than required to meet the refueling boron concentration limit necessary to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes, including temperature increases when operating with a positive moderator temperature coefficient, must also be evaluated to not result in operation below the required refueling boron concentration limit. Control rod withdrawal is not allowed except that it is permissible to unlock the control rods for rapid refueling. To unlock the control rods, they must be withdrawn at least one step. However, since the control rods are above the active fuel when the unlocking process occurs, there is no reactivity addition.

3/4.9.3 (Not Used)

3/4.9.4 (Not Used)

Exam Bank No.: 2783 **Last used on an NRC exam:** 2019

SRO Sequence Number: 84

A reactor startup is in progress.

- Source range channel N31 indicates 3.0 E+4 cps.
- Source range channel N32 indicates 3.0 E+4 cps.
- Intermediate range channel N35 indicates 9.5 E-11 amps.
- Intermediate range channel N36 indicates 2.0 E-11 amps.

Complete the following statements:

Based on the indications shown above, _______(1)______.

Power for this condition must be limited to ______(2)______.

A. (1) N35 is undercompensated
(2 the lower of 5.0 E+4 cps or the P-6 setpoint

B. (1) N35 is undercompensated
(2) 10%

C. (1) N36 is overcompensated
(2) the lower of 5.0 E+4 cps or the P-6 setpoint

D. (1) N36 is overcompensated
(2) 10%

Answer: C (1) N 36 is overcompensated (2) the lower of 5.0 E+4 cps or the P-6 setpoint.

Exam Bank No.: 2783 **Source:** Bank **Modified From** N/A

K/A Catalog Number: APE 033 AA2.04 Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Satisfactory overlap between source-range, intermediate-range and power-range instrumentation

SRO Importance: 3.6 **Tier:** 1 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(2)

SRO Justification:

Facility Operating Limitations in the Technical Specificatons and their Bases: Applicaton of required actions (TS Section 3)

STP Lesson: LOT 503.01 **Objective Number:** 80056

Determine the applicable Technical Specification and/or Technical Requirements Manual (TRM) Limiting Conditions for Operation (LCOs) and the required actions to be taken.

Reference: Technical Specification 3.3.1, Functional Unit 5, Action 3, 0PSP03-ZQ-0028, page 28, 0POP03-ZG-0004, p. 24

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible if the student confuses the effect of compensating voltage. Part 2 is correct.

B: INCORRECT: Plausible if the student confuses the effect of compensating voltage and if the student believes that limiting power to 10% is the correct action per TS.

C: CORRECT: IR 35 is reading correctly for the given SR readings. Since IR 36 is reading lower by a factor of almost 4, it is overcompensated and does not meet the channel check of less than or equal to a factor of 3 between readings. It also does not meet the overlap requirements of 0POP03-ZG-0004, Reactor startup. With an inoperable IR instrument during a reactor startup with power < P-6, power must be maintained below the P-6 setpoint of 1.0 E-10 amps per the Tech Spec 3.3.1 action 3 but since it also does not meet the overlap requirements of the POP03 power can not exceed 5.0 E+4 cps per the procedure.

D: INCORRECT:Plausible as this is the correct failure and if the student believes that moving power below P-6 is the correct action per TS.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the conditions presented regarding SR and IR instruments and determine the failure. Then, the student must apply TS to the situation.

Initials

CAUTION

- Control Banks SHALL be inserted, as required to stabilize count rate, until NI overlap is verified.
- 5.0E4 cps SHALL **NOT** be exceeded on EITHER SR Channel UNTIL the following conditions are met:
  - ○ Overlap is achieved on both IR Channels
  - ○ BOTH P-6 bistables are received
  - ○ BOTH SR HI-Flux Trips are blocked
- Source Range High Voltage SHALL be immediately blocked after verifying P-6. (SR Hi-Flux Reactor Trip is at 1.0E5 cps)
- IF P-6 is **NOT** verified on both Intermediate Range Channels, THEN 5.0E4 cps SHALL **NOT** be exceeded and Control Banks SHALL be used as necessary to stabilize power until P-6 is verified on both Intermediate Range Channels.

6.22 DO **NOT** exceed 5.0E4 cps until Source Range Reactor Trips are blocked. _____

6.23 WHEN both Intermediate Range power levels are greater than or equal to 5E-11 amps, AND Both Source Range NI's are indicating (i.e., SR Trips **NOT** blocked), THEN VERIFY Intermediate Range overlap. _____

6.24 WHEN Intermediate Range Channels are approximately 1.0E-10 amps, THEN VERIFY P-6 permissive lights illuminated. _____

• BSMP Status Light	"P6 IR CH1"
• BSMP Status Light	"P6 IR CH2"
• Permissive Lampbox (5M24 A-1)	"P6 SOURCE RANGE RX TRIP BLOCK PERM"

6.25 BLOCK Source Range Hi Flux Reactor Trip by turning BOTH "SR TRN R" and "SR TRN S" switches momentarily to BLOCK. _____

- "SR TRN R"
- "SR TRN S"

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Chapter 16 in the Updated Final Safety Analysis Report (UFSAR).

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1 and at the frequency specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit at a frequency in accordance with the Surveillance Frequency Control Program. Each verification shall include at least one train such that both trains are verified at a frequency in accordance with the Surveillance Frequency Control Program and one channel per function such that all channels are verified at least once every N times the frequency specified in the Surveillance Frequency Control Program where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1###, 2	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3*, 4*, 5*	10
7. Extended Range, Neutron Flux	2	0	2	3, 4, 5	5
8. Overtemperature ΔT	4	2	3	1, 2	6
9. Overpower ΔT	4	2	3	1, 2	6
10. Pressurizer Pressure -- Low (Interlocked with P-7)	4	2	3	1	6
11. Pressurizer Pressure-- High	4	2	3	1, 2	6
12. Pressurizer Water Level--High (Interlocked with P-7)	4	2	3	1	6

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 72 hours, or immediately suspend all operations involving positive reactivity changes.

Note: Plant temperature changes or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

b. With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement,

Immediately suspend all operations involving positive reactivity changes,

AND

Within 15 minutes isolate unborated water flow paths from the reactor makeup water system to the reactor coolant system,

AND

Perform either of the following:

Restore at least one channel to OPERABLE status within 1 hour,

OR

1. Within 2 hours secure each unborated water flow path to the reactor coolant system by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured,

   AND

2. Within 4 hours and once per 12 hours thereafter, verify SHUTDOWN MARGIN is within limits.

Note: Operations involving plant temperature changes may proceed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

UNIT: _____ DATE: ____________

PARAMETER	LOCATION	INSTR	00-02	12-14	LIMIT	TECHNICAL SPECIFICATION	MODE	NOTE
AXIAL FLUX DIFFERENCE	CP-005	NI-0041C			+5,-10% OF TARGET AFD	3.2.1	MODE 1 > 15% POWER	(1) IF ALARM IS INOP, THEN PERFORM LOGSHEET 4. (2) IF ANY CONTROL BOARD AXIAL FLUX DIFFERENCE INDICATOR IS INOPERABLE, THEN THE "POWER RANGE NEUTRON FLUX" CHART RECORDER BLUE "DELTA V" DIGITAL DISPLAY FOR PEN 3 MAY BE USED TO CALCULATE THE VALUE FOR EACH CHANNEL USING THE EQUATION FOR DELTA-I% IN FIGURE 5.1 OF THE PLANT CURVE BOOK. (CP-018) (Channel I, NR-0041 for NI-0041C) (Channel II, NR-0042 for NI-0042C) (Channel III, NR-0043 for NI-0043C) (Channel IV, NR-0044 for NI-0044C)
		NI-0042C						
		NI-0043C						
		NI-0044C						
	AFD HI ANNUNC WINDOW	5M3-D3			ALARM OPERABLE (1)			
SOURCE RANGE FLUX	CP-005	NI-0031B			CPS CHNL CHECK (1)	3.3.1 Table 3.3-1, Item 6 Actions 4,10 Item 5, Action 3	2,3,4 ONLY	(1) ≤ FACTOR OF 3 BETWEEN READINGS.
		NI-0032B						
INTERMED. RANGE FLUX		NI-0035B			AMPS CHNL CHECK (1)		1,2 ONLY	
		NI-0036B						
RCP A IND LIGHTS	CP-005	IND LIGHTS			POWER AVAIL (1) (2)	3.4.1.2	3,4 ONLY	(1) IF NO OPERATIONS ARE PERMITTED WHICH CAUSE DILUTION AND CORE OUTLET TEMP IS MAINTAINED AT LEAST 10°F SUBCOOLED, THEN ALL RCPs AND RHR PUMPS MAY BE DEENERGIZED FOR UP TO 1 HR. (2) IN MODE 3 AT LEAST 2 RCPs SHALL HAVE POWER. IN MODE 4 AT LEAST 2 OF ANY COMBINATION OF RCPs AND/OR RHR PUMPS SHALL HAVE POWER.
RCP B IND LIGHTS								
RCP C IND LIGHTS								
RCP D IND LIGHTS								

Exam Bank No.: 3221 **Last used on an NRC exam:** Never

SRO Sequence Number: 85

The unit is at 100% power.

- A Loss of Offsite Power occurred.

The crew is performing 0POP05-EO-ES02, Natural Circulation Cooldown. Complete the following:

If NO CRDM Vent Fans are available, then the MAXIMUM allowed cooldown rate is __(1)__ºF/hour.

The basis of this cooldown rate limit is to minimize ________(2)________ in the reactor vessel head.

A. (1) 100
(2) void formation

B. (1) 100
(2) brittle fracture

C. (1) 70
(2) void formation

D. (1) 70
(2) brittle fracture

Answer: C (1) 70 (2) void formation

STP LOT-26 NRC SRO EXAM

Exam Bank No.: 3221 **Source:** New **Modified From**

K/A Catalog Number: W/E09 G2.4.18 Knowledge of the specific bases of EOPs.

SRO Importance: 4.0 **Tier:** 1 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(1)

SRO Justification:

The student is required to know the specific mitigative strategy during a Natural Circulation Cooldown and the basis for this strategy.

STP Lesson: LOT 504.25 **Objective Number:** 92228

STATE the basis for maximum cooldown rate associated with natural circulation cooldown.

Reference: 0POP05-EO-ES02, WOG ERG ES-0.2, LOT 504.25

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as this is a common cooldown limit in procedures and in Technical Specifications and brittle fracture is a concern with rapid cooldown in the vessel.

B: INCORRECT: Plausible as this is a common cooldown limit in procedures and in Technical Specifications and this is the correct reason for the limit.

C: CORRECT: With no CRDM fans available, the cooldown rate is lower at 70F. This is to minimize vessel head voiding.

D: INCORRECT: Plausible as this is the correct limit, and brittle fracture is a concern with rapid cooldown in the vessel.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall information from a procedure.

Step 11 continued from previous page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		3) IF SG PORV(s) will NOT operate from the control room, THEN PERFORM the following: a) ENSURE SG PORV(s) in manual. b) DEPRESS and HOLD SG PORV(s) down arrow pushbutton for GREATER THAN 20 SECONDS. c) DISPATCH operator to operate SG PORV(s) per ADDENDUM 4, SG PORV LOCAL OPERATION. d) IF SG PORV can NOT be operated locally, THEN PERFORM ADDENDUM 7, ESTABLISHING DEAERATOR VENT PATH.
	___b. ESTABLISH RCS Cooldown Rate	
	___1) SG ALL Active And Intact	1) PERFORM the following: a) MAINTAIN cooldown rate in RCS cold legs BETWEEN 15°F/HR and 20°F/HR. b) GO TO Step 11.d
	___2) CRDM Vent Fans - at least one running	2) PERFORM the following: a) MAINTAIN cooldown rate in RCS cold legs LESS THAN 70°F/HR. b) GO TO Step 11.d.
	___c. MAINTAIN Cooldown Rate in RCS Cold Legs - LESS THAN 100°F/HR	

Step 11 continued on next page.

STEP DESCRIPTION TABLE FOR ES-0.2 Step 6

STEP: Verify All CRDM Fans - RUNNING

PURPOSE: To ensure that as much heat as possible is being removed from the vessel head

BASIS:

The results from several tests at domestic and foreign plants indicate that the Control Rod Drive Mechanism (CRDM) cooling fans aid significantly in removing heat from the upper head area. For this reason it is necessary to have as many CRDM cooling fans in operation as possible. If the CRDM cooling fans are not in operation, subsequent RCS cooldown/depressurization instructions are affected.

Refer to the DESCRIPTION section for a more detailed discussion of the heat removal capabilities of the CRDM cooling fans.

KNOWLEDGE:

- This step is a continuous action step.
- In subsequent steps, the ability to remove the postulated heat load assumed in the analysis (e.g., 456 kw for a 4-loop plant) is measured by percent of CRDM fan cooling. For example, "100% CRDM fan cooling" is defined as sufficient CRDM fans running to remove 100% of the postulated heat load. This should be determined based on plant design.
- If additional CRDM fans become available after this step is encountered, they should be started to enhance cooldown of the upper head. It may be possible to satisfy the plant-specific definition of 50% or 100% CRDM fan cooling by the time Step 14 is reached to allow less restrictions on the subsequent RCS cooldown and depressurization to RHR operating conditions.

PLANT-SPECIFIC INFORMATION:

N/A

STEP DESCRIPTION TABLE FOR ES-0.2 Step 7

STEP: Initiate RCS Cooldown To Cold Shutdown

PURPOSE: To begin a controlled RCS cooldown to cold shutdown at a specified maximum rate, with preferred and alternate methods

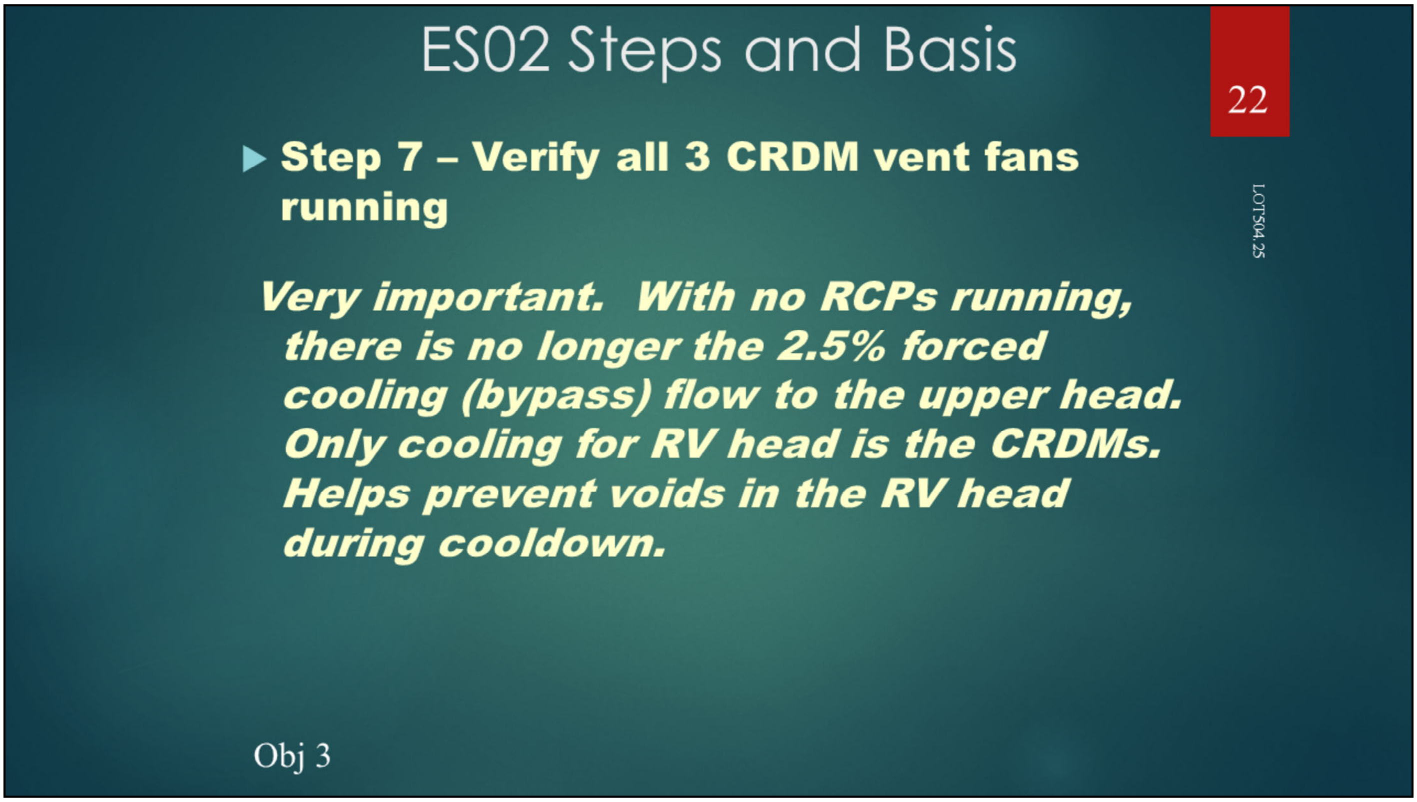
BASIS:

Prior to initiating the cooldown of the RCS, a check is made to determine if an inactive loop(s) exists. An inactive RCS loop exists if the capability to feed the respective SG and/or the capability to release steam from the respective SG is lost due to equipment failures.

If all loops are active, then steam should be released through the condenser steam dump valves. If the main condenser is not available for steam dump, the cooldown should be established by use of the steam generator power-operated relief valves, releasing steam to the atmosphere or other plant specific means. The cooldown rate should be controlled and maintained less than 100ºF/hr, except for T_{COLD} plants with less than 50% CRDM fan cooling for which a maximum cooldown rate of 70ºF/hr is imposed (refer to the DESCRIPTION section). Steam dump must be discontinued if the actual cooldown rate exceeds these permissible values.

If one or more of the RCS loops is inactive, a more restrictive cooldown rate may be imposed to prevent flow stagnation in the inactive loop(s). The analysis performed to address stagnant loop cases is documented in WCAP-16632. Information is also provided in the “Natural Circulation Cooldown with a Stagnant Loop” section of this background document. The maximum cooldown rate is dependent on the decay heat level of the core (based on time after trip) and the elevation from the bottom of the SG plenum to the top of the U-tube bend. The maximum allowable cooldown rate decreases as the SG elevation increases. Similarly, the maximum cooldown rate decreases as the decay heat level decreases.

The maximum cooldown rate is determined from Figure ES02-1. This is a generic curve which is used to determine the maximum cooldown rate versus the active loop(s) ΔT. Note that the active loop(s) ΔT is directly proportional to the decay heat level. Each utility must develop a plant specific curve for its plant from Figure ES02-1 and this curve would be included in the plant specific EOPs. This plant specific curve can be developed by modifying Figure ES02-1 as follows: The X-axis should be the lowest nominal or indicated ΔT (Thot - Tcold) of the active loop(s) without uncertainties, and decay heat values should not be included. In light of the conservatism inherent in Figures 14 through 19 in the DESCRIPTION section it is acceptable for plants to remove the approximate symbol (~) from the X-axis.



ES02 Steps and Basis

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LOT504.25

- **Step 7 – Verify all 3 CRDM vent fans running**

Very important. With no RCPs running, there is no longer the 2.5% forced cooling (bypass) flow to the upper head. Only cooling for RV head is the CRDMs. Helps prevent voids in the RV head during cooldown.

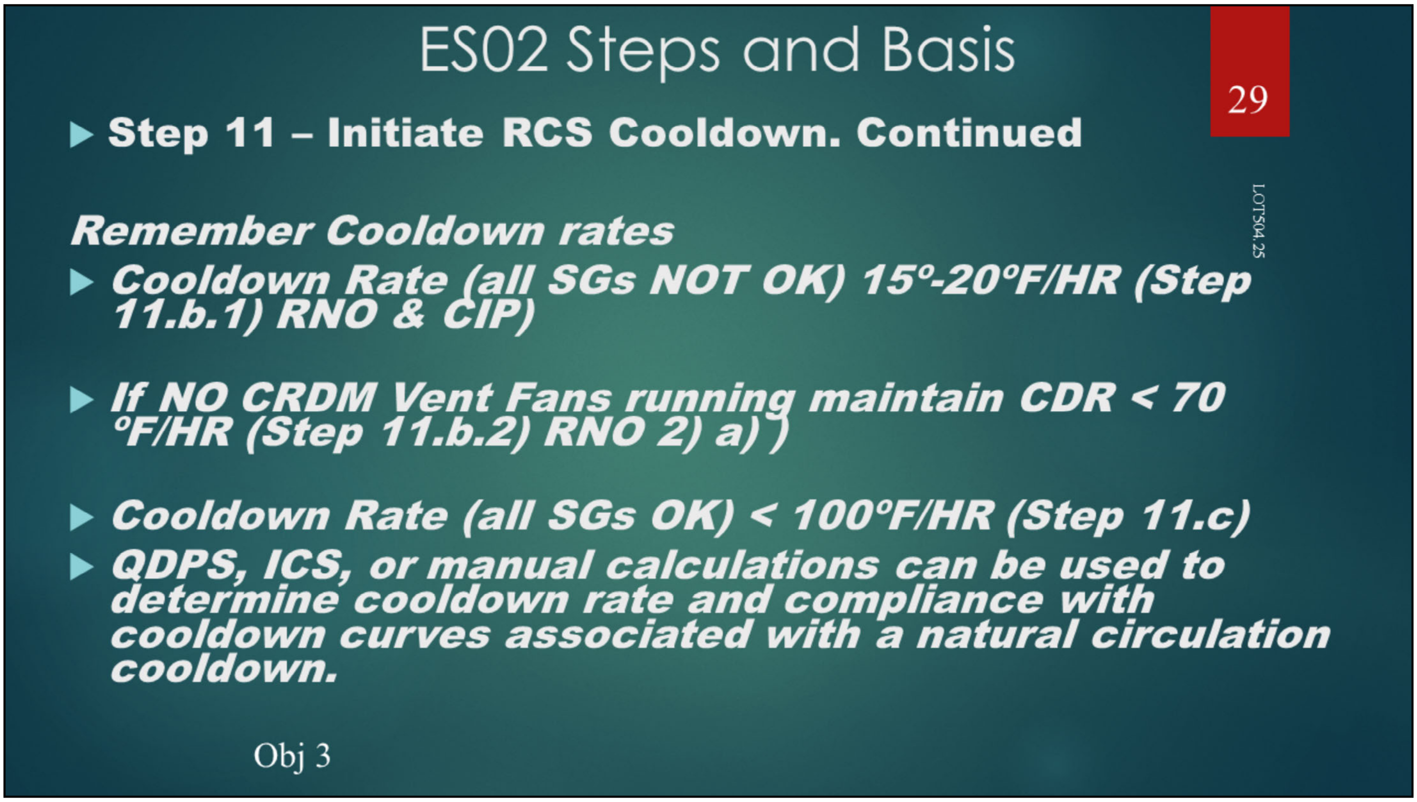
Obj 3

PURPOSE: To ensure that as much heat as possible is being removed from the vessel head

Control Rod Drive Mechanism (CRDM) cooling fans aid significantly in removing heat from the upper head area. For this reason it is necessary to have as many CRDM cooling fans in operation as possible. If the CRDM cooling fans are not in operation, subsequent RCS cooldown/depressurization instructions are affected.

The ability to remove the postulated heat load assumed in the analysis (e.g., 456 kw for a 4-loop plant) is measured by percent of CRDM fan cooling. For example, "100% CRDM fan cooling" is defined as sufficient CRDM fans running to remove 100% of the postulated heat load.

If additional CRDM fans become available after this step is encountered, they should be started to enhance cooldown of the upper head.



ES02 Steps and Basis

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LOT504.25

- **Step 11 – Initiate RCS Cooldown. Continued**

Remember Cooldown rates

- ***Cooldown Rate (all SGs NOT OK) 15°-20°F/HR (Step 11.b.1) RNO & CIP)***
- ***If NO CRDM Vent Fans running maintain CDR < 70 °F/HR (Step 11.b.2) RNO 2) a))***
- ***Cooldown Rate (all SGs OK) < 100°F/HR (Step 11.c)***
- ***QDPS, ICS, or manual calculations can be used to determine cooldown rate and compliance with cooldown curves associated with a natural circulation cooldown.***

Obj 3

MAINTAIN Cooldown Rate in RCS Cold Legs

LESS THAN 100°F/HR if everything's OK

LESS THAN 70°F/HR if there's NO CRDM fans running

Between 15-20 °F/HR if any SG is not intact

Exam Bank No.: 2774 **Last used on an NRC exam:** 2019

SRO Sequence Number: 86

The unit is at 100% power.

- The following alarms are in:
  - 4M07-C-1, RCP 1A NO 2 SEAL LKF FLOW HI
  - 4M07-B-2, RCP 1A STDPIPE LVL HI
  - 4M07-E-3, RCDT LEVEL HI-HI/LO-LO

These alarms are in because the RCP 1A _____(1)_____ has failed. The Unit Supervisor should direct the crew to perform _________(2)__________.

A. (1) #2 seal
(2) 0POP04-RC-0002, RCP Off Normal, Addendum 4, RCP #2 Seal Leakoff Flow High

B. (1) #2 seal
(2) 0POP04-RC-0003, Excessive RCS Leakage, Addendum 2, RCS Leakage to the RCDT

C. (1) #3 seal
(2) 0POP04-RC-0002, RCP Off Normal, Addendum 4, RCP #2 Seal Leakoff Flow High

D. (1) #3 seal
(2) 0POP04-RC-0003, Excessive RCS Leakage, Addendum 2, RCS Leakage to the RCDT

Answer: A (1) #2 seal (2) 0POP04-RC-0002, RCP Off Normal, Addendum 4, RCP #2 Seal Leakoff Flow High

Exam Bank No.: 2774 **Source:** Bank **Modified From** N/A

K/A Catalog Number: 003 A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems with RCP seals, especially rates of seal leak-off

SRO Importance: 3.9 **Tier:** 2 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. Knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps.

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given a plant condition, state the actions required to be performed per the applicable off-normal procedure.

Reference: 0POP04-RC-0002, pgs 15,39,40 0POP09-AN-04M7 pgs 26,27,45,46

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: With a #2 seal failure, the #2 seal passes more flow and will more quickly fill the RCDT and also can backflow into the RCP standpipe through the #3 seal. The proper procedure is 0POP04-RC-0002 and the addendum implemented will be Addendum 4. The user is directed to this procedure by 4M07-B2 and C1 alarm responses. 4M07-E-3, RCDT LEVEL HI-HI/LO-LO can be caused by an RCS leak, but with this combination of alarms, a #2 Seal Failure is indicated.

B: INCORRECT: Plausible if the student wrongfully equates seal leakage with RCS leakage. The first part is correct.

C: INCORRECT: Plausible as the standpipe provides water for #3 seal operation, but with a #3 seal failure the standpipe would drain more quickly or empty and a STANDPIPE LVL LO alarm would actuate. The second part is correct.

D: INCORRECT: Plausible as the standpipe does provide water for the #3 seal operation, but with a #3 seal failure the standpipe would drain more quickly or empty and a STANDPIPE LVL LO alarm would actuate, and the student could wrongfully equate seal leakage with RCS leakage.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the conditions given and choose a course of action.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ **8.0**	**CHECK All RCP Number 1 Seal Leakoff Flows - IN THE NORMAL OPERATING RANGE PER ADDENDUM 2**	PERFORM the following: **a.** IF Lampbox 4M08, Window E-3 "VCT PRESS HI/LO" is illuminated, THEN TAKE corrective action per 0POP09-AN-04M8, Annunciator Lampbox 4M08 Response Instructions. **b.** PERFORM Addendum 3.
_____ **9.0**	**CHECK All RCP Number 1 Seal DP LO Annunciators On Lampbox 4M07 - EXTINGUISHED**	PERFORM the following: **a.** ENSURE respective RCP Number 1 Seal leakoff isolation valve OPEN. • RCP 1A(2A) "SEAL NO 1 LKF ISOL FV-3154" • RCP 1B(2B) "SEAL NO 1 LKF ISOL FV-3155" • RCP 1C(2C) "SEAL NO 1 LKF ISOL FV-3156" • RCP 1D(2D) "SEAL NO 1 LKF ISOL FV-3157" **b.** ENSURE Seal Return Containment Isolation Valves - OPEN {CP004} • "SEAL RTN ICIV MOV-0077" • "SEAL RTN OCIV MOV-0079" **c.** ENSURE RCS pressure GREATER THAN 325 psig to provide adequate RCP(s) Number 1 Seal DP.
_____ **10.0**	**CHECK All RCP Number 2 Seal Leakoff Flow HI Annunciators On Lampbox 4M07 - EXTINGUISHED**	PERFORM Addendum 4.

Addendum 4 RCP Number 2 Seal Leakoff Flow High

NOTE

A high leakoff flow on a RCP Number 2 Seal may be caused by any of the following:

- RCP Number 2 Seal **NOT** fully seated
- Damage to RCP Number 2 Seal
- High leakage past outer dam of Number 3 Seal

NOTE

IF RCP Number 2 Seal high flow occurred during normal operation, THEN the cause may be a damaged RCP Number 3 Seal. IF this condition is suspected, THEN RCP operation may continue provided the RCP leak rate and vibration are acceptable. Number 3 Seal should be replaced as soon as practical.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ **1.0**	**DETERMINE Respective RCP Standpipe Fill Rate - NORMAL FILL FREQUENCY**	IF RCP Number 3 Seal leakage is contributing to Number 2 Seal leakage rate, THEN CONSULT with System Engineering and Plant Management concerning continued RCP operation.
_____ **2.0**	***MONITOR* RCP Vibration - LESS THAN RCP TRIP LIMITS**	IF any RCP vibration limit is exceeded, THEN PERFORM the following: **a.** IF the Reactor is critical, THEN PERFORM the following: **1)** TRIP the Reactor. **2)** ENSURE Main Turbine tripped. **b.** STOP affected RCP(s). **c.** IF reactor was tripped, THEN PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection. **d.** *CONTINUE* actions of this procedure as resources permit. **e.** RETURN TO Procedure step in effect.

NOTE

RCP Number 2 Seal **NOT** being fully seated may be the cause of RCP Number 2 Seal leakoff high flow. This condition usually occurs during initial RCP operation following seal maintenance or after a plant shutdown. IF this condition is suspected AND increased leak rate can be accepted for a limited time, THEN RCP operation may continue. A period of up to 24 hours may be required before Number 2 Seal seats and operates normally.

NOTE

IF RCP Number 2 Seal leakoff high flow is caused by damage to RCP Number 2 Seal, THEN RCP operation may continue provided there is no abnormal RCP vibration AND RCDT is capable of handling increased seal leakoff flow. RCP Number 2 Seal should be replaced as soon as practical.

_____ **3.0 CONSULT With System Engineering And Plant Management Concerning Continued RCP Operation**

_____ **4.0 RETURN TO Procedure Step In Effect**

RCP 1A(2A) STDPIPE LVL LO

NOTE

RCP 1A and 2A Seals have the Station Blackout Seal (SBO) installed

Automatic Actions: None

NOTE

"STANDPIPE FILL LCV-0178" automatically opens at a level of 52 inches from standpipe flange face to fill the standpipe and closes at a level of 37 inches from standpipe flange face.

Immediate Actions: None

Subsequent Actions:

NOTE

- Increased frequency of standpipe fill may indicate either a leak in the RHR System or failure of the RCP Number 3 Seal.
- Loss of RCP standpipe level will result in increased wear of the Number 3 Seal. RCP operation may continue; however, standpipe level should be re-established as soon as possible.

_____ 1) IF manual filling of RCP Standpipe is required, THEN PERFORM 0POP02-RM-0001, Reactor Makeup Water System Operations for section "Reactor Makeup Water System Operation For Manually Filling RCP Standpipes".

_____ 2) IF a failure of the Number 3 Seal is suspected, THEN GO TO 0POP04-RC-0002, Reactor Coolant Pump Off Normal.

_____ 3) EVALUATE the need to perform a RHR System inspection.

RCP 1A(2A) STDPIPE LVL LO

Probable Causes:

1) Normal seal leakoff
2) "STANDPIPE FILL LCV-0178" malfunction
3) RMW containment isolation valve "OCIV FV-3651" closed
4) Leak in the RHR System
5) Damaged or worn RCP Number 3 Seal
6) RCP Number 3 Seal not fully seated
7) Instrument failure
8) "1(2)-RM-0025 RMW TO RCB OCIV MANUAL ISOLATION" closed.

Origin and Setpoint:

1) 1(2)-CV-LSL-0178 58 in. from standpipe flange face

References:

1) Chemical and Volume Control System P&ID, 5R179F05005
2) Residual Heat Removal System P&ID, 5R169F20000
3) Logic Diagram, 6R-17-9-Z-42418
4) Precautions, Limitations and Setpoints, 5Z010ZS1101
5) Reactor Coolant Pump Manual, VTD W351-0128
6) 0POP04-RC-0002, Reactor Coolant Pump Off Normal
7) 0POP02-RM-0001, Reactor Makeup Water System Operation
8) CR 13-14396, Station Blackout Seals

RCP 1A(2A) STDPIPE LVL HI

NOTE

RCP 1A and 2A Seals have the Station Blackout Seal (SBO) installed.

Automatic Actions: None

Immediate Actions: None

NOTE

If RCP Standpipe high level alarm is due to manual fill or suspected leak by, then high level alarm can be monitored for three shifts before actions to lower RCP Standpipe level are taken. The nominal RCP Standpipe supply rate to RCP number 3 seal should lower RCP Standpipe level within three shifts. (Ref. CR 05-9053-5)

Subsequent Actions: _____ 1) ENSURE "STANDPIPE FILL LCV-0178" closed.

_____ 2) IF "STANDPIPE FILL LCV-0178" is suspected of leaking by, THEN PERFORM the following:

_____ a) CLOSE RMW containment isolation valve "OCIV FV-3651".

_____ b) INITIATE a Condition Report to repair "STANDPIPE FILL LCV-0178".

_____ c) IF manual filling or draining of RCP Standpipe, THEN PERFORM 0POP02-RM-0001, Reactor Makeup Water System Operations for section "Reactor Makeup Water System Operation For Manually Filling RCP Standpipes" or "Lowering RCP Standpipe Level".

_____ 3) IF RCP Number 2 Seal is suspected of excessive leakage, THEN GO TO 0POP04-RC-0002, Reactor Coolant Pump Off Normal.

RCP 1A(2A) STDPIPE LVL HI (Continued)

Probable Causes:

1) "STANDPIPE FILL LCV-0178" open or leaking by
2) RCP Number 2 Seal excessive leakage
3) Instrument failure

Origin and Setpoint:

1)	1(2)-CV-LSH-0178A	31 in. from standpipe flange face

References:

1) Chemical and Volume Control System P&ID, 5R179F05005
2) Logic Diagram, 6R-17-9-Z-42418
3) Precautions, Limitations and Setpoints, 5Z010ZS1101
4) Reactor Coolant Pump Manual, VTD W351-0128
5) 0POP04-RC-0002, Reactor Coolant Pump Off Normal
6) 0POP02-RM-0001, Reactor Makeup Water System Operations
7) CR 13-14396, Station Blackout Seals

Exam Bank No.: 3202 **Last used on an NRC exam:** Never

SRO Sequence Number: 87

Unit 1 is at 100% power.

- An RO is performing the stroke test for RCFC 11A/12A CCW SPLY OCIV, MOV-0057 in accordance with 0PSP03-CC-0007, Component Cooling Water System Train 1A(2A) Valve Operability Test.
- The RO notes intermediate indication on the open stroke when the indicating lights go OFF and the associated BYP/INOP light illuminates.

The Unit Supervisor should enter the Technical Specification LCO Actions of ___(1)___.

The most limiting time to restore MOV-0057 to operable status per Technical Specifications without performing additional actions is ______(2)_____.

A. (1) 3.6.3, Containment Isolation Valves and 3.6.2.3, Containment Cooling System
(2) 24 hours

B. (1) 3.6.3, Containment Isolation Valves and 3.6.2.3, Containment Cooling System
(2) 7 days

C. (1) 3.6.3, Containment Isolation Valves only
(2) 24 hours

D. (1) 3.6.3, Containment Isolation Valves only
(2) 7 days

Answer: A (1) 3.6.3, Containment Isolation Valves and 3.6.2.3, Containment Cooling System (2) 24 hours

Exam Bank No.: 3202 **Source:** New **Modified From** N/A

K/A Catalog Number: 008 G2.2.37 Ability to determine operability and/or availability of safety related equipment.

SRO Importance: 4.6 **Tier:** 2 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(2)

SRO Justification:

The student must apply greater than one hour Technical Specification actions.

STP Lesson: LOT 503.01 **Objective Number:** 80056

DETERMINE the applicable Technical Specification and/or the Technical Requirements Manual (TRM) Limiting Conditions for Operation (LCOs) and the required action(s) to be taken.

Reference: Technical Specification 3.6.3 and 3.6.2.3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. MOV-0057 is a containment isolation valve but also affects the ability to supply CCW to RCFCs 11A/12A on a Safety injection. Therefore both TS 3.6.3 and 3.6.2.3 apply. TS 3.6.3 requires that the valve be restored in 24 hours or additional actions must be taken.

B: INCORRECT. Plausible because the action time for 3.6.2.3 is 7 days. The first part is correct.

C: INCORRECT. Plausible because even though the valve is de-energized the RCFCs still have cooling water from RCB Chill Water. The second part is correct.

D: INCORRECT Plausible because the action time for 3.6.2.3 is 7 days and even though the valve is de-energized the RCFCs still have cooling water from RCB Chill Water.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions to apply Technical Specifications and then determine the action and restoration times

CONTAINMENT SYSTEMS

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Three independent groups of Reactor Containment Fan Coolers (RCFC) shall be OPERABLE with a minimum of two units in two groups and one unit in the third group.

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

ACTION:

A. With one group of the above required Reactor Containment Fan Coolers inoperable, within 7 days restore the inoperable group of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one group of the above required Reactor Containment Fan Coolers inoperable, within 1 hour restore at least two groups of RCFC to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of Reactor Containment Fan Coolers shall be demonstrated OPERABLE:

a. At a frequency in accordance with the Surveillance Frequency Control Program by:

   1) Starting each non-operating fan group from the control room, and verifying that each fan group operates for at least 15 minutes, and

   2) Verifying a component cooling water flow rate of greater than or equal to 1800 gpm to each cooler.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that each fan group starts automatically on a Safety Injection test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE with isolation times less than or equal to the required isolation times.

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

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation barrier(2) OPERABLE in each affected penetration that is open and within 24 hours:

a. Restore the inoperable valve(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or check valve with flow through the valve secured(1) (3), or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange(1), or

d. Apply the requirements of the CRMP.

Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at a frequency in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Containment Ventilation Isolation test signal, each purge and exhaust valve actuates to its isolation position; and

c. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position.

d. Verifying that on a Phase "A" Isolation test signal, coincident with a low charging header pressure signal, that each seal injection valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

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(1) Penetration flow paths (except for Containment Purge flow paths) may be unisolated intermittently under administrative controls.

(2) An isolation barrier may either be a closed system (i.e., General Design Criteria 57 penetrations) or an isolation valve.

(3) A check valve may not be used to isolate an affected penetration flow path in which more than one isolation valve is inoperable or in which the isolation barrier is a closed system with a single isolation valve (i.e., General Design Criteria 57 penetration)

Exam Bank No.: 3239 **Last used on an NRC exam:** Never

SRO Sequence Number: 88

The unit is at 100% power.

- Class 1E 125VDC Bus E1A11 deenergizes.
- Shortly thereafter, a Reactor Trip and Safety Injection occur.

"A" Train ECCS Pumps ____(1)____ start automatically.

The Unit Supervisor should direct the performance of _____(2)____.

A. (1) will NOT
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection ONLY

B. (1) will
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection ONLY

C. (1) will NOT
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection, AND 0POP04-DJ-0001, Loss of Class 1E 125VDC Power (as time permits)

D. (1) will
(2) 0POP05-EO-EO00, Reactor Trip or Safety Injection, AND 0POP04-DJ-0001, Loss of Class 1E 125VDC Power (as time permits)

Answer: C (1) will NOT; (2) 0POP05-EO-EO00, Reactor Trip or Safety Injection, AND 0POP04-DJ-0001, Loss of Class 1E 125VDC Power (as time permits)

Exam Bank No.: 3239 **Source:** New **Modified From** N/A

K/A Catalog Number: 013 A2.05 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of dc control power.

SRO Importance: 4.2 **Tier:** 2 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

This question requires knowledge of administrative procedures that specify hierarchy, implementation, or coordination of plant normal, abnormal, and emergency procedures.

STP Lesson: LOT 505.01 **Objective Number:** 98035

Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: POP01-ZA-0018; 0POP04-DJ-0001

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because EO00 is the highest priority procedure and not all off-normals will be implemented with the EOPs but some off-normal procedures such as DJ-0001 are important enough to the overall impementation strategy to allow implementation as time permits. The indiviuall ONP will state this. The first part is correct.

B: INCORRECT. Plausible because SI should start the "A" Train ECCS pumps but loss of E1A11 causes loss of the sequencer and breaker control power to the "A" Train ECCS pumps and becsuse EO00 is the highest priority procedure and not all off-normals will be implemented with the EOPs but some off-normal procedures such as DJ-0001 are important enough to the overall impementation strategy to allow implementation as time permits.

C: CORRECT. Loss of E1A11 causes loss of the sequencer and breaker control power to the "A" Train ECCS pumps. DJ-0001 requires implementation as time and resources permit. Ths is further supported by POP01-ZA-0018, EOP User's Guide.

D: INCORRECT. Plausible because control of components is normally performed from the control room but in this case control power is lost to the Train A components and local action will be needed to start this equipment if needed. The first part is correct.

Question Level: H **Question Difficulty** 3

Justification:

Student must anayze conditions given in the stem and determine correct procedures to implement and the affect on Train A ECCS.

CAUTION

IF the AFW cross-connects are opened, THEN care SHALL be taken NOT to exceed 675 gpm flow on any AFW pump to prevent runout of the pump.

4.26.3 IF specific EOP SG(s) levels OR minimum AFW flow requirements are met, THEN the following actions can be taken:

- Reset the AFW system (including resetting the SI signal, SG LO-LO LVL and ESF Load Sequencers), open the AFW cross-connect valves, and decrease the number of running AFW pumps.
- Reset the AFW system (including resetting the SI signal and SG LO-LO LVL, if closure of the AFW OCIV(s) is needed) and manually control AFW flow.
- Isolate AFW flow to a faulted OR ruptured SG by placing the appropriate AFW pump in Pull-To-Lock.

4.26.4 Actions should be taken per Off Normal Operating Procedures and Annunciator Response Procedures that DO **NOT** conflict with the actions of the EOPs if adequate resources are available. The Off Normal Operating Procedure or Annunciator Response Procedure should be entered and procedure steps followed. (e.g., IF during the performance of the EOPs there are indications of abnormal RCP conditions, THEN the RCP Off Normal Operating Procedure SHOULD be entered.)

4.26.5 IF necessary to reset SI, Phase A, or ESF Load Sequencers to mitigate the consequences of the accident, prevent equipment damage OR protect the health and safety of plant personnel, THEN the SM/US should direct these actions prior to direction by the EOPs.

CAUTION

- A loss of CCW to CCP Supplemental Cooler may cause respective CCP motor failure in as little as 4 minutes.
- A loss of CCW to CCP Lube Oil Cooler may cause pump failure in as little as 8 minutes.

NOTE

- A Loss of all Class 1E DC power may require entry into the Emergency Plan. Refer to 0EPR04-UA-0001, EAL Wall Charts.
- Foldout CIP should be opened.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	**1.0**	**CHECK Reactor – TRIPPED**	GO TO Step 3.0
_____	**2.0**	**PERFORM The Following:**	
_____		**a. PERFORM 0POP05-EO-EO00, Reactor Trip Or Safety Injecton**	
_____		**b. CONTINUE Actions Of This Procedure As Resources Permit**	
_____	**3.0**	**CHECK Charging System Status - ANY PUMP RUNNING WITH CCW FLOW ESTABLISHED**	GO TO Step 5.0.
_____	**4.0**	**GO TO Step 9.0 Observing Note Prior To The Step**	
_____	**5.0**	**CHECK Centrifugal Charging Pump 1A(2A) Status - RUNNING**	GO TO Step 7.0.

Exam Bank No.: 3193 **Last used on an NRC exam:** Never

SRO Sequence Number: 89

Unit 1 is at 7% power.

- The Reactor Operator reports that Computer Point AFTA7526, "AFWP 14 DISCH TEMP" is in alarm.
- The Unit Supervisor enters 0POP04-AF-0001, Auxiliary Feedwater Discharge Header High Temperature.

Per 0PSP03-ZQ-0028, Operator Logs, the temperature limit for AFW pump discharge header temperature is ___(1)___.

Per 0POP04-AF-0001, the Unit Supervisor should direct ___(2)___ to prevent potential pump damage due to vapor binding.

A. (1) 115ºF
(2) running AFW Pump 14 on long path recirculation

B. (1) 115ºF
(2) uncapping and opening 1-AF-0246 AFW Pump 14 Discharge Line Vent Valve to gravity drain water from the AFWST

C. (1) 200ºF
(2) running AFW Pump 14 on long path recirculation

D. (1) 200ºF
(2) uncapping and opening 1-AF-0246 AFW Pump 14 Discharge Line Vent Valve to gravity drain water from the AFWST

Answer: A (1) 115ºF (2) running AFW Pump 14 on long path recirculation

Exam Bank No.: 3193 **Source:** New **Modified From** N/A

K/A Catalog Number: 061 A2.06 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: Back leakage of MFW

SRO Importance: 3.0 **Tier:** 2 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must know the content of the procedure vs. the overall mitigative strategy.

STP Lesson: LOT 505.01 **Objective Number:** 92108

Given a plant condition, STATE the actions required to be performed per the applicable Off-Normal procedure.

Reference: 0POP04-AF-0001; 0PSP03-ZQ-0028

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Per the Operator Logs, a temperature above 115ºF is the limit for AFW pump discharge temperature. The computer point alarms at 200ºF. Per 0POP04-AF-0001, the pump is run on long path recirculation until temperature goes below 115ºF which is necessary to prevent vapor binding.

B: INCORRECT. Plausible because 115ºF is the correct temperature and opening 1-AF-0246 would cause water to gravity drain from the AFWST which would cool the line down but this is not the method used per the POP04.

C: INCORRECT. Plausible as 200ºF is the temperature setpoint for the computer alarm. The second part is correct.

D: INCORRECT. Plausible as 200ºF is the temperature setpoint for the computer alarm and opening 1-AF-0246 would cause water to gravity drain from the AFWST which would cool the line down but this is not the method used per the POP04.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall the indications for AFW pump check valve backleakage and determine the correct procedure action.

UNIT: _____ DATE: ____________

PARAMETER	LOCATION	INSTR	00-02	12-14	LIMIT	TECHNICAL SPECIFICATION	MODE	NOTE
PRI AIR DT	PLANT COMPUTER OR DIALUP CONNECTION	EMTN0004			CHNL CHECK °F (2)	TRM 3.3.3.4, Table 3.3-8		(1) IF ANY OF THE FOLLOWING LIMITS ARE EXCEEDED, THEN ANALYZE TREND OF PREVIOUS READINGS TO ENSURE INSTRs ARE **NOT** STUCK: (Reference. 6.17.13) NOTE: THESE LIMITS ARE ONLY APPLICABLE FOR WIND SPEEDS > 5 MPH. A) >30 DEG BETWEEN 60M & 10M WIND DIR. B) >30 DEG BETWEEN 10M PRI WIND DIR & 10M BACKUP WIND DIR. C) >10 MPH BETWEEN 10M PRI & 10M BACKUP WIND SPEED. (2) PRI 10M & 60M TEMP 15 MIN AVG'S SHOULD BE COMPARED TO EMTN0004 FOR CHANNEL CHECK. •EMTN0010 PRI 10M TEMP 15 MIN AVG •EMTN0011 PRI 60M TEMP 15 MIN AVG
PRI 10M SPEED		EMSN0001			MPH (1)			
PRI 10M DIR		EMXN0002			DEGREES (1)			
60M SPEED		EMSN0008			CHNL CHECK MPH			
60M DIR		EMSN0009			DEGREES (1)			
B/U 10M SPEED		EMXN0012			MPH (1)			
B/U 10M DIR		EMXN0021			DEGREES (1)			
PRI ROOM TEMP		EMTN0003			60°F ≤ TEMP ≤ 90°F	N/A		
BACKUP ROOM TEMP		EMTN0044						
AF TEMP C	PLANT COMPUTER	AFTA7523			≤115°F (1)	N/A		(1) IF LIMIT IS EXCEEDED, THEN REFER TO 0POP04-AF-0001, Auxiliary Feedwater Discharge Header
AF TEMP B		AFTA7524						
AF TEMP A		AFTA7525						
AF TEMP D		AFTA7526						

This form, when completed, SHALL be retained for a minimum of five years.

PURPOSE

This procedure provides guidelines for responding to Auxiliary Feedwater (AFW) check valve backleakage as indicated by AFW pump discharge header high temperature. Check valve back leakage of hot water from the Steam Generators could lead to steam binding of the AFW pumps when saturation temperature is reached for existing pressure. If a pump were to be started, the lowest pressure would be inside the casing of the pump and under these conditions this is where the steam binding would occur.

AFW pump discharge header high temperature is monitored once per shift in 0PSP03-ZQ-0028, Operator Logs. The Plant Computer will alarm if any header temperature reaches 200°F.

The response to AFW pump discharge header high temperature is to run the pump in that train on long-path recirculation until temperature goes below 115°F. On A and D trains the long path recirculation line is within 1 foot of the temperature element. Cooling down B and C trains will take longer because the long path recirculation line for B train is 13 feet from the temperature element, and the long path recirculation line for C train is 2 and one-third feet from the temperature element.

A Condition Report should be initiated each time this procedure is entered to trend the possible check valve backleakage.

SYMPTOMS OR ENTRY CONDITIONS

1. Plant Computer Points indicating AFW pump discharge temperature GREATER THAN OR EQUAL TO 115°F:
    - AFTA7525, AFWP 11(21) DISCH TEMP
    - AFTA7524, AFWP 12(22) DISCH TEMP
    - AFTA7523, AFWP 13(23) DISCH TEMP
    - AFTA7526, AFWP 14(24) DISCH TEMP

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1.0	**DISPATCH An Operator To The Affected AFW Pump Room(s) To Inspect AFW Pump(s) And Associated Piping For Indication Of Steam Or Water Leaks {10 ft IVC}**	
_____ 2.0	**REVIEW Operability Assessment System (OAS) To Ensure No Equipment Is Out Of Service That Could Cause Multiple Trains To Be Inoperable While Performing This Activity**	

NOTE

IF more than one discharge header temperature is high, THEN cooldown should be performed in order of the header with the highest temperature.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 3.0	**CHECK The Following Plant Computer Points - LESS THAN 115°F**	PERFORM the following:
	• AFTA7525 "AFWP 11(21) DISCH TEMP"	a. IF AFTA7525 "AFWP 11(21) DISCH TEMP" is GREATER THAN OR EQUAL TO 115°F, THEN GO TO Addendum 1.
	• AFTA7524 "AFWP 12(22) DISCH TEMP"	b. IF AFTA7524 "AFWP 12(22) DISCH TEMP" is GREATER THAN OR EQUAL TO 115°F, THEN GO TO Addendum 2.
	• AFTA7523 "AFWP 13(23) DISCH TEMP"	c. IF AFTA7523 "AFWP 13(23) DISCH TEMP" is GREATER THAN OR EQUAL TO 115°F, THEN GO TO Addendum 3.
	• AFTA7526 "AFWP 14(24) DISCH TEMP"	d. IF AFTA7526 "AFWP 14(24) DISCH TEMP" is GREATER THAN OR EQUAL TO 115°F, THEN GO TO Addendum 4.
_____ 4.0	**INITIATE A Condition Report To Trend The Possible Check Valve Backleakage**	

-END-

This Procedure is Applicable in ALL Modes

Step 2.0 continued from previous page

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	l.	**CHECK "AFWP 14(24) T & T MECH OVERSP TRIP" Annunciator 06M4 Window C8 - EXTINGUISHED**	--

CAUTION

The turbine driven AFW Pump 14(24) must be rapidly accelerated to operating speed to avoid pump damage.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	m.	**CHECK Turbine Driven AFW Turbine speed on digital tachometer N1(2)AFSI7538A - GREATER THAN OR EQUAL TO 1900 RPM 45 SECONDS FOLLOWING PUMP START** **{Turbine Control Panel N1(2)AFZLP123}**	**m.** Immediately TRIP Turbine Driven AFW Pump 14(24). --
_____	n.	**OPEN "TURB TRIP/THROT MOV-514" to start AFW Pump 14(24). {CP006}**	--
_____	o.	***MONITOR* AFW Pump 14(24) discharge pressure on PI-7529.**	--
_____	3.0	**THROTTLE AFW To SG 1D(2D) "AFW REG FV-7526" to establish approximately 200 GPM recirc flow**	--

NOTE

- Components are located on CP006 unless otherwise noted.
- Turbine driven AFW Pump 14(24) should be stopped by tripping the Trip/Throttle valve. This will avoid unnecessary operation at slow speeds (low head).

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 4.0	**CHECK Plant Computer Point AFTA7526 “AFWP 14(24) DISCH TEMP” - LESS THAN 115°F**	PERFORM the following: **a.** *MONITOR* Plant Computer Point AFTA7526 “AFWP 14(24) DISCH TEMP". **b.** WHEN Plant Computer Point AFTA7526 “AFWP 14(24) DISCH TEMP" indicates LESS THAN 115°F, THEN GO TO Step 5.0.
_____ 5.0	**TRIP "TURB TRIP/THROT MOV-514" to stop AFW Pump 14(24)**	
_____ 6.0	**ENSURE AFW To SG 1D(2D) Regulating Valve "AFW REG FV-7526" - OPEN**	
_____ 7.0	**ENSURE AFW To SG 1D(2D) Regulating Valve “AFW REG FV-7526” - INDEPENDENTLY VERIFIED OPEN**	

NOTE

Plant Computer Point T7601 “AFW PMP STM T&T VALVE LEAKBY” may be monitored to check for leakage by MOV-514.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 8.0	**DIRECT The Operator To Visually Check The Trip And Throttle Valve Is Reset (Run To Minimum) AND The Solenoid Is Latched.**	

Exam Bank No.: 3194 **Last used on an NRC exam:** Never

SRO Sequence Number: 90

Unit 1 is in Mode 3.

- Annunciator 03M2 Window D-1, ‘125V DC SYSTEM E1C11 TRBL” is in alarm.
- The Plant Operator reports that Battery Charger E1C11-1 AC Input CB-1 is tripped.

Based on the above conditions, _________(1)_________ should be declared INOPERABLE.

The most limiting Technical Specification action time to restore the inoperable component(s) is _____(2)_____ hours. (Assume CRMP will NOT be applied)

A. (1) Battery Charger E1C11-1 and the “C” Train Battery
(2) 12

B. (1) Battery Charger E1C11-1 and the “C” Train Battery
(2) 2

C. (1) Battery Charger E1C11-1 ONLY
(2) 12

D. (1) Battery Charger E1C11 ONLY
(2) 2

Answer: B (1) Battery Charger E1C11-1 and the "C" Train Battery (2) 2

Exam Bank No.: 3194 **Source:** New **Modified From** N/A

K/A Catalog Number: 063 G2.2.40 Ability to apply Technical Specifications for a system.

SRO Importance: 4.7 **Tier:** 2 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(2)

SRO Justification:

The student must have knowledge of greater than one hour Technical Specification actions.

STP Lesson: LOT 503.01 **Objective Number:** 80056

DETERMINE the applicable Technical Specification and/or the Technical Requirements Manual (TRM) Limiting Conditions for Operation (LCOs) and the required action(s) to be taken.

Reference: 0POP02-EE-0001; Technical Specification 3.8.2.1 Action c and Action a

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because 12 hours is also a requirement but this is for verifying float current does not exceed 2 amps.The first part is correct.

B: CORRECT. 0POP02-EE-0001 states that the battery is inoperable anytime it is not aligned to an operating a battery charger. The Safety Related 125 VDC bateries are aligned to only one charger at a time. Technical Specification 3.8.2.1.c,1 requires restoring battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours. Additionally, 3.8.2.1.a reqiures that the battery bank be restored to operable within 2 hours.

C: INCORRECT. Plausible because all but the safety related batteries have both chargers on-line and load sharing, but the safety batteries do not, and 12 hours is also a requirement in the action but this is for verifying float current does not exceed 2 amps.

D: INCORRECT. Plausible because all but the safety related batteries have both chargers on-line and load sharing, but the safety batteries do not. The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student analyze conditons and determine Technical Specification actions and procedure requirements.

3/4.8.2 DC SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following DC electrical sources shall be OPERABLE:

a. Channel I 125-volt Battery Bank E1A11 (Unit 1), E2A11 (Unit 2) and one of its two associated chargers,

b. Channel II 125-volt Battery Bank E1D11 (Unit 1), E2D11 (Unit 2) and one of its two associated full capacity chargers,

c. Channel III 125-volt Battery Bank E1B11 (Unit 1), E2B11 (Unit 2) and one of its two associated full capacity chargers, and

d. Channel IV 125-volt Battery Bank E1C11 (Unit 1), E2C11 (Unit 2) and one of its two associated chargers.

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

ACTION:

NOTE

If the batteries discharge for more than 2 hours as the sole source of power to their DC bus while the CRMP is being applied and no alternate source of power is available, the LCO shall be considered not met.

a. With one of the required battery banks inoperable, within 2 hours restore the inoperable battery bank to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With more than one of the required battery banks inoperable, within 1 hour restore at least three battery banks to OPERABLE status or apply the requirements of the CRMP or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one channel with no battery chargers OPERABLE,

   1. Restore battery terminal voltage to greater than or equal to the minimum established float voltage within 2 hours, AND

   2. Verify float current for the affected battery does not exceed 2 amps once per 12 hours, AND

   3. Restore one battery charger to OPERABLE status within 72 hours.

   If the battery terminal voltage cannot be restored in the allowed time, float current is excessive, or a battery charger is not restored to operability in the time allowed, apply the requirements of the CRMP or the affected reactor unit is to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4.0 Notes and Precautions

4.1 Battery room ventilation SHALL remain in continuous operation.

4.2 IF battery room ventilation is lost during an equalizing charge, THEN Electrical Maintenance SHALL be notified to secure the equalizing charge.

4.3 IF battery acid comes in contact with the skin or eyes, THEN the following SHALL be performed:

- The affected area SHALL be flushed at the Emergency Wash Station.
- Control Room personnel SHALL be notified.

4.4 Smoking and activities which could produce sparks or flames are prohibited in the battery room.

4.5 Tools used in the battery room SHALL be insulated and sparkless.

4.6 All battery cell flash arrestors and fill caps SHALL be kept secure in place.

NOTE

The 15 minute allowance to not perform Section 10.0, Class 1E Battery Operability Following a Discharge Transient, is based upon the Class 1E battery's design capacity, even with one battery cell jumpered out of the battery bank. IF more than one cell in the battery bank is jumpered out of the battery bank, THEN there is no allowance for 15 minutes to restore the battery bank to the charger and Section 10.0 must be performed.

4.7 A Class 1E battery is inoperable anytime it is **NOT** aligned to an operating battery charger.

4.7.1 IF a Class 1E battery is realigned to an operating battery charger in less than or equal to **15 minutes** AND **only up to one battery cell is jumpered out**, THEN the Class 1E battery may be declared OPERABLE after verifying the operating battery charger normal float voltage is greater than or equal to 128.5 VDC. (References 2.20, 2.21, 2.22 and 2.29)

4.7.2 IF a Class 1E battery is **NOT** realigned to an operating battery charger within **15 minutes** OR **more than one battery cell is jumpered out**, THEN Section 10.0, Class 1E Battery Operability Following a Discharge Transient, must be performed prior to declaring the battery OPERABLE. (References 2.18, 2.21 and 2.22)

Exam Bank No.: 3222 **Last used on an NRC exam:** Never

SRO Sequence Number: 91

During Refueling, a fuel assembly in close contact with other fuel assemblies will not move from the lower core plate pins.

This situation can be addressed by using the _____(1)_____ switch on the Refueling Machine.

After agreement is reached on the action plan, use of this switch is directed by the _____(2)_____.

A. (1) OPEN WATER PERMISSIVE
(2) Core Load Supervisor

B. (1) OPEN WATER PERMISSIVE
(2) Shift Manager

C. (1) LOAD SELECT
` (2) Core Load Supervisor

D. (1) LOAD SELECT
(2) Shift Manager

Answer: C (1) LOAD SELECT (2) Core Load Supervisor

Exam Bank No.: 3222 **Source:** New **Modified From**

K/A Catalog Number: 034 K4.03 Knowledge of design feature(s) and/or interlock(s) which provide for the following: Overload protection

SRO Importance: 3.3 **Tier:** 2 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(7)

SRO Justification:

Unique to the SRO position (Core Load Supervisor)

STP Lesson: LOT 201.45 **Objective Number:** 29315

Describe the bypasses associated with the operation of the Refueling Bridge, Trolley, and Hoist.

Reference: 0POP08-FH-0001, Refueling Operations, Add. 4, Form 1 and 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as "OPEN WATER" would be a correct direction to move to with a fuel assembly and with confusion over the various refueling switches. The Core Load Supervisor does have responsibility for these switches.

B: INCORRECT: Plausible as "OPEN WATER" would be a correct direction to move to with a fuel assembly and with confusion over the various refueling switches and who is responsible for their use.

C: CORRECT: The LOAD SELECT switch is normally in position 1 but can be changed with Core Load Supervisor permission to remove a stuck assembly.

D: INCORRECT: Plausible as this is the correct switch to use and the Shfit Manager does have responsibilities for the refueling process.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess conditions and select a course of action.

5. Lateral Movements of Fuel Assemblies

   Any lateral movement of the fuel assembly when not fully withdrawn into the refueling machine mast should only be performed with concurrence of the Fuel Handling Supervisor and with observers stationed to monitor fuel movement. An on index move may be an option to Lateral Movement of fuel assembly based on close contact move hazards.

   Particular care should be exercised when laterally moving fuel assemblies in close contact with opposing adjacent fuel assemblies. Observers should provide visual confirmation the assembly moves freely. The bottom nozzle should move simultaneously with the top nozzle with no indications of hang-ups or excessive movement of assemblies in close contact with the one being moved. Such movements should be limited to a single pair of opposing assemblies.

   Fuel movements should be planned to avoid lateral movement of a fuel assembly between a baffle corner and another fuel assembly. If an assembly is moved past a baffle corner it should be at least two inches away from the baffle. An exception can be made if the assembly is jogged to open water in approximately ½ inch increments in a zig zag fashion. The first increment should move the assembly away from the baffle.

   Fuel assemblies in close contact should not be moved laterally unless it is verified that the grids and nozzles are at an elevation that will not allow grid to grid or grid to nozzle interaction with an adjacent fuel assembly.

   When fuel is moved laterally in close contact with other fuel assemblies the load cell should be monitored for sudden increases in load. Ideally no load increase should be detected. Limits will depend on the sensitivity of the load monitoring device. The lowest detectable load that will not cause spurious work stoppage should be used. If an unexpected load increase of 30 pounds is detected, movement should stop and the possibility of a hang-up investigated. (Reference 2.11)

6. IF inserting or raising a fuel assembly in close contact with other fuel assemblies, and/or the baffle wall, THEN it may be necessary to increase the underload or overload setting using the "LOAD SELECT" switch or bypass the underload/overload. Closely monitor load indication to ensure an unexpected load change of 200 lbs. does NOT occur. IF it is necessary to use the "LOAD SELECT" switch to increase an underload/overload setting, THEN notify the Core Load Supervisor and Reactor Engineering. IF it is necessary to bypass the underload or overload, THEN Form 2 SHALL be utilized.

Typical Close Contact movement cases are illustrated and described in Pages 3 and 4 of this Addendum.

CAUTION

- Load Cell indication SHALL NOT exceed 3250 pounds, maximum value for automatic overload cutoff. (TRM 3.9.6)
- The noted weight of the fuel assembly SHALL NOT be exceeded by greater than 1000 lbs. (Reference 2.14)

NOTE

- The Core Loading Supervisor SHALL notify Reactor Engineering prior to each switch position incremental increase. (Reference 2.14)
- Cable shakes and off-index moves may be used as necessary to help free the stuck assembly during each lift increment increase.
- Weight values for each force increase and the listed pull force above assembly weight listed in the subsequent steps are approximate values.

4.0 ATTEMPT to free the fuel assembly as follows:

4.1 IF fuel assembly has a control rod inserted, THEN perform the following:

4.1.1 N/A Form 1 Data Sheet 1 for the first load increase.

4.1.2 GO TO Step 4.6.

4.2 Initial Form 1 Data Sheet 1 for a 2650 lb. setting (Nominal F/A + RCCA + 200 lbs.), indicating agreement to proceed.

4.3 WHEN directed by the Core Loading Supervisor, THEN PLACE “LOAD SELECT” switch in position 3.

4.4 MOVE the "HOIST JOG" switch to the UP position.

4.5 IF the fuel assembly is freed, THEN slowly RAISE the fuel assembly approximately one foot to ensure the fuel assembly is clear AND GO TO Step 4.19.

4.6 IF the fuel assembly is **NOT** free, THEN INITIAL Form 1 Data Sheet 1 for a 2850 lb. setting (Nominal F/A + RCCA + 400 lbs.), indicating agreement to proceed, OTHERWISE proceed to Step 4.18.

4.7 WHEN directed by the Core Loading Supervisor, THEN PLACE THE "LOAD SELECT" switch in position 4.

4.8 MOVE the "HOIST JOG" switch to the UP position.

4.9 IF the fuel assembly is freed, THEN slowly RAISE the fuel assembly approximately one foot to ensure the fuel assembly is clear AND GO TO Step 4.19.

4.10 IF the fuel assembly is **NOT** free, THEN INITIAL Form 1 Data Sheet 1 for a 3050 lb. setting (Nominal F/A + RCCA + 600 lbs.), indicating agreement to proceed, OTHERWISE proceed to Step 4.18.

4.11 WHEN directed by the Core Loading Supervisor, THEN PLACE “LOAD SELECT” switch in position 5.

4.12 MOVE the "HOIST JOG" switch to the UP position.

4.13 IF the fuel assembly is freed, THEN slowly RAISE the fuel assembly approximately one foot to ensure the fuel assembly is clear AND GO TO Step 4.19.

4.14 IF the fuel assembly is NOT free, THEN INITIAL Form 1 Data Sheet 1 for a 3150 lb. setting (Nominal F/A + RCCA + 700 lbs.), indicating agreement to proceed, OTHERWISE proceed to Step 4.18.

4.15 WHEN directed by the Core Loading Supervisor, THEN PLACE "LOAD SELECT" switch in position 6.

4.16 MOVE the "HOIST JOG" switch to the UP position.

4.17 IF the fuel assembly is freed, THEN slowly RAISE the fuel assembly approximately one foot to ensure the fuel assembly is clear AND GO TO Step 4.19.

4.18 IF the fuel assembly is NOT free, THEN PERFORM the following:

4.18.1 CONTACT Reactor Engineering. (Reference 2.14)

4.18.2 N/A Step 4.19 on Form 1 Data Sheet 1.

4.18.3 GO TO Step 4.20.

4.19 RECORD the following on Form 1 Data Sheet 1:

- Date/Time assembly was freed.
- Total force required to free assembly.

4.20 ENSURE the "GRIPPER OVERLOAD BYPASS" (TS-1) switch in the OFF position.

4.21 VERIFY the “GRIPPER OVERLOAD BYPASS" light is OFF.

4.22 PLACE the “LOAD SELECT” switch in position 1, 2, or 3 as applicable for the fuel assembly that was freed.

4.23 RETURN TO procedure step in effect.

Form 1

Stuck Fuel Assembly Removal Instructions, Data Sheet 1

Page 5 of 5

Date/Time (Step 2.0)	F/A ID (2.0)	Core Location (2.0)	Force Increase Required (3.1)	2650 lb Setting SW 3 (4.1/4.2)	2850 lb Setting SW 4 (4.6)	3050 lb Setting SW 5 (4.10)	3150 lb Setting SW 6 (4.14)	Date/Time Assembly Freed (4.19)	Total Force Required (4.19)	Comments/Notes
1										
2										
3										
4										
5										
6										
7										
8										
9										
10										
11										
12										
13										
14										

This form, when completed, SHALL be retained for the life of the plant.

Form 2	Bypass Mode Operation Justification and Authorization	Page 1 of 1

IF a fuel assembly is in the mast AND it is required to place an Interlock Bypass Switch in the ON position, THEN the Core Loading Supervisor or designee SHALL complete Form 2, documenting the reason, review, authorization and any special precautions required, for using the bypass. This form is to be maintained in the Control Room.

RECORD Bypass Switch to be used: ____________________

Reason for operation in Bypass: ____

____

____

____

Special limits/precautions to be observed: ____

____

____

____

The following personnel have reviewed and authorized placement of the Bypass Switch in the ON position:

	Signature	Date
Shift Manager:	__________________	/ _____________
Rx Engineering Designee:	__________________	/ _____________

RECORD: Bypass Sw. _________________ Placed in ON position _________/_____________
Time Date

Core Loading Supervisor: _______________/_________
Signature Date

This form, when completed, SHALL be retained for the life of the plant.

Exam Bank No.: 3196 **Last used on an NRC exam:** Never

SRO Sequence Number: 92

Unit 2 was at 100% power and started shutting down due to a S/G Tube Leak.

At 0830 the following conditions were observed:

- RT-8039, Failed Fuel Rad Monitor is reading 32 μCi/ml and slowly rising.
- S/G 1A Steam Line Rad Monitor RT-8046 is reading 2E-01 μCi/ml and slowly rising.
- Estimated RCS leak rate is 5 gpm and stable.

The emergency classification level should be an _____(1)_____. (Reference Provided)

If the original emergency classification was declared at 0840 the **NRC** must be notified by ___(2)___.

A. (1) Unusual Event
   (2) 0855

B. (1) Unusual Event
   (2) 0940

C. (1) Alert
   (2) 0855

D. (1) Alert
   (2) 0940

Answer: B (1) Unusual Event (2) 0940

Exam Bank No.: 3196 **Source:** New **Modified From** N/A

K/A Catalog Number: 072 G2.4.38 Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

SRO Importance: 4.4 **Tier:** 2 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The task performed in this question is unique to the SRO position (Emergency Director).

STP Lesson: LOT 803.14 **Objective Number:** SRO-74206

Given an emergency condition and a copy of the emergency classification tables, CLASSIFY the emergency condition.

Reference: 0EPR01-UA-0001, 0EPR02-CK-OPS01

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☑ **Attachment:** 0EPR01-UA-0001

Distractor Justification

A: INCORRECT. Plausible because 0855 is the time required to notify other agencies such as state and county. The first part is correct.

B: CORRECT. An unusual event should be declared based on the RT-8039 reading per SU3 EAL-1 , there is no time requirement as with RT-8046 Steam Line Monitor (student could also call a UE on RU1 EAL-1 if they believe the readings will be at this level for greater than 60 minutes). The NRC should be notified as soon as possible but must be notified within 1 hour which would be 0940.

C: INCORRECT. Plausible because the Main Steam Line monitor readings are elevated but do not exceed the Alert value but are still rising toward the value and 0855 is the time required to notify other agencies such as state and county.

D: INCORRECT. . Plausible because the Main Steam Line monitor readings are elevated but do not exceed the Alert value but are still rising toward the value. The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess given conditions and determine emergency classification and have knowledge of assigned responsibilities.

F – FISSION PRODUCT BARRIER (MODES 1,2,3,4)

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT
FG1 – LOSS of **ANY** two barriers and the LOSS or POTENTIAL LOSS of the third barrier. Clad RCS Cont	FS1 – LOSS or POTENTIAL LOSS of **ANY** two barriers. Clad RCS Cont	FA1 – **ANY** LOSS or **ANY** POTENTIAL LOSS of **EITHER** the the Fuel Clad or RCS BARRIER. Clad RCS

	Fuel Clad		RCS		**Containment**	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
1.RCS or SG Tube Leakage	▨	A. Core Cooling – Orange entry conditions met.	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube leakage	A. Operation of a standby charging pump is required by **EITHER** of the following: • UNISOLABLE RCS leakage⍰ **OR** • SG tube leakage. **OR** B. Integrity – Red entry conditions met.	A. A leaking or RUPTURED SG is FAULTED outside of containment.	▨
2. Inadequate Heat Removal	A. Core Cooling – Red entry conditions met.	A. Core Cooling – Orange entry conditions met. **OR** B. Heat Sink – Red entry conditions met.	▨	A. Heat Sink – Red entry conditions met.	▨	A. Core Cooling – Red entry conditions met for 15 minutes or longer.
3. RCS Activity / Containment Radiation	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 2100 R/hr. **OR** A2. HATCH MONITOR greater than 4200 mR/hr. **OR** B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.	▨	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr. **OR** A2. HATCH MONITOR greater than 20 mR/hr.	▨	▨	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 45,000 R/hr. **OR** A2. HATCH MONITOR greater than 90,000 mR/hr.
4. Containment Integrity or Bypass	▨	▨	▨	▨	A. Containment isolation is required AND **EITHER** of the following: • Containment integrity has been lost based on Emergency Director judgment. **OR** • UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment – Red entry conditions met. **OR** B. Explosive mixture exists inside containment ($H_2 \geq 4\%$). **OR** C1. Containment pressure greater than 9.5 psig. **AND** C2. Less than one full train of Containment Spray is operating per design for 15 minutes or longer.
5. Other Indications	▨	▨	▨	▨	▨	▨
6. Emergency Director Judgment	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

"S" - System Malfunction (Mode 1, 2, 3, 4)

Table S1: Plant Parameters
• Reactor Power • RCS Level • RCS Pressure • Core Exit Temperature • Levels in at least two steam generators • Steam Generator Auxiliary Feed Water Flow

Table S2: Communications Methods			
METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
• Dedicated Ring-down lines		X	
• ENS line			X

Table S3: Hazardous Events
• Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall • Other events with similar hazard characteristics as determined by the Shift Manager

	AC Power	Control Room Indications	Coolant Activity	RCS Leakage	RPS Failure	Communication	Containment Isolation	DC Power	Hazards	
General Emergency	**SG1** Prolonged loss of **ALL** offsite and **ALL** onsite AC power to emergency buses **SEE NOTE 4** 1a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses. **AND** b. **EITHER** of the following • Restoration of at least one 4160V AC ESF bus in less than 4 hours is not likely • Core Cooling – Red entry condition met							**SG8** Loss of **ALL** AC and Vital DC power sources for 15 minutes or longer **SEE NOTE 1** 1a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF buses for 15 minutes or longer **AND** b. Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer		**General Emergency**
Site Area Emergency	**SS1** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. **SEE NOTE 1** 1. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses for 15 minutes or longer.				**SS5** Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal. 1a. Entry into 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS **AND** b. **ALL** manual actions to shutdown the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • Core Cooling – Red entry conditions met **OR** • Heat Sink- Red entry conditions met Modes 1, 2 Only			**SS8** Loss of **ALL** Vital DC power for 15 minutes or longer **SEE NOTE 1** 1. Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer		**Site Area Emergency**
Alert	**SA1** Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. **SEE NOTE 1** 1. a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer. **AND** b. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. **SEE NOTE 1** 1a. An UNPLANNED event results in the inability to monitor **ANY** of the following parameters in Table S1 from within the Control Room for 15 minutes or longer. **AND** b. **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor trip • ECCS (SI) actuation			**SA5** Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken at the reactor control panels are not successful in shutting down the reactor. 1. Entry into 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS. Modes 1, 2 Only				**SA9** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. 1a. The occurrence of **ANY** of the following hazardous events in Table S3. **AND** b. **EITHER** of the following: • Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.	**Alert**
Unusual Event	**SU1** Loss of **ALL** offsite AC power capability to emergency buses for 15 minutes or longer **SEE NOTE 1** 1. Loss of **ALL** offsite AC power capability to **ALL** three 4160V AC ESF Buses for 15 minutes or longer	**SU2** UNPLANNED loss of Control Room indications for 15 minutes or longer. **SEE NOTE 1** 1. An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer	**SU3** Reactor coolant activity greater than Technical Specification allowable limits. 1. RT-8039 reading greater than 30 μCi/ml 2. Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications. • Greater than 1 μCi/gm Dose Equivalent I-131 **OR** • Greater than 540 μCi/gm Dose Equivalent Xe-133	**SU4** RCS leakage for 15 minutes or longer **SEE NOTE 1** 1. RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer. 2. RCS identified leakage greater than 25 gpm for 15 minutes or longer. 3. Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer	**SU5** Automatic or manual trip fails to shutdown the reactor **SEE NOTE 12** 1 a. An automatic trip did not shutdown the reactor **AND** b. A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor 2 a. A manual trip did not shutdown the reactor **AND** b. **EITHER** of the following: • A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor **OR** • A subsequent automatic trip is successful in shutting down the reactor Modes 1, 2 Only	**SU6** Loss of **ALL** onsite or offsite communications capabilities. 1. Loss of **ALL** of the following Onsite communication methods in Table S2 2. Loss of **ALL** of the following Offsite Response Organization (ORO) communications methods in Table S2 3. Loss of **ALL** of the following NRC communications methods in Table S2	**SU7** Failure to isolate containment or loss of containment pressure control. 1.a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not isolated within 15 minutes of the actuation signal. 2.a. Containment pressure greater than 9.5 psig. **AND** b. No Containment Spray train is operating per design for 15 minutes or longer.			**Unusual Event**

NOTES		
1	CU1, CU2, CU3, CU4, CA1, CA2, HS6, SU1, SU2, SU4, SA1, SA2, SS1, SS8, SG8	The Emergency Director should declare the event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
2	CS1, CG1	The Emergency Director should declare the event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.
3	RU1, RG2	The Emergency Director should declare the event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
4	SG1	The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.
5	RA1, RS1, RG1, RG2, HU4, CA3	The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
6	RA1, RS1, RG1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
7	RU1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
8	RU1, RA1, RS1, RG1	If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer VALID for classification purposes.
9	RA1, RS1, RG1	The pre-calculated effluent monitor values presented in EAL#1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
10	RA3, HA5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
11	HU3	EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
12	SU5	A manual action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving control rods or implementation of boron injection strategies.

H-HAZARDS (ALL MODES)

Security

General Emergency

HG1 HOSTILE ACTION resulting in loss of physical control of the FACILITY.

1a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANY of the following personnel in Table H1
AND
b. EITHER of the following has occurred
- ANY of the following safety functions in Table H2 cannot be controlled or maintained in MODES 1, 2 or 3 ONLY.
  OR
- Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency

HS1 HOSTILE ACTION within the PROTECTED AREA.

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANY of the following personnel in Table H1.

Alert

HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANY of the following personnel in Table H1.
2. A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Unusual Event

HU1 Confirmed SECURITY CONDITION or threat.

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANY of the following personnel in Table H1.
2. Notification of a CREDIBLE SECURITY THREAT directed at the site.
3. A validated notification from the NRC providing information of an aircraft threat.

Table H1: Security Supervision
• Security Force Supervisor • Acting Security Manager • Security Manager

Table H2: Safety Functions
• Reactivity Control • Core Cooling • RCS Heat Removal

Table H3/R2: Plant Areas Requiring Access

MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16 KV Switchgear Rooms

Table H4: Plant Rooms/Areas
• Mechanical/Electrical Auxiliary Building (MEAB) • Fuel Handling Building (FHB) • Reactor Containment Building (RCB) • Essential Cooing Water Intake Structure (ECWIS) • Isolation Valve Cubicle (IVC) • Diesel Generator Building (DGB)

Seismic

Unusual Event

HU2 Seismic event greater than OBE levels.

1.a. EITHER of the following conditions exist.
- "SEISMIC EVENT" alarm in Unit 1 Control Room?
  OR
- Control Room personnel feel an actual or potential seismic event.

AND
b. The occurrence of a seismic event is confirmed in a manner deemed appropriate by the Shift Manager or Emergency Director.

Hazards

Unusual Event

HU3 Hazardous event

SEE NOTE 11

1. A tornado strike within the PROTECTED AREA.
2. Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.
3. Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).
4. A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.
5. Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall.

Fire

Unusual Event

HU4 FIRE potentially degrading the level of safety of the plant.

SEE NOTE 5

1.a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications
- Report from the field (i.e., visual observation)
  OR
- Receipt of multiple (more than 1) fire alarms or indications
  OR
- Field verification of a single fire alarm

AND
b. The FIRE is located within ANY of the plant rooms or areas in Table H4.

2.a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).
AND
b. The FIRE is located within ANY of the plant rooms or areas in Table H4.
AND
c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

3. A FIRE within the ISFSI OR plant PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

4. A FIRE within the ISFSI OR plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Toxic Gas

Alert

HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.

SEE NOTE 10

1.a. Release of a toxic, corrosive, asphyxiant or flammable gas into the Control Room or ANY of the plant rooms or areas listed in Table H3/R2.
AND
b. Entry into the room or area is prohibited or impeded.

Control Room Evacuation

Site Area Emergency

HS6 Inability to control a key safety function from outside the Control Room.

SEE NOTE 1

1.a. An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP).
AND
b. Control of ANY of the following key safety functions in Table H2 is not reestablished within 15 minutes in Modes 1, 2, or 3 ONLY.

Alert

HA6 Control Room evacuation resulting in transfer of plant control to alternate locations.

1. An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP).

Judgement

General Emergency

HG7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the FACILITY. Releases can be reasonably expected to exceed EPA PROTECTIVE ACTION GUIDELINE exposure levels offsite for more than the immediate site area.

Site Area Emergency

HS7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PROTECTIVE ACTION GUIDELINE exposure levels beyond the SITE BOUNDARY.

Alert

HA7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a SECURITY EVENT that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PROTECTIVE ACTION GUIDELINE exposure levels.

Unusual Event

HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to FACILITY protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

R-RADIATION (ALL MODES)

Effluent Release

General Emergency

RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE.

SEE NOTES 5, 6, 8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "GE" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY.
   - Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
     OR
   - Analyses of field survey samples indicate THYROID CDE greater than 5,000 mrem for one hour of inhalation.

Site Area Emergency

RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE.

SEE NOTES 5, 6, 8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "SAE" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY.
   Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
   OR
   Analyses of field survey samples indicate THYROID CDE greater than 500 mrem for one hour of inhalation.

Alert

RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem THYROID CDE.

SEE NOTES 5, 6 ,8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "ALERT" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY for one hour of exposure.
4. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY:
   - Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
     OR
   - Analyses of field survey samples indicate THYROID CDE greater than 50 mrem for one hour of inhalation.

Unusual Event

RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

SEE NOTES 3, 7 & 8

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "UE" for 60 minutes or longer.
2. Reading on gaseous effluent radiation monitor RT-8010B greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.
3. Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.

Spent Fuel Pool

General Emergency

RG2 Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

SEE NOTE 5

1. Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

Site Area Emergency

RS2 Spent fuel pool level at 40'-4" or lower.

1. Lowering of spent fuel pool level to 40'-4" or lower.

Alert

RA2 Significant lowering of water level above, or damage to, irradiated fuel.

1. Uncovery of irradiated fuel in the REFUELING PATHWAY.

2.a Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following FHB radiation monitor readings:
- FHB Exhaust, RT-8035 or RT-8036 greater than 1.00 E-1 µCi/ml
  OR
- ARM (68' FHB), RE-8090 greater than 1,500 mR/hr

OR
b. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following RCB radiation monitors readings (Mode 5 or 6 only):
- ARMs (68' RCB), RE-8055 or RE-8099 greater than 850 mR/hr

3. Lowering of spent fuel pool level to 49'-10" or lower.

Unusual Event

RU2 UNPLANNED loss of water level above irradiated fuel.

1.a UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following
- Visual Observation
  OR
- Annunciator alarm on lampbox 22M02 Window F-5 "SFP WATER LVL HI/LO"
  OR
- Spent fuel in the ICSA AND Annunciator alarm on lampbox 22M02 Window F-6 "SFP Trouble" AND Plant Computer point FCLC1420 "REFLNG CAV LVL IN CNTMT" (ICSA Water Level HI/LO) is in alarm

AND
b. UNPLANNED rise in area radiation levels on ANY of the following area radiation monitors
- RE-8055 (68' RCB) Mode 5 or 6 only
  OR
- RE-8099 (68' RCB) Mode 5 or 6 only
  OR
- RE-8090 (68' FHB)

Abnormal Rad Levels

Alert

RA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

SEE NOTE 10

1. Dose rate greater than 15 mR/hr in ANY of the following areas:
   - Control Room ARM (RE-8066)
     OR
   - Central Alarm Station (CAS) by radiation survey
2. An UNPLANNED event results in radiation levels that prohibit or impede access to ANY of the areas listed in Table H3/R2

Table R1: Effluent Monitors

	Release Point	Monitor	GE	SAE	ALERT	UE
Gas	Unit Vent	RT-8010B	1.50 E+08 µCi/sec	1.50 E+07 µCi/sec	1.50 E+06 µCi/sec	1.40 E+05 µCi/sec
	Steam Lines	RT-8046 thru 8049	4.00 E+02 µCi/ml	4.00 E+01 µCi/ml	4.00 E+00 µCi/ml	5.00 E-02 µCi/ml

E-ISFSI (ALL MODES)

ISFSI

Unusual Event

E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY.

1. Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than:
   - 60 mrem/hr (gamma + neutron) on the top surface of the spent fuel cask
     OR
   - 600 mrem/hr *(gamma + neutron)* on the side surface of the spent fuel cask
     OR
   - 7000 mrem/hr (*gamma + neutron*) on the side surface of the transfer cask.

C - COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTIONS (Mode 5, 6, D)

LOCA
AC POWER
MAINTAIN SHUTDOWN
DC POWER
COMMUNICATION
HAZARDS

General Emergency

CG1 Loss of RCS inventory affecting fuel clad integrity with containment challenged.

SEE NOTE 2

1a. RCS level is less than 0% of plemum for 30 minutes or longer.
AND
b. **ANY** indication from Table C1.

2a. RCS level cannot be monitored for 30 minutes or longer.
AND
b. Core uncovery is indicated by **ANY** of the following:
- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr
  OR
- Erratic source range monitor indication
  OR
- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

AND
c. **ANY** indication from Table C1.

Modes 5, 6 Only

General Emergency

Table C1: Containment Challenge

- CONTAINMENT CLOSURE not established*
- ≥ 4% hydrogen exists inside containment
- UNPLANNED rise in containment pressure

* IF CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute limit, THEN declaration of a General Emergency is not required.

Table C2: RCS Leakage

- Containment Normal Sump
- Pressurizer Relief Tank (PRT)
- Reactor Coolant Drain Tank (RCDT)
- MAB Sumps 1 thru 4
- Containment Penetration Area Sump
- SIS/CSS Pump Compartment Sump

Table C3: RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but not at reduced inventory)	Not applicable	60 minutes*
Not intact (or at reduced inventory)	Established	20 minutes*
	Not Established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

Table C4: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
Plant PA system	X		
Plant Radios	X		
Plant telephone system	X	X	X
Satellite phones		X	X
Direct line from Control Rooms to Bay City		X	X
Microwave Lines to Houston		X	X
Security radio to Matagorda County		X	
Dedicated Ring-down lines		X	
ENS line			X

Table C5: Hazardous Events

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Predicted or actual breach of Main Cooling Reservoir retaining dike along the North Wall
- Other events with similar hazard characteristics as determined by the Shift Manager

Site Area Emergency

CS1 Loss of RCS inventory affecting core decay heat removal capability.

SEE NOTE 2

1a. CONTAINMENT CLOSURE not established.
AND
b. RCS level is less than 33% of plenum.

2a. CONTAINMENT CLOSURE established
AND
b. RCS level less than 0% of plenum.

3a. RCS level cannot be monitored for 30 minutes or longer.
AND
b. Core uncovery is indicated by **ANY** of the following:
- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr
  OR
- Erratic source range monitor indication
  OR
- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

Modes 5, 6 Only

Site Area Emergency

Alert

CA1 Loss of RCS inventory.

SEE NOTE 1

1. Loss of RCS inventory as indicated by level less than 32' 9" (+6 inches above hot leg centerline).

2a. RCS level cannot be monitored for 15 minutes or longer.
AND
b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 due to a loss of reactor vessel/RCS inventory.

Modes 5, 6 Only

CA2 Loss of ALL offsite and ALL onsite AC power to emergency buses for 15 minutes or longer.

SEE NOTE 1

1 Loss of **ALL** offsite **AND ALL** onsite AC Power to **ALL** three 4160V AC ESF Busses for 15 minutes or longer.

CA3 Inability to maintain the plant in cold shutdown.

SEE NOTE 5

1. UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg) for greater than the duration specified in Table C3.
2. UNPLANNED RCS pressure rise greater than 10 psig. (This EAL does not apply during water-solid plant conditions.)

Modes 5, 6 Only

CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

1a. The occurrence of **ANY** of the following hazardous events in Table C5.
AND
b. **EITHER** of the following:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.

  OR
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.

Modes 5, 6 Only

Alert

Unusual Event

CU1 UNPLANNED loss of RCS inventory for 15 minutes or longer.

SEE NOTE 1

1. UNPLANNED loss of reactor coolant results in RCS level below the procedurally required limit for 15 minutes or longer.

2a. RCS level cannot be monitored.
AND
b. UNPLANNED rise in ANY of the following sump or tank levels in Table C2.

Modes 5, 6 Only

CU2 Loss of ALL but one AC power source to emergency buses for 15 minutes or longer.

SEE NOTE 1

1a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer.
AND
b. **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.

CU3 UNPLANNED rise in RCS temperature.

SEE NOTE 1

1. UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg).
2. Loss of ALL RCS temperature and RCS level indication for 15 minutes or longer.

Modes 5, 6 Only

CU4 Loss of Vital DC power for 15 minutes or longer.

SEE NOTE 1

1. Indicated voltage is less than 105.5 VDC on required Vital DC buses for 15 minutes or longer.

Modes 5, 6 Only

CU5 Loss of ALL onsite or offsite communications capabilities.

1. Loss of **ALL** of the following Onsite communication methods in Table C4.
2. Loss of **ALL** of the following Offsite Response Organization (ORO) communication methods in Table C4.
3. Loss of **ALL** of the following NRC communication methods in Table C4.

Unusual Event

NOTES

1	CU1, CU2, CU3, CU4, CA1, CA2, HS6, SU1, SU2, SU4, SA1, SA2, SS1, SS8, SG8	The Emergency Director should declare the event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
2	CS1, CG1	The Emergency Director should declare the event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.
3	RU1, RG2	The Emergency Director should declare the event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
4	SG1	The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.
5	RA1, RS1, RG1, RG2, HU4, CA3	The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
6	RA1, RS1, RG1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
7	RU1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
8	RU1, RA1, RS1, RG1	If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer VALID for classification purposes.
9	RA1, RS1, RG1	The pre-calculated effluent monitor values presented in EAL#1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
10	RA3, HA5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
11	HU3	EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
12	SU5	A manual action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving control rods or implementation of boron injection strategies.

B. IF at **UNUSUAL EVENT** Classification, THEN select appropriate ENRS code: • 0001 Information only – ERO does not report to their facility • 0002 Activate the TSC, OSC, EOF and JIC (All facilities) • 0003 ERO does not respond to the site. Activate alternate TSC and OSC. Activate EOF and JIC	
C. IF at **ALERT OR Higher** Classification, THEN select appropriate ENRS code: • 0002 Activate the TSC, OSC, EOF and staff the JIC (All facilities) • 0003 ERO does not respond to the site. Activate alternate TSC and OSC. Activate EOF and JIC	
D. Direct the ENS Communicator to activate ENRS using the **selected code** per 0EPR01-IP-0002, Emergency Notifications.	
1.2.4. Direct the following notifications be made per 0EPR01-IP-0002, Emergency Notifications: • State / County Agencies within 15 minutes of declaration • NRC upon completion of State / County notification and no later than one hour after declaration	**Tab 3**
1.2.5. Ensure the following on-shift personnel have responded to their emergency duty station: • Dedicated On-Shift SRO • State/County Communicator • ENS Communicator • Shift Technical Advisor • Lead RP Technician • Security Force Supervisor	
1.2.6. Perform or direct a briefing to the following as time permits: • Duty Operations Manager • Duty Plant Manager • Duty NRC Resident Inspector	
NOTES: Assembly and Accountability may be delayed if conduct will place personnel in danger. The Shift Manager may call for Assembly and Accountability at any time. 1.2.7. IF at a Site Area or General Emergency, THEN go to Section 3.2, Assembly and Accountability, Rapid Site Evacuation.	
2.0 ONGOING ACTIONS	
2.1. Actions as Emergency Director	
2.1.1. Continue to evaluate plant conditions against EALs to determine if changes are required to current emergency classification level.	**EAL Wall Charts**

Exam Bank No.: 3224 **Last used on an NRC exam:** Never

SRO Sequence Number: 93

The Unit 1 MAIN TRANSFORMER deluge system inadvertently actuates and can NOT be reset.

- Fire Protection System pressure lowers to 98 psig.
- 15 minutes later the crew isolates the deluge system and Fire Protection System pressure rises to its normal value.

Complete the following:

____(1)____ Fire Pumps started.

Per 0PGP03-ZF-0018, with the MAIN TRANSFORMER deluge system isolated, a Fire Watch _____(2)_____ be established.

A. (1) 2
(2) does not have to be

B. (1) 2
(2) must

C. (1) 3
(2) does not have to be

D. (1) 3
(2) must

Answer: C (1) 3 (2) does not have to be

Exam Bank No.: 3224 **Source:** New **Modified From**

K/A Catalog Number: 086 A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the Fire Protection System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Low FPS header pressure

SRO Importance: 3.3 **Tier:** 2 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

Unique to the SRO position.

STP Lesson: LOT 201.29 **Objective Number:** 91385

Describe the requirements of the following procedures to include purpose, scope, definitions, responsibilities, applicable fire system evolutions, and forms covered by the procedure: 0PGP03-ZF-0018

Reference: 0PGP03-ZF-0018, LOT 201.29

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible with a misunderstanding of start setpoints and a fire watch does not have to be established.

B: INCORRECT: Plausible with a misunderstanding of start setpoints and a fire watch is required for other deluge systems.

C: CORRECT: With Fire Header pressure dropping below 110 psig, all 3 Fire Pumps would start. For a non-Technical Specification deluge system, a Fire Watch is not required.

D: INCORRECT: Plausible as all 3 Fire Pumps would start, and fire watches are required for some deluge systems.

Question Level: H **Question Difficulty** 3

Justification:

Unique to the SRO Position

OBJ #6

Fire Engine Controls

- In “**AUTO**” the Diesel Driven Fire Pump will Auto Start on:
  - Low Fire Water Header Pressure (sequential Fire Pump Starts)
    - #1 starts in 5 seconds (130 psig)
    - #2 starts in 10 seconds (120 psig)
    - #3 starts in 15 seconds (110 psig)
  - Loss of AC Power to Panel
    - Ensures Engine available as batteries could self-discharge
  - Remote Start Signal-either Control Room

4.2 Fire Protection Water Supply System (continued)

5. Fire Pump Diesel Engine Starting Battery and Charger Functional Testing and Inspection Requirements:

   A. Per the Preventive Maintenance Program, the batteries SHALL be checked to verify the following:

      1) VERIFY the battery float voltage is 27.4 (27.1 to 27.7) VDC.

      2) CHECK battery cells 1 and 10 for a temperature less than 120°F.

      3) VERIFY the individual cell voltage ≥ 1.35 VDC.

      4) VERIFY the battery racks and area are clean and free from corrosion.

      5) VISUALLY INSPECT battery cells for cracks and electrolyte leakage.

   B. Per the Preventive Maintenance Program the batteries SHALL be checked to verify the following:

      1) VERIFY battery float current meets the vendor recommendation.

      2) VERIFY the battery charger voltage is 28.4 (28.4 to 28.9) VDC.

      3) VERIFY the electrolyte level Is > the minimum level line on the battery cell.

      4) VERIFY the intercell connections are clean, tight, and free from corrosion.

4.3 Sprinkler/Spray Systems

1. The SPRINKLER or SPRAY SYSTEMS identified in Addendum 2 SHALL be functional whenever the equipment protected by them is required to be operable by Technical Specifications:

2. Exception to the functionality requirement statement above MAY be taken if APPROVED alternative compensatory actions are taken in accordance with 0PGP03-ZF-0016 **OR** applicable compensatory actions, as described below, are taken:

   A. With one or more of the required sprinkler or spray systems impaired/nonfunctional for an area(s) in which redundant systems or components could be damaged, within one hour establish a continuous fire watch(es) with backup suppression equipment.

   B. With one or more of the required sprinkler or spray systems impaired/nonfunctional area(s) that does not contain redundant systems or components, within one hour, establish an hourly fire watch(es).

Automatic Wet Pip Sprinkler Systems (continued)

Area 20	Zone 129	Non-Radiological Pipe Chase, MAB El. 27'-0"
Area 75	Zone 112	Electrical Chase Train C, MAB El. 10'-0" to 60'-0"
Area 03	Zone 116	Non-Radiological Pipe Penetration Area, MAB El. 41'-0"
Area 03	Zone 130	Corridor, MEAB El. 41’-0”
Area 03	Zone 147	Locker Rooms and Clothing Issue, MAB El. 41'-0"
Area 30	Zone 144	Stairwell #2 Foyer, MAB El. 41'-0"
Area 32	Zone 145	Personnel Access to Containment, MAB El. 60'-0" (Partial Coverage)
Area 32	Zone 122	Machine Shop and HVAC Area, MAB El. 60'-0"
Area 59	Zone 801	Fire Pump #1 Cubicle
Area 60	Zone 802	Fire Pump #2 Cubicle
Area 61	Zone 803	Fire Pump #3 Cubicle
Area 61	Zone 804	Fire Pump House Common Area
Area 67	Zone 058	TSC, EAB El. 72'-0"
Area 69	Zone 008	Electrical Chase Train C, EAB El. 10'-0" to 60'-0"
Area 71	Zone 007	Electrical Chase Train B, EAB El. 10'-0"

Area Deluge or Preaction Sprinkler Systems

Main Transformer Deluge is not listed.

Area 63	Zone 207 to 210	RCB Peripheral Area at El. 37'-3"
Area 02	Zone 004	Switchgear Room Train A, EAB El. 10'-0"
Area 03	Zone 042	Switchgear Room Train B, EAB El. 35'-0"
Area 04	Zone 052	Switchgear Room Train C, MAB El. 60'-0"
Area 36	Zone 500	Diesel Generator Train C, Engine Room
Area 37	Zone 501	Diesel Generator Train B, Engine Room
Area 38	Zone 502	Diesel Generator Train A, Engine Room

Foam Water Sprinkler Systems

Area 39	Zone 503	Diesel Generator Train C, Fuel Oil Storage Tank Room
Area 40	Zone 504	Diesel Generator Train B, Fuel Oil Storage Tank Room
Area 41	Zone 505	Diesel Generator Train A, Fuel Oil Storage Tank Room

Exam Bank No.: 3198 **Last used on an NRC exam:** Never

SRO Sequence Number: 94

In accordance with 0POP01-ZA-0049, Conditions Report Operations Evaluation (CROE) an approved _______(1)_______ will be used if contingency operating instructions are needed for use during plant conditions that are **NOT** covered by existing procedures until restoration of normal plant conditions.

Standing Orders and Operating CROEs must be approved by the Cognizant ______(2)______ Manager.

A. (1) Operating CROE
(2) Operations

B. (1) Operating CROE
(1) Shift

C. (1) Standing Order
(2) Operations

D. (1) Standing Order
(2) Shift

Answer: A (1) Operating CROE (2) Operations

Exam Bank No.: 3198 **Source:** New **Modified From** N/A

K/A Catalog Number: G2.1.15 Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.

SRO Importance: 3.4 **Tier:** 3 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(3)

SRO Justification:

The student must know the content and approval process for Condition Report Operations Evaluations.

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrative procedure, DISCUSS the requirements associated with the referenced procedure.

Reference: 0POP01-ZA-0049 pgs 2,3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT. Per Procedure 0POP01-ZA-0049A an approved Operating CROE will be used for the conditions given in the stem. Standing Order can not be used in lieu of, or to modify existing procedures or plant configuration. The Cognizant Operations Manager is the approval authority for Operating CROEs and Standing Orders.

B: INCORRECT. Plausible because the Shift Manager is responsible for being aware of the content of Operating CROEs and implementing them on -shift, but only the Cognizant Operations Manager has authority for their approval. The first part is correct.

C: INCORRECT. Plausible because Standing orders are another type of CROE activity covered by the CROE process. The second part is correct.

D: INCORRECT. Plausible because Standing orders are another type of CROE activity covered by the CROE process and the Shift Manager is responsible for being aware of the content of Operating CROEs and implementing them on -shift, but only the Cognizant Operations Manager has authority for their approval.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of the CROE process.

1.0 Purpose and Scope

1.1 The purpose of this procedure is to identify the requirements for the processing of Condition Report Operations Evaluations (CROE). This procedure applies to the processing of a required change to or variation of any plant doctrine resulting from an identified condition, action or task (e.g. Nonconformance disposition, Maintenance activity affecting the availability of plant systems or components, Operability/Reportability recommendation, Procedure Change Request, etc.) documented in the Corrective Action Program (CAP) database.

1.2 The CROE process provides interim written instructions, which receive the same level of technical review and approval as a plant procedure.

2.0 Definitions

2.1 CONDITION REPORT OPERATIONS EVALUATION (CROE): Formal process for assessing and documenting any required interim change in normal plant operating doctrine.

2.2 CROE ACTIVITY: The interim method used to change existing or establish new doctrine. The format of the CROE Activity will be as described in Form 1, Condition Report Operations Evaluation Form. There are two categories of CROE Activities:

2.2.1 OPERATING CROE: Provides new or revised controls or guidance for plant operation. OPERATING CROEs are used when:

2.2.1.1 There is a need to revise plant operations within normal existing approved procedural guidance or operating envelope.

2.2.1.2 Contingency operating instructions are needed for use during plant conditions NOT covered by existing plant procedures until the restoration of normal plant conditions.

2.2.1.3 Special operating conditions are needed to support plant configuration required for, or as a result of, maintenance.

2.2.2 STANDING ORDER: Management instructions encompassing special shift operations, data taking, plotting process parameters, or other similar matters which have short-term applicability and require dissemination. STANDING ORDERs SHALL not be used in lieu of, or to modify existing procedures or plant configuration.

2.3 EVALUATOR: An individual assigned to determine the required CROE disposition/evaluation necessary to resolve the condition described. The Evaluator is normally the CROE Preparer.

2.4 QUALIFIED REVIEWER: An individual who is a qualified 10CFR50.59 Screener, as prescribed by 0PGP05-ZA-0002, 10CFR50.59 Evaluations and 10CFR72.48 Screener, as prescribed by 0DCS02-ZN-0002, 10CFR72.48 Screening and Evaluations as applicable.

3.0 Responsibilities

3.1 The Operations Division Manager-Production Support and Programs is the owner of the Condition Report Operations Evaluation Program.

3.2 Each Cognizant Operations Manager has the approval authority for any CROE activity.

3.3 The Procedure Supervisor is responsible for:

3.3.1 Initiating an OPERATING CROE action and assigning a responsible person to perform as the CROE Evaluator (designee may perform this function).

3.3.2 Controlling access to the CROE controlled directory.

3.3.3 Maintaining the effective CROE and CROE index current in the access controlled directory.

3.3.4 Performing a periodic review of effective CROEs.

3.4 The Cognizant Operations Manager is responsible for processing STANDING ORDERS in accordance with Step 5.3.

3.5 The CROE Evaluator is responsible for determining the required CROE disposition/evaluation necessary to resolve the condition described in the initiating CR.

3.6 The CROE Evaluator SHALL process the CROE activity in accordance with Section 5.0 of this procedure using Form 1, Condition Report Operations Evaluation Form.

3.7 Each Operations Shift, Unit, and Field Supervisor, as well as each Operations watchstander is responsible for reviewing the CROE Index for CROEs applicable to their watchstation each shift to ensure cognizance of their existence and compliance with their contents.

4.0 Requirements

4.1 General Requirements:

4.1.1 Condition reports SHALL be processed in accordance with 0PGP03-ZX-0002A, CAQ Resolution Process, or 0PGP03-ZX-0008, Condition Not Adverse to Quality (CNAQ) Resolution Process. This procedure may be used in parallel to process associated actions or tasks.

4.1.2 All CROEs SHALL only be initiated via an action or a task associated with the condition report (CR) that is related to the concern requiring the CROE. The CROE SHALL be tracked and identified by this action or task number.

4.1.3 Actions required to address a cited condition (e.g. FSAR discrepancy, setpoint error, configuration management concern, etc.) SHALL be processed in accordance with the applicable governing procedures. The Condition Reporting Process is to be utilized for documenting and tracking all actions requiring implementation of this procedure, even if the applicable governing procedure for that action does **NOT** address the Condition Reporting Process.

Exam Bank No.: 2862 **Last used on an NRC exam:** 2019

SRO Sequence Number: 95

Per 0POP08-FH-0009, Core Refueling, the Core Loading Supervisor is responsible for…

A. preparing Fuel Transfer Forms (FTF).

B. assigning refueling crews to operate the refueling machine.

C. providing appropriate radiological coverage on the refueling machine.

D. granting approval to the Refueling Machine Operator to disengage from a fuel assembly.

Answer: D granting approval to the Refueling Machine Operator to disengage from a fuel assembly.

Exam Bank No.: 2862 **Source:** Bank **Modified From** N/A

K/A Catalog Number: G2.1.35 Knowledge of the fuel-handling responsibilities of SROs

SRO Importance: 3.9 **Tier:** 3 **Group/Category:** 1

10CFR Reference or SRO Objective: 55.43(b)(7)

SRO Justification:

This responsibility is unique to the SRO position.

STP Lesson: LOT 801.01 **Objective Number:** 67635

Discuss the requirements of the Core Refueling, 0POP08-FH-0009, to include: A. Prerequisites, B. Notes and Precautions, C. Major Procedural Steps, D. Checklists

Reference: 0POP08-FH-0009, Step 5.27, 0PEP02-ZM-0005 step 5.2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as per 0POP08-FH-0009, step 3.2 the CLS must concur with changes to an approved FTF but is not responsible for preparing the changes.

B: INCORRECT: Plausible as per 0POP08-FH-0009, step 3.3 the CLS supervises the refueling crews but the Unit Operations Manager assigns them.

C: INCORRECT: Plausible as the CLS is directly involved with refueling but as per 0POP08-FH-0009, step 3.5 this is performed by the HP Manager.

D: CORRECT: The CLS is responsible for granting approval to the Refueling Machine Operator to disengage from a fuel assembly per 0POP08-FH-0009, step 5.27.

Question Level: F **Question Difficulty** 3

Justification:

The student must recall the responsibilities of the Core Loading Supervisor.

5.1.1.10 Attach the transfer sheets, Form 2, and Form 3 (if applicable) to the Fuel Transfer Form Cover Sheet (Form 1) and sign Form 1.

5.1.1.11 Submit the FTF to the Verifier.

5.1.2 The Verifier SHALL:

5.1.2.1 Review the FTF, verify the fuel transfer sheets and FTF Checklist, and ensure that Form 3 (or computer code output) is completed and verified (if applicable).

5.1.2.2 Refer to Addendum 6, SFP Storage Configuration Development (Reference 6.31).

5.1.2.3 Refer to Addendum 7, Transfer Canal Gate Requirements (if applicable, References 6.11, 6.32, and 6.34).

5.1.2.4 Sign and date the Fuel Transfer Form Checklist (Form 2) AND initial each page of the transfer sheets.

5.1.2.5 Forward the FTF to the Reactor & Fuel Engineering Supervisor.

5.1.3 The Reactor & Fuel Engineering Supervisor SHALL review and approve the FTF, and then sign the FTF cover sheet.

5.1.4 Each person moving fuel assemblies, verifying placement of fuel assemblies, or step verifying SHALL attend a briefing covering 0PGP03-ZO-0047. Attendance of the briefing SHALL be documented on the 0PGP03-ZO-0047 briefing attendance sheet.

5.1.5 Attach the 0PGP03-ZO-0047 briefing attendance sheet to the FTF.

5.1.6 Forward the FTF to the Shift Manager.

5.1.7 The Shift Manager SHALL review the FTF. IF the transfer(s) is disapproved, THEN the FTF SHALL be returned to the Reactor & Fuel Engineering Supervisor, along with the reason for the disapproval.

5.1.8 Upon approving the FTF, the Shift Manager SHALL sign the FTF cover sheet.

5.2 Fuel Transfer Form (FTF) Revision

5.2.1 IF a transfer sheet needs to be revised prior to the movement in question, THEN Reactor Engineering shall perform the following:

5.2.1.1 Make the necessary changes.

5.2.1.2 Review the movements that have taken place, the movements that will take place.

5.2.1.3 Review the FTF checklist.

5.2.1.4 IF a Core Loading Supervisor has been assigned for the fuel assembly movement, THEN the preparer SHALL obtain verbal concurrence of the revision from the Core Loading Supervisor.

5.2.1.5 Enter the Revision number and a description of the changes on Form 1 of the FTF.

5.2.1.6 Annotate each changed page of the FTF with the revision number.

5.2.1.7 Re-initial the affected pages of the FTF and sign Form 1 of the FTF.

5.2.1.8 The Verifier SHALL verify the FTF by performing the following:

a. Verify the revisions including the movements that have taken place and the movements that will take place.

b. Review the FTF checklist.

c. Re-initial the affected pages of the FTF.

5.2.1.9 The Reactor & Fuel Engineering Supervisor SHALL approve the FTF revision and sign Form 1 of the FTF.

5.2.1.10 The Shift Manager SHALL approve the FTF revision and sign Form 1 of the FTF.

5.2.2 IF a transfer sheet needs to be corrected after the movement in question (i.e. fuel assembly moved to an incorrect location), THEN refer to the section in 0PGP03-ZO-0047 on Problem Reporting.

5.3 Use of the approved Fuel Transfer Form

5.3.1 The fuel movement performer, verifier, and step verifier SHALL follow 0PGP03-ZO-0047.

5.3.2 IF allowed by the FTF, THEN the performer, verifier, step verifier, or Reactor Engineering may mark movement steps as not applicable (N/A).

5.17 IF a valid high flux at shutdown alarm occurs, THEN all personnel in containment SHALL immediately exit the containment in a safe and orderly manner. A P.A. announcement should be made to inform personnel in containment if an evacuation is required.

5.18 IF a valid high flux at shutdown alarm occurs AND a fuel assembly is still attached to the Refueling Machine, THEN the Core Loading Supervisor and the Refueling Machine Operator may remain at their post long enough to remove the assembly from the core and place it in the RCCA Change Fixture, Fuel Transfer System Upender, or an In-Containment Storage Area cell.

5.19 WHEN the cause of a valid high flux at shutdown alarm is identified AND the Core Loading Supervisor deems that conditions are safe for all personnel, THEN Core Alterations may continue.

5.20 A valid extended range Core Monitoring NI doubling alarm should be evaluated according to Step 5.5.1 and 5.5.2.

5.21 IF there is a malfunction OR suspicion of malfunction of any fuel handling equipment, THEN all fuel movement SHALL be terminated AND the Core Loading Supervisor SHALL be notified.

5.22 Handling tools SHALL NOT be left attached to fuel during planned work stoppages.

5.23 WHEN moving fuel assemblies between the fuel transfer canal and the spent fuel pool, THEN special care SHALL be used to ensure the fuel assembly does NOT make contact with the transfer canal gate walls.

5.24 IF during the course of refueling fuel damage occurs, THEN ALL refueling operations SHALL be immediately stopped AND the Core Loading Supervisor notified. This step does NOT apply to damage identified during core offload visual checks.

5.25 IF an accident involving fuel assemblies occurs, THEN personnel should go to 0POP04-FH-0001, “Fuel Handling Accident”.

5.26 The Refueling Machine Operator SHALL NOT lower a fuel assembly into the Reactor Vessel until the ICRR data from loading the previous fuel assembly has been evaluated.

5.27 The Refueling Machine Operator SHALL NOT disengage from any assembly in the core until nuclear instrumentation indicates safe conditions AND approval to disengage is received from the Core Loading Supervisor.

5.28 WHEN irradiated fuel is being transported through the fuel transfer tube, THEN high radiation conditions may exist in the FHB penetration space in the vicinity of the fuel transfer tube. Access to this area SHALL be controlled per 0PRP07-ZR-0016, “Lockdown and Posting for Transfer of Spent Fuel/Irradiated Material Through Transfer Tube”.

6.2.10 WHEN moving a fuel assembly in the RCB to a core location, THEN PERFORM the following:

6.2.10.1 Prior to lowering a fuel assembly over the core, VERIFY that the Reactor Engineer has completed all required count data from the previously loaded fuel assembly.

6.2.10.2 VERIFY fuel assembly will not become entangled with the shoehorn cabling prior to lowering the fuel assembly in the core.

CAUTION

The Refueling Machine Operator SHALL NOT disengage from the fuel assembly until nuclear instrumentation indicates stable conditions and approval to disengage is received from the Core Loading Supervisor.

6.2.10.3 NOTIFY a Licensed Control Room Operator to monitor the Core Monitoring NI channels during and following the insertion of each fuel assembly into the core AND the Core Loading Supervisor SHALL be notified of the stability or instability of the count rate.

a. IF the count rate is unstable, THEN DIRECT the Refueling Machine Operator to WITHDRAW the fuel assembly AND suspend core alterations.

b. IF the count rate is stable, THEN DIRECT the Refueling Machine Operator to unlatch the fuel assembly.

6.2.10.4 NOTIFY the Reactor Engineer to obtain count rate data from the Core Monitoring NI channels by following the instructions in Addendum 2.

6.2.11 When core loading is complete, logging on Forms 2 and 3 should be suspended.

6.2.12 WHEN refueling is complete, THEN PERFORM a Level 1 Inventory of the Reactor Vessel and core alignment check per 0PEP02-ZM-0004, “Physical Inventory of Fuel Assemblies”.

Exam Bank No.: 3190 **Last used on an NRC exam:** Never

SRO Sequence Number: 96

Unit 1 is at 100% power.

- 0PSP03-AF-0001, Auxiliary Feedwater Pump 11(21) Inservice Test is being performed as a post maintenance test following repair to the pump bearing.
- A procedure deficiency is found which requires correction to complete the test and declare the pump operable before the LCO action time expires.
- The change does **NOT** alter the intent of the procedure or conflict with Technical Specifications.

In accordance with 0PAP01-ZA-0102, Plant Procedures, the change should be processed as a ___(1)___. The change becomes effective when it is approved by ___(2)___.

A. (1) Field Change Request
(2) the Cognizant Manager

B. (1) Field Change Request
(2) a Senior Reactor Operator

C. (1) Temporary Procedure
(2) the Cognizant Manager

D. (1) Temporary Procedure
(2) a Senior Reactor Operator

Answer: B (1) Field Change Request (2) Senior Reactor Operator

Exam Bank No.: 3190 **Source:** Modified **Modified From** 790

K/A Catalog Number: G2.2.6 Knowledge of the process for making changes to procedures.

SRO Importance: 3.6 **Tier:** 3 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(3)

SRO Justification:

Student must know SRO responsibilities in the administrative procedure for making Field Changes to procedures.

STP Lesson: LOT 802.15 **Objective Number:** SRO-71029

Given a proposed Field Change Request, IDENTIFY individuals authorized to approve the Field Change.

Reference: : LOT 802.15 PPT slide 13; 0PAP01-ZA-0102, Section 22.0

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible because the Cognizant Manage approves procedures, however the change becomes effective when a Senior Reactor Operator approves it.

B: CORRECT. Per 0PAP01-ZA-0102 a Field Change is processed for a technical procedure in which the lack of the change would delay in-process or critical evolutions important to the plant and does not alter the intent or conflict with Technical Specifications. A Field Change becomes effective when it is approved by a Senior Reactor Operator.

C: INCORRECT. Plausible as a Temporary Procedure is defined by 0PAP01-ZA-0102 as written for a temporary situation or condition that is not expected to recur. The the Cognizant Manage approves procedures, however the change becomes effective when a Senior Reactor Operator approves it.

D: INCORRECT. Plausible as a Temporary Procedure is defined by 0PAP01-ZA-0102 as written for a temporary situation or condition that is not expected to recur. The second part is correct.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of the process for approving changes to plant procedures.

SOURCE FOR SRO #96

Exam Bank No.: 790 Last used on an NRC exam: 2020

A Field Change (FC) is being written against a Plant Operating Procedure that does NOT require review by the Plant Operations Review Committee (PORC).

In accordance with 0PAP01-ZA-0102, Plant Procedures, when is the FC considered "Effective?"

As soon as it is approved by the…

A. Plant Manager.

B. Cognizant Manager.

C. Technical Reviewer.

D. Senior Reactor Operator.

Answer: D Senior Reactor Operator

Exam Bank No.: 790 **Source:** Bank **Modified from:**

K/A Catalog Number: G2.2.6 Knowledge of the process for making changes to procedures.

RO/SRO Importance: 3.0 / **Tier:** 3 **Group/Category:**

RO-10CFR55.41 # 10 **SRO-10CFR55.43 #** or **SRO Obj:**

SRO Justification:

STP Lesson: LOT 507.01 **Objective Number:** 92183

Given the title of an administrative procedure, IDENTIFY the individuals (by job title) with specific responsibilities in the procedure.

Reference: 0PAP01-ZA-0102, Step 22.3.4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible because if the procedure was required to be reviewed by PORC then the final signature would come from the Plant Manager.

B: INCORRECT: Plausible because the cognizant manager must give concurrence to the FC preparer and sign Form 7. This is done before it is sent for approval.

C: INCORRECT: Plausible because the Technical Reviewer is required to sign the FC but the FC is not effective until signed by the SRO.

D: CORRECT: The SRO approves the field change and this is when it is effective.

Level F **Difficulty** 2

Justification:

Student must have fundamental knowledge of the plant procedure for field changes.

NOTE

The Electronic Procedure Routing and Processing application encompasses this procedure and 0PAP01-ZA-0105.

21.0 Department Instructions and Handbooks

1. Department Instructions and Handbooks are written per 0PAP01-ZA-0105, Department Instruction Guidelines.

NOTE

Field Changes are intended for field work restrained by the need for procedure change.

22.0 Field Changes

22.1 Field Change Evaluation

1. A Field Change (FC) SHALL NOT be used to change a procedure unless ALL of the following criteria are met:
2. The intent of the procedure is not changed (as defined in Step 2.2.11, ANSI 18.7-1976).
3. The change does not conflict with the Technical Specifications or bases.
4. The Cognizant Manager determines that the FC is needed based on any one of the following guidelines.
   - A. The need for the change is discovered for a technical procedure and the lack of a FC would delay in-process or critical evolutions important to the plant.
   - B. The need to correct a compliance or adherence problem or resolve a licensing issue is discovered.
   - C. A FC has been approved to a procedure on another unit and/or train and a FC is needed to resolve procedure differences between units and/or trains.
   - D. The need to correct a configuration management item.

22.2 Preparation of FCs

1. Obtain a FC number from Document Control and enter it on Form 7 [Rev. 1], Field Change. During DC off duty hours, obtain the number from the Unit 1 Shift Supervisor.
2. IF the scope of the procedure change affects related procedures (other unit, train, channel, etc.), THEN FCs to those procedures may be necessary. Evaluate the impact on related or similar procedures. Process all affected procedures together or initiate a Condition Report(s) to track necessary actions.

22.2 Preparation of FCs (continued)

14. Deliver the FC package to an independent Technical Reviewer. The independent Technical Reviewer SHALL: (LER 88-013, LER 88 034, SPR 880024, SER 88-001)
    - A. Ensure that any word-processed pages exactly reflect the latest approved version of the procedure, including incorporation of old FCs. (SPR 920018)
    - B. Use Form 4 [Rev. 3], Technical Review Checklist. (SPR 880076)
    - C. Sign, print name and date Form 7 [Rev. 1] and return the FC package to the Preparer.
15. Prior to approval of ALL FCs, the Preparer SHALL obtain concurrence from the Cognizant Manager on Form 7 [Rev. 1], Field Change. This concurrence may be obtained by telecon.
16. A PDF generated from the procedure's effective Word document copy with the incorporated FC MAY be sent to Document Control for processing of the FC.

22.3 Implementation Approval of Field Changes

1. Deliver the FC package to a Senior Reactor Operator for review and approval. The Senior Reactor Operator SHALL: (SER 88-001)
    - A. Ensure that the FC meets the criteria of Section 22.1.
    - B. Ensure that the FC can be implemented as written.
    - C. Review and sign the LCR form or ensure a Qualified Reviewer signs the "Reviewed By" block on the LCR form, if applicable.
    - D. Review and sign the 10CFR72.48 Applicability and Screening form or ensure a Qualified Reviewer signs the "Reviewed By" block on the 10CFR72.48 Applicability and Screening, if applicable.
    - E. Review and sign the SCR form or ensure a Qualified Reviewer signs the "Reviewed By" block on the SCR form, if applicable.
    - F. IF the FC is acceptable, THEN complete Block 13 of Form 7 [Rev. 1].
2. For special process welding procedures, Design Engineering SHALL co-sign with the Senior Reactor Operator.
3. IF the FC does not require PORC review in accordance with 0PAP01-ZA-0104, Plant Operations Review Committee, THEN "N/A" Plant Manager block on Form 7 [Rev. 1], Field Change.
4. The FC is effective as soon as it is approved by the Senior Reactor Operator. No changes may be made to the body of the FC after approval. Problems identified after approval require a new FC to correct, except for administrative problems which may be corrected with addition of material (e.g., revision bars, FC numbers) only. (SPR 880226, SPR 920018)

- SRO reviews for approval
  - Sign LCR, ISFSI, and SCR (as qualified reviewer or ensure a qualified reviewer signs)
  - When acceptable, SRO signs for approval
  - Design Engineering co-signs for special process welding procedures
  - If PORC review not required (IAW 0PAP01-ZA-0104, Plant Operations Review Committee), Then N/A the plant manager block3
  - FC is effective as soon as it is signed by the SRO

- Problems identified after approval require a new FC to correct

- The preparer is responsible for updating working copies and providing document control the original FC package

Exam Bank No.: 2935 **Last used on an NRC exam:** 2020

SRO Sequence Number: 97

The crew is performing an in-service test on a safety related pump as part of post-maintenance operability testing.

- One of the vibration readings falls into the ALERT range.

Per 0PGP03-ZE-0022, Inservice Testing Program for Pumps, the Shift Manager should determine that the pump will _________(1)_________.

Testing frequency of the pump will ________(2)________.

A. (1) remain OPERABLE
(2) remain the same

B. (1) remain OPERABLE
(2) be doubled

C. (1) be declared INOPERABLE
(2) remain the same

D. (1) be declared INOPERABLE
(2) be doubled

Answer: B (1) remain OPERABLE (2) be doubled.

Exam Bank No.: 2935 **Source:** Bank **Modified From** N/A

K/A Catalog Number: G2.2.12 Knowledge of surveillance procedures

SRO Importance: 4.1 **Tier:** 3 **Group/Category:** 2

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

Unique to the SRO position.

STP Lesson: LOT 802.17 **Objective Number:** SRO-60145

Describe the responsibilities of the individual performing the review and second review of a surveillance test, actions required if acceptance criteria are not met, Alert Range is reached, or Required Action Range is reached as per 0PGP03-ZE-0004, 0PGP03-ZE-0021 and 0PGP03-ZE-0022.

Reference: 0PGP03-ZE-0022,

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: This is the correct operability decision, and if the student believes that ALERT, although serious, does not warrant doubling the frequency.

B: CORRECT: With vibration in the ALERT range, the pump will still remain operable, but be subject to increased frequency testing.

C: INCORRECT: Plausible as ALERT indicates a degraded condition that could cause the student to believe this equates with inoperability, and if the student believes that further testing is not necessary as maintenance will be performed.

D: INCORRECT: Plausible as ALERT indicates a degraded condition that could cause the student to believe this equates with inoperability.The second part is correct.

Question Level: H **Question Difficulty** 3

Justification:

The student must assess the information and determine operability.

2.0 Definitions

2.1 ALERT RANGE: That range for a given pump quantity, outside the normal operating range, for which an increased testing frequency is required. See Addendum 2.

2.2 GROUP A PUMP: A pump that is operated continuously or routinely during normal operation, cold shutdown, or refueling operations. (ISTB-2000) Pump classifications are identified in the IST Plan.

2.3 GROUP B PUMP: A pump in standby systems that is not operated routinely except for testing. (ISTB-2000) Pump classifications are identified in the IST Plan.

2.4 INSERVICE TEST (IST): A special test, performed in accordance with the ASME OM Code, to determine, through measurement and/or observation, the operational readiness of a pump.

2.5 IST COORDINATOR – Owner of the IST Program, also known as IST Program Engineer. (Use of the older term "Section XI Coordinator" is also acceptable and is often used in other plant procedures)

2.6 IST PUMP LIST: A list of pumps included in the controlled Unit 1/2 Pump and Valve Inservice Test Plan, which is subject to the requirements of the ASME OM Code.

2.7 IST BASES DOCUMENT: Documents the basis for the inclusion or exclusion of Class 1, 2, or 3 components in the IST Program.

2.8 IST DATABASE: An electronic database containing component test history, reference values, acceptance criteria and trending capabilities that is available to site personnel. The database is located in Oracle and is a level 2 database controlled by 0PGP07-ZA-0014, Software Quality Assurance. Desktop instructions will be used to guide the entry and verification of data in the database.

2.9 REFERENCE VALUES: One or more fixed sets of values for the quantities shown in Addendum 1, as measured or observed when the pump is determined to be operating within design limits. Reference values shall be at points of operation readily duplicated during subsequent inservice testing. (ISTB-3300)

2.10 REFERENCE POINT: A point of operation at which Reference Values are established and inservice test parameters are measured for comparison with applicable Acceptance Criteria.

2.11 REQUIRED ACTION RANGE: That range for given pump quantity, outside the upper and lower limits of the Alert Range, in which the pump is considered inoperable until further action is taken. See Addendum 2.

4.1.3 Major elements of an Appendix V Test.

4.1.3.1 Identify those certain pumps with specific design basis flow rates in the Owner's credited safety analysis (e.g. technical specifications, technical requirements program, or updated safety analysis report) for inclusion in this program .

4.1.3.2 Perform the pump periodic verification test at least once every 2 yr.

4.1.3.3 Determine whether the pump periodic verification test is required before declaring the pump operable following replacement, repair, or maintenance on the pump.

4.1.3.4 Declare the pump inoperable if the pump periodic verification test flow rate and associated differential pressure cannot be achieved.

4.1.3.5 Maintain the necessary records for pump periodic verification tests, including the applicable test parameters (e.g. flow rate and associated differential pressure and speed for variable speed pumps) and their basis.

4.1.3.6 Account for the pump periodic verification test instrument accuracies in the test Acceptance Criteria.

Appendix V Footnote:

This Mandatory Appendix contains requirements to augment the rules of Subsection ISTB, Inservice Testing of Pumps in Light Water Reactor Nuclear Power Plants. The Owner is not required to perform a pump periodic verification test, if the design basis accident flow rate in the Owner's safety analysis is bounded by the comprehensive pump test or Group A test.

4.1.4 If a pump is replaced or a repair is made which may have affected the pump's performance, as determined by the cognizant System Engineer or IST Coordinator, an inservice or reference values measurement test must be performed on the affected pump prior to declaring the pump operable. (ISTB-3310)

4.1.5 If any pump quantity measured during an inservice test falls within the Alert Range, the frequency of testing shall be doubled until the cause of the deviation is determined and the condition corrected, or an analysis of the pump is performed and the new reference values are established in accordance with ISTB-6200(c). [ISTB-6200(a)]

Exam Bank No.: 2847 **Last used on an NRC exam:** 2019

SRO Sequence Number: 98

In order to take an action to protect large populations during a large break LOCA, an operator needs to enter an area with high radiation levels:

- The area has a dose rate of 66 Rem/ hour.
- The operator will be in the area for 20 minutes.
- The TSC is NOT staffed.

Per 0EPR01-IP-0004, Emergency Exposure Controls, the operator ____(1)___ be a volunteer. This radiation exposure must be approved by the ____(1)_____.

A. (1) does not have to
(2) Radiation Protection Manager

B. (1) does not have to
(2) Emergency Director

C. (1) must
(2) Radiation Protection Manager

D. (1) must
(2) Emergency Director

Answer: B (1) does not have to (2) Emergency Director

Exam Bank No.: 2847 **Source:** Bank **Modified From**

K/A Catalog Number: G2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

SRO Importance: 3.7 **Tier:** 3 **Group/Category:**

10CFR Reference or SRO Objective: 55.43(b)(4)

SRO Justification:

Unique to the SRO position.

STP Lesson: EPT 003.00 **Objective Number:** 4

Discuss radiation exposure controls associated with emergency conditions. Include emergency dose guidelines and access requirements.

Reference: 0EPR01-IP-0004

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT: Plausible as the individual does not have to be a volunteer, and the Radiation Protection Manager does approvals when the TSC is staffed.

B: CORRECT: The operator would receive 22 rem. This requires ED approval but under 25 rem, the individual does not have to be a volunteer.

C: INCORRECT: Plausible with confusion about the different levels covered in the procedure and the RPM does approvals when the TSC is staffed.

D: INCORRECT Plausible with confusion about the different levels covered in the procedure. This is the correct approval authority.

Question Level: H **Question Difficulty** 3

Justification:

The student must recall the requirements for emergency exposure, do a calculation, and apply the requirements to the situation.

3.6 EOF Radiation Protection Coordinator

3.6.1 Approve requests for exposure extensions up to 10CFR20 limits for offsite STPEGS personnel following ERO activation.

3.6.2 Monitor offsite STPEGS personnel exposure when individuals are authorized to exceed normal administrative levels.

3.6.3 Evaluate the need for KI of offsite STPEGS personnel.

NOTE

Declaration of an emergency allows personnel exposure controls to switch from 10CFR20 occupational worker limits to EPA-400 emergency worker limits. Decision makers can elect to continue to follow administrative or 10CFR20 exposure limits at any classification level as radiological conditions allow to maintain As Low as Reasonably Achievable (ALARA) principles.

4.0 PROCEDURE

4.1 General Guidelines

4.1.1 Ensure individual(s) have been issued a Dosimeter of Legal Record (DLR).

4.1.2 Verify considerations have been made to reduce radiation exposure to ALARA.

4.1.3 Determine the amount of radiation exposure to complete the task.

4.2 Emergency Exposure

NOTES

Emergency exposure limits are exclusive of current occupational exposure.

Dose to lens of the eye is limited to three times listed value.

Dose to other organs, including skin and body extremities, is limited to ten times listed value.

4.2.1 Determine the appropriate emergency Total Effective Dose Equivalent (TEDE) exposure level using the following:

TEDE Limit (Rem)	Activity
5	All activities during the emergency.
10	Protecting valuable property when lower dose is not practicable.
25	Lifesaving or protection of large populations when lower dose is not practical.
Greater Than 25	Lifesaving or protection of large populations, only if individuals receiving exposure is a volunteer, and fully aware of risks involved.

Exam Bank No.: 3188 **Last used on an NRC exam:** Never

SRO Sequence Number: 99

Unit 1 has experienced a reactor trip and safety injection.

- Critical Safety Functions (CSFs) are now being monitored and 0POP05-EO-EO00, Reactor Trip or Safety Injection, Addendum 5, Verification of SI Equipment Operation is complete.
- The following conditions are observed:
    - Upper Range Flux is 1% and slowly rising
    - SUR is +0.1 Decades/Minute
    - RCS Cold Leg Temperatures lowered from 562ºF to 200ºF over 30 minutes
    - RCS Pressure is 1500 psig and rising

The Unit Supervisor should address the ___(1)____ CSF first because _____(2)____.
(Reference provided)

A. (1) SUBCRITICALITY
(2) it is the highest priority ORANGE path CSF

B. (1) SUBCRITICALITY
(2) it is the highest priority RED path CSF

C. (1) INTEGRITY
(2) it is the highest priority ORANGE path CSF

D. (1) INTEGRITY
(2) it is the highest priority RED path CSF

Answer: D (1) INTEGRITY; (2) it is the highest priority RED path CSF

STP LOT-26 NRC SRO EXAM

Exam Bank No.: 3188 **Source:** New **Modified From** N/A

K/A Catalog Number: G2.4.22 Knowledge of the bases for prioritizing safety functions during abnormal/ emergency operations.

SRO Importance: 4.4 **Tier:** 3 **Group/Category:** 4

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

The student must have knowledge of administrative procedures that specify hierarchy and implementation, to determine the highest priority safety function.

STP Lesson: LOT 504.04 **Objective Number:** 80488

STATE the basis for monitoring the Critical Safety Function Status Trees.

Reference: : 0POP01-ZA-0018, 0POP05-EO-FO01, 0POP05-EO-FO04

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☑ **Attachment:** 0POP05-EO-FO04 Addendum 1

Distractor Justification

A: INCORRECT. Plausible because the student could determine that both CSFs are in an Orange path condition In that case Subcriticality is the highest priority CSF.

B: INCORRECT. Plausible because the student could determine that both CSFs are in a Red Path condition.. In that case Subcriticality is the highest priority CSF.

C: INCORRECT. Plausible because the student could determine that Subcriticality is in a Yellow path condition and that Itegrity is in an Orange path which would make Integrity the highest prority Orange path CSF.

D: CORRECT. For the conditions given, the Integrity CSF is in a RED path condition and the Subcriticality CSF is in an ORANGE path condition. In accordance with 0POP01-ZA-0018, the highest prioity RED path should be implemented first.

Question Level: H **Question Difficulty** 3

Justification:

The student has to analyze conditions given and have knowledge of the basis for prioritizing Critical Safety Functions.

6.0 CSF Status Tree Usage

NOTE

- The verification of SI Equipment Operation (addendum 5 of 0POP05-EO-EO00 and similar addenda in 0POP05-EO-FRH1 and 0POP05-EO-FRS1) ensures proper operation of ESF equipment necessary for mitigation of events addressed by the Emergency Operating Procedures.
- Once started, the completion of Addendum 5 of 0POP05-EO-EO00 is a priority to ensure that timely transition to Function Restoration Procedures can occur when necessary.
- The term "Monitor" in regards to use of CSF Status Trees implies that the Status Trees are monitored and appropriate FRPs are implemented per the guidance of section 6.0 and 7.0.

6.1 Monitoring of the CSF Status Trees **SHALL** begin:

- When directed in 0POP05-EO-EO00, Reactor Trip OR Safety Injection, to begin monitoring CSF status trees.
- When exiting 0PO05-EO-EO00, Reactor Trip Or Safety Injection.

  Exception: IF 0POP05-EO-EO00, Reactor Trip Or Safety Injection, is exited at Step 4 due to safety injection **NOT** actuated or required AND subsequent plant conditions results in safety injection being actuated while still performing steps within 0POP05-EO-ES01, Reactor Trip Response, THEN 0POP05-EO-EO00, Reactor Trip Or Safety Injection **SHALL** be reentered at Step 1 and discontinue CSF Status Tree monitoring. The CSF **SHALL NOT** be implemented until re-directed to monitor CSF in 0POP05-EO-EO00.

- IF 0POP05-EO-EO00, Reactor Trip Or Safety Injection, is exited at Step 1 due to an inability to trip the Reactor OR at Step 7 due to an inability to establish an adequate Heat Sink, THEN begin monitoring CSF status trees. When 0POP05-EO-FRS1 OR 0POP05-EO-FRH1 is exited back to 0POP05-EO-EO00, THEN applicable FRPs **SHALL** be implemented.
- When 0POP05-EO-EC00, Loss Of All AC Power, is entered, THEN begin monitoring CSF status tree for information only:
  - FRPs **SHALL NOT** be implemented until directed to do so in 0POP05-EO-EC01, Loss Of All AC Power Recovery Without SI Required; OR 0POP05-EO-EC02, Loss Of All AC Power Recovery With SI Required.
  - IF 0POP05-EO-EC00, Loss of All AC Power, is exited at Step 6 or 7 back to 0POP05-EO-EO00, THEN applicable FRPs **SHALL** be implemented.

6.2 WHEN 0POP05-EO-EO00, Reactor Trip Or Safety Injection, is entered and Addendum 5 is **started**, THEN ORPs may be transitioned to AND implemented as directed in 0POP05-EO-EO00 while Addendum 5 is still in progress.

6.3 CSF Status Trees are listed below in order of priority:

SUBCRITICALITY	(S)	(i.e., 0POP05-EO-FO01)
CORE COOLING	(C)	(i.e., 0POP05-EO-FO02)
HEAT SINK	(H)	(i.e., 0POP05-EO-FO03)
INTEGRITY	(P)	(i.e., 0POP05-EO-FO04)
CONTAINMENT	(Z)	(i.e., 0POP05-EO-FO05)
INVENTORY	(I)	(i.e., 0POP05-EO-FO06)

6.4 CSF Status Trees color/symbol coding priorities are listed below:

COLOR	LINE CODE	SYMBOL CODE	STATUS/RESPONSE
RED	▬	●	The critical safety function is under extreme challenge: immediate operator action is required.
ORANGE	■ ■ ■	◕	The critical safety function is under severe challenge: prompt operator action is required.
YELLOW	● ● ●	◔	The critical safety function condition is off-normal. Operator action may be taken.
GREEN	▭	○	The critical safety function is satisfied. No operator action is needed.

NOTE

When implementing Addendum 5 “Verification of ESF Equipment Operation” of 0POP05-EO-EO00, then CSF Status Trees are monitored and NOT implemented even though a transition to an FRP is determined. The transition is delayed in most cases (refer to Step 6.1 for exceptions) until the verification of ESF equipment (per Addendum 5) is complete since this may resolve the CSF Status Tree abnormal indication.

6.5 WHEN monitoring CSF Status Trees, THEN:

6.5.1 Always perform the evaluations in the priority sequence listed in Section 6.3.

6.5.2 Enter at the box marked “ENTER” located at the left side of the status tree.

6.5.3 Answer the questions based on plant conditions at the time and follow the appropriate branch line to the next question.

6.5.4 WHEN a color-coded terminus is reached, THEN the individual status tree evaluation is complete.

6.6 IF a RED condition is reached, THEN immediately stop any ORP or yellow path FRP actions in progress (i.e., DO **NOT** complete the step in progress) AND perform the FRP required by the RED condition.

6.7 IF during the performance of a RED condition FRP, a RED condition of higher priority as listed in Section 6.3 arises, THEN the higher priority condition should be addressed first AND the lower priority condition FRP suspended (i.e., complete the step in progress).

6.8 IF an ORANGE condition arises, THEN monitor all of the remaining status trees. IF no RED condition exists, THEN suspend any ORP in progress (i.e., complete the step in progress) and perform the FRP required by the ORANGE condition.

6.9 IF during the performance of an ORANGE condition FRP, a RED condition OR higher priority ORANGE condition arises, THEN the RED or higher priority ORANGE condition is to be addressed first, and the original ORANGE condition FRP suspended (i.e., complete the step in progress). IF a FRP specifically states that a higher priority condition should **NOT** be addressed, THEN this requirement does **NOT** apply.

6.10 IF a YELLOW condition arises, THEN the YELLOW condition FRP implementation is based on operator judgement.

- In most cases the YELLOW condition FRP is indicative of an off-normal OR temporary condition which should be restored by actions already in progress.
- In other cases, a YELLOW condition may provide early indication of a potential RED or ORANGE condition.
- IF a YELLOW condition FRP is implemented, AND a higher priority condition is reached, THEN the YELLOW condition FRP should be treated as an ORP for transition to the higher priority FRP.
- YELLOW condition FRPs may be selectively entered by the US/SM based on judgement and the priorities listed in Section 6.3 are **NOT** applicable for YELLOW condition FRP selection.

6.11 IF identical color priorities for RED or ORANGE conditions are reached on different status trees, THEN the required action priority is determined by the order of priority listed in Section 6.3.

6.12 WHEN a RED or ORANGE priority FRP is entered, THEN perform the FRP to its completion, unless preempted by a higher priority condition.

6.13 IF QDPS OR Plant Computer is available, THEN it is acceptable to use QDPS and Plant Computer to monitor the CSF Status Trees.

7.0 EOP Network Usage

7.1 Entry into the EOPs is limited to the following conditions:

7.1.1 WHEN the Reactor is in Mode 1, 2, OR 3 with RCS pressure GREATER THAN 1000 PSIG AND any reactor trip or safety injection occurs OR is required (this includes a manual reactor trip and / or safety injection in response to approaching a reactor trip / safety injection setpoint such that an automatic action is imminent), THEN 0POP05-EO-EO00, Reactor Trip Or Safety Injection, SHALL be entered, unless the Control Room has been evacuated OR a complete loss of all AC ESF busses has occurred.

7.1.2 WHEN the Reactor is in Mode 1, 2, 3, OR 4 AND a complete loss of power on all AC ESF busses occurs, THEN 0POP05-EO-EC00, Loss Of All AC Power, SHALL be entered, unless the Control Room has been evacuated. This entry condition also applies during the performance of ANY other EOP.

7.1.3 IF the Control Room has been evacuated, THEN 0POP04-ZO-0001, Control Room Evacuation, SHALL take precedence over all EOPs AND 0POP04-ZO-0008 and 0POP04-ZO-0009.

7.1.4 IF a fire occurs in Fire Areas 02 – 78, THEN 0POP04-ZO-0009, Safe Shutdown Fire Response, SHALL take precedence over all EOPs.

- IF a reactor trip occurs or is directed during the performance of 0POP04-ZO-0009, THEN the actions of 0POP04-ZO-0009 SHALL be performed as directed by the CIP page or procedure step.

- Operator actions of this procedure section 4.0 that DO **NOT** conflict with the actions of 0POP04-ZO-0009 may be taken during 0POP04-ZO-0009 following a reactor trip to limit the effects of the reactor trip on the plant. The actions to establish and maintain a heat sink, limit RCS cooldown and establish RCS pressure and inventory control are examples of (but not limited to) prudent actions that should be taken.

- Actions may be taken per EOP’s, Off Normal Operating Procedures and Annunciator Response Procedures that DO **NOT** conflict with the actions of 0POP04-ZO-0009 if adequate resources are available. The EOP, Off Normal Operating Procedure or Annunciator Response Procedure should be entered and procedure steps followed. (e.g., IF during the performance of the 0POP04-ZO-0008/9 there are indications of abnormal RCP conditions, THEN the RCP Off Normal Operating Procedure SHOULD be entered.)

04/05/2017
DATE EFFECTIVE

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION

0POP05-EO-FO01 Rev. 2

SUBCRITICALITY CRITICAL SAFETY FUNCTION STATUS TREE

DEPARTMENT PROCEDURE

SAFETY RELATED (Q)

USAGE CONTROL: In Hand Controlling Station

LIST OF ATTACHMENTS:

None

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

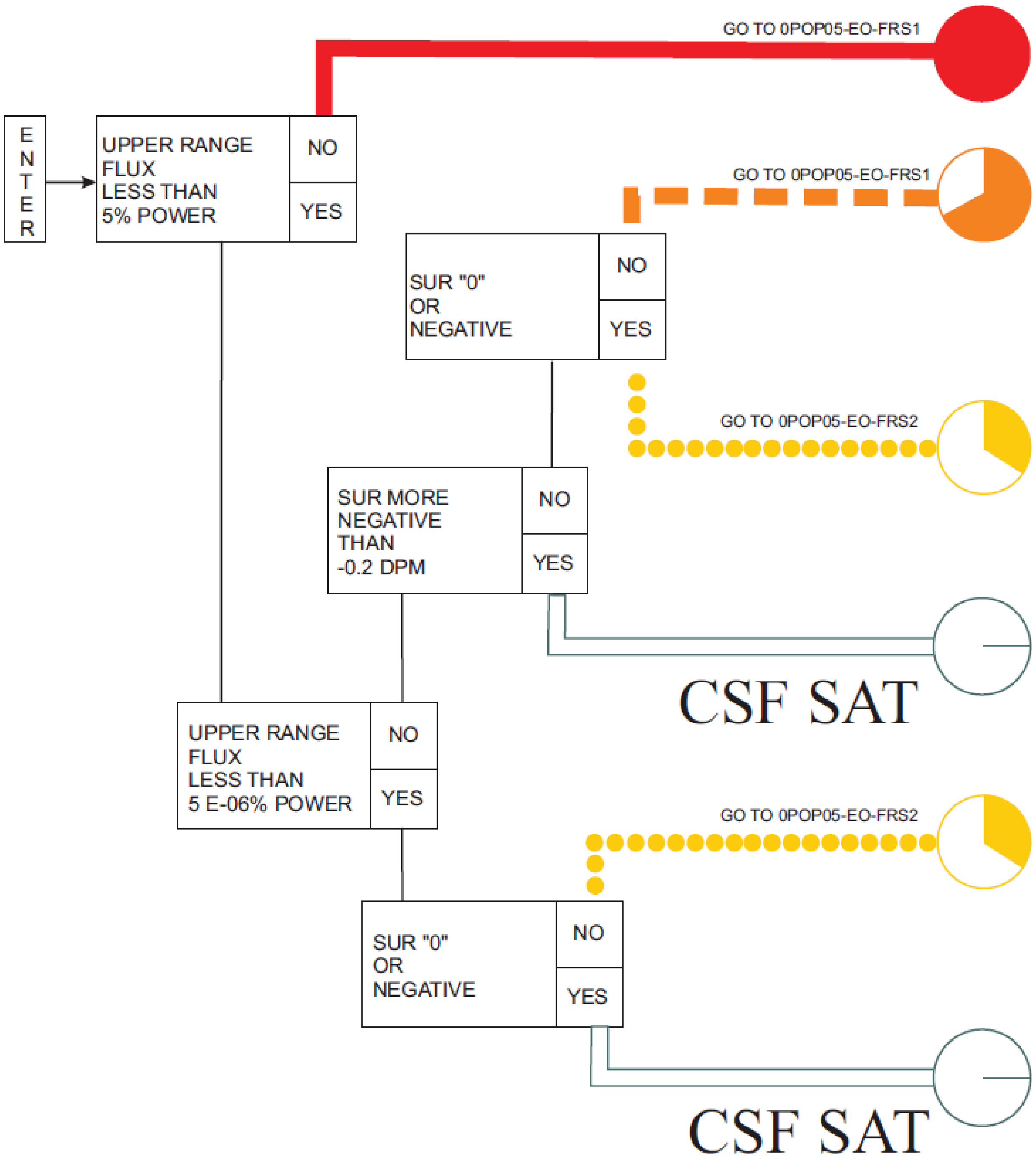
STI# 34471910

0POP05-EO-F001

SUBCRITICALITY CRITICAL SAFETY FUNCTION STATUS TREE

REV. 2

PAGE 1 OF 1



CFL0722(06/27/16)

04/05/2017
DATE EFFECTIVE

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION

0POP05-EO-FO04 Rev. 5

INTEGRITY CRITICAL SAFETY FUNCTION STATUS TREE

DEPARTMENT PROCEDURE

SAFETY RELATED (Q)

USAGE CONTROL: In Hand Controlling Station

LIST OF ATTACHMENTS:

- Addendum 1, Integrity Operational Limits
- Addendum 2, Cold Overpressure Limits

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

STI# 34472283

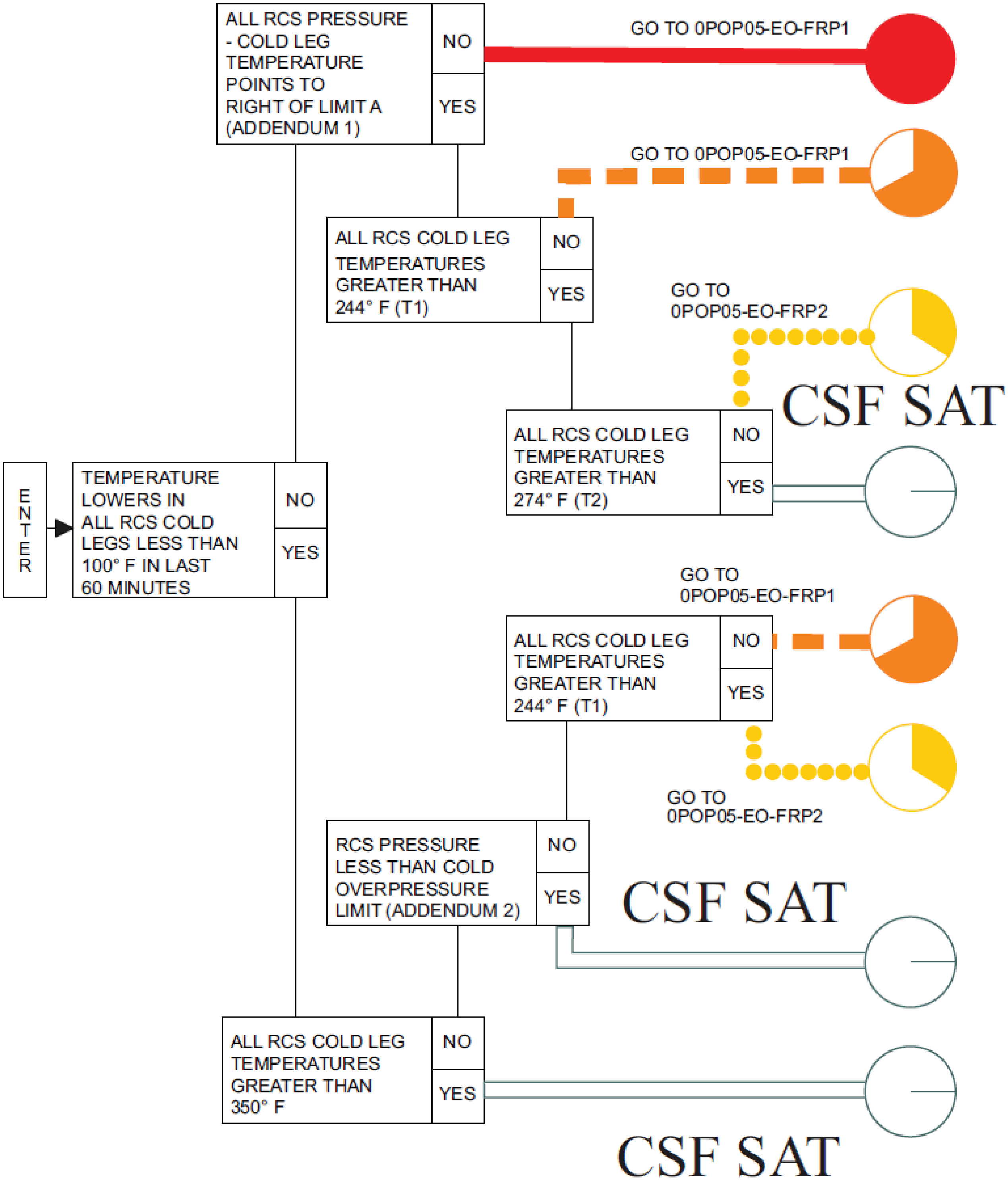
0POP05-EO-FO04 INTEGRITY CRITICAL SAFETY FUNCTION STATUS TREE REV. 5

PAGE 1 OF 2

ADVERSE CONTAINMENT CONDITIONS

IF any of the following conditions are met, THEN USE adverse containment values:

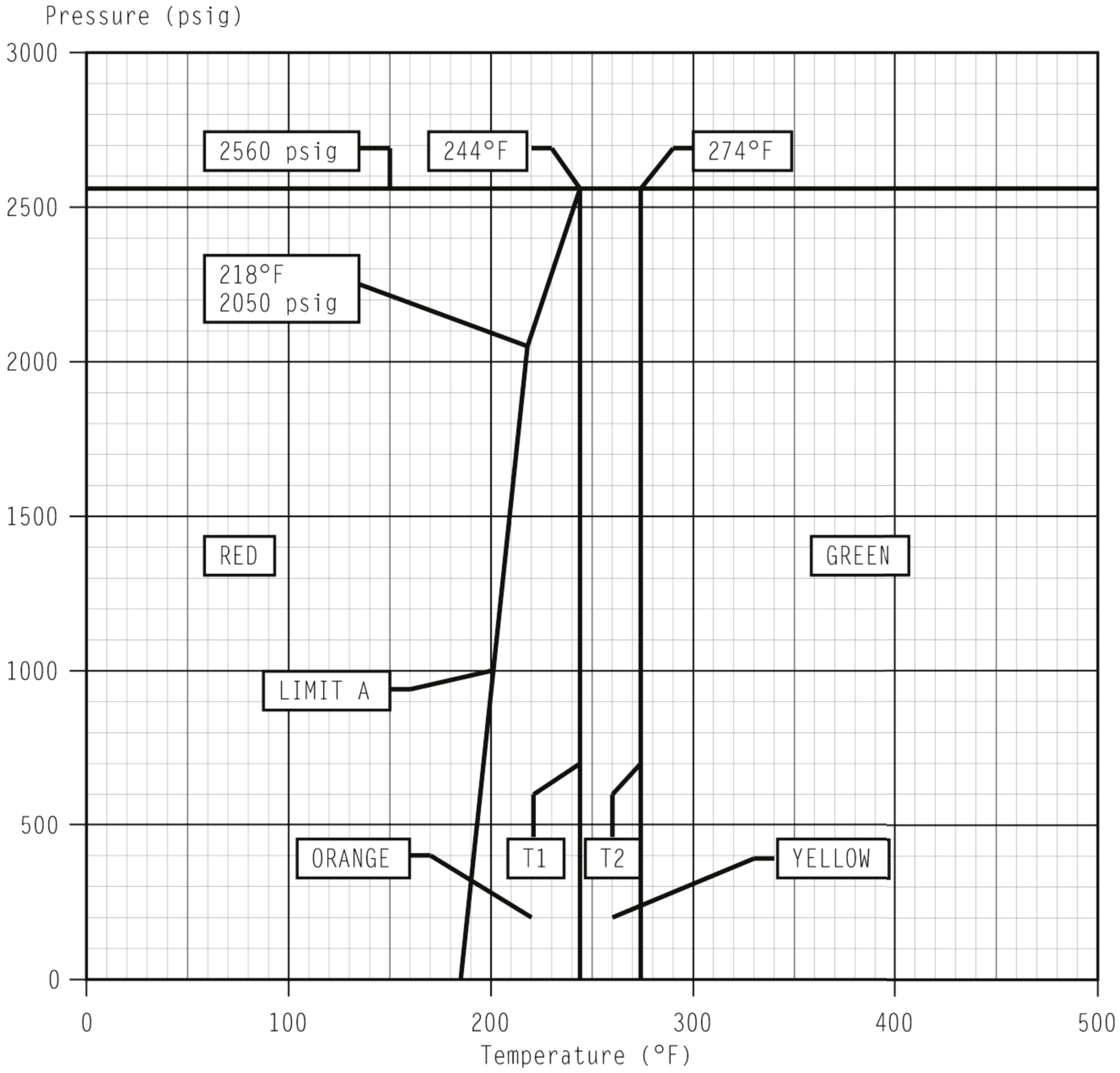
- Containment pressure GREATER THAN OR EQUAL TO 5 PSIG.
- Containment radiation levels GREATER THAN OR EQUAL TO 10^5 R/HR.
- Containment integrated radiation dose GREATER THAN OR EQUAL TO 10^6 RADS.



CFL00164(07/12/16)

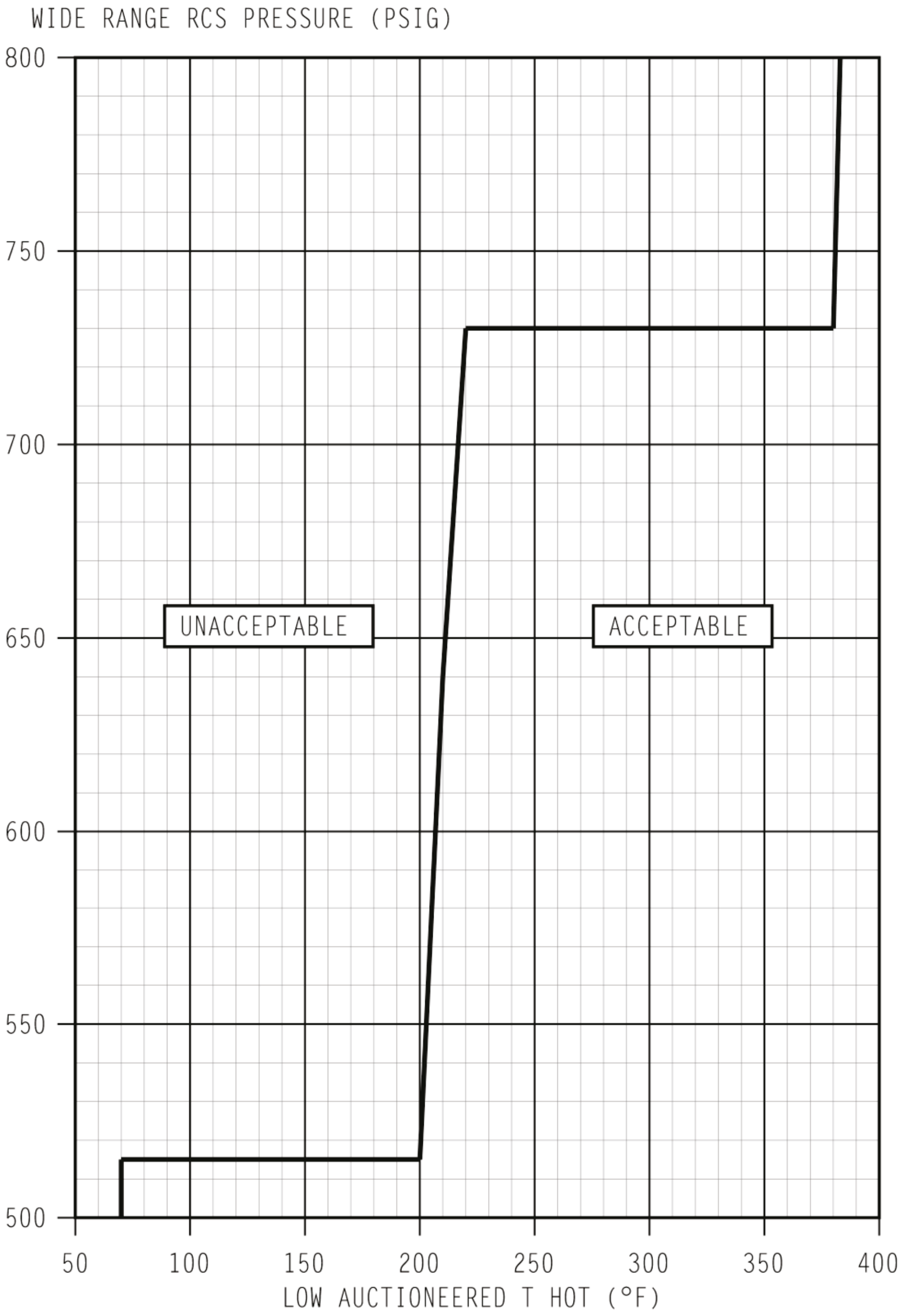
ADDENDUM 1
INTEGRITY OPERATIONAL LIMITS

Integrity Operational Limits



ADDENDUM 2
COLD OVERPRESSURE LIMITS

COLD OVERPRESSURE LIMITS



Exam Bank No.: 3189 **Last used on an NRC exam:** Never

SRO Sequence Number: 100

Given the following:

- An ALERT has been declared at the site.
- The TSC and EOF have been activated.

In accordance with EP-0001.000, STPEGS Emergency Plan, the responsibility for ____________ will remain with the Shift Manager after turnover of responsibilities to the EOF Manager is complete.

A. event declaration

B. notifications to state and county authorities

C. protective action recommendations for the general public

D. approving departures from license conditions per 50.54(x)

Answer: D approving departures from license conditions per 50.54(x)

Exam Bank No.: 3189 **Source:** Modified **Modified From** 2668

K/A Catalog Number: G2.4.40 Knowledge of SRO responsibilities in emergency plan implementation.

SRO Importance: 4.5 **Tier:** 3 **Group/Category:** 4

10CFR Reference or SRO Objective: 55.43(b)(5)

SRO Justification:

Student must have knowledge of SRO responsibilities within the Emergency Response Plan.

STP Lesson: LOT 803.14 **Objective Number:** SRO-47030

Discuss the duties and responsibilities of the Shift Manager as delineated in 0ERP01-ZV-SH01.

Reference: EP-0001.000

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: INCORRECT. Plausible as event declaration is a non-delegable responsibility but is passed on to the EOF Manager as the Emergency Director.

B: INCORRECT. Plausible as notification of offsite authorities is a non-delegable responsibility but is passed on to the EOF Manager as the Emergency Director.

C: INCORRECT. Plausible as protective action recommendations is a non-delegable responsibility but is passed on to the EOF Manager as the Emergency Director.

D: CORRECT. When the Shift Manager is relieved as Emergency Director, the non-delegable responsibilities of classification, notification and PARs are passed to the EOF Manager. Approving departures from license conditions per 10CFR50.54(x) remains with the Shift Manager as the EOF Manager may or may not have an SRO license.

Question Level: F **Question Difficulty** 3

Justification:

The student must have knowledge of responsibilities of the Shift Manager as Emergency Director.

SOURCE FOR SRO #100

Exam Bank No.: 2668 Last used on an NRC exam: 2020

In accordance with 0ERP02-CK-OPS01, Shift Manager, the Emergency Director is allowed to delegate approving…

A. press releases prior to issuance.

B. notifications to State and County.

C. departures from license conditions.

D. Protective Action Recommendations.

Answer: A press releases prior to issuance.

Exam Bank No.: 2668 **Source:** Bank **Modified from:**

K/A Catalog Number: G2.4.40 Knowledge of SRO responsibliities in emergency plan implementation.

RO/SRO Importance: / 4.5 **Tier:** 3 **Group/Category:**

RO-10CFR55.41 # **SRO-10CFR55.43 #** 5 or **SRO Obj:**

SRO Justification:

Task is unique to the SRO position.

STP Lesson: LOT 803.14 **Objective Number:** SRO-470

DISCUSS the duties and responsibilities of the Shift Manager as delineated in 0ERP01-ZV-SH01, Shift Manager.

Reference: 0ERP01-ZV-SH01, Shift Manager, steps 4.4.1 and 4.4.2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Distractor Justification

A: CORRECT: Approving press releases prior to issuance is a Emergency Director responsibility that may be delegated.

B: INCORRECT: Plausible because all distractors are responsibilities of the Emergency Director listed in 0ERP01-ZV-SH01.

C: INCORRECT: Plausible because all distractors are responsibilities of the Emergency Director listed in 0ERP01-ZV-SH01.

D: INCORRECT: Plausible because all distractors are responsibilities of the Emergency Director listed in 0ERP01-ZV-SH01.

Level F **Difficulty** 3

Justification:

Student must know the responsibilities of the Emergency Director.

K. IT Coordinator

- Coordinate information and activities with offsite agency personnel in the facility
- Monitor facility equipment (computer, communications, etc.) for proper operation

5. Joint Information System (JIS) / Joint Information Center (JIC)

STPEGS maintains a program and process for Corporate Communications and key business unit staff to operate in the Joint Information Center or within a Joint Information System for any event that can impact the company. This organization provides media and public information and communications for the ERO during all declared events.

Refer to Section G for JIC/JIS details.

B.2	An individual is designated as the on-shift emergency coordinator (individual title may vary) who has the authority and responsibility to immediately and unilaterally initiate any emergency response measures, including approving protective action recommendations (PARs) to be disseminated to authorities responsible for implementing offsite emergency response measures.

The Emergency Director is the STPEGS individual who has overall command and control of the emergency.

The Shift Manager is the individual who is on-shift at all times and who has the authority and responsibility to immediately and unilaterally initiate any emergency actions, including providing protective action recommendations (PARs) to authorities responsible for implementing offsite emergency measures, and assumes the role of Emergency Director upon emergency declaration.

The Shift Manager is responsible for providing ERO command and control until relieved.

B.2.a	The functional responsibilities assigned to the ERO are established and the responsibilities that may not be delegated to other members of the ERO are clearly specified in the emergency plan.

The Shift Manager is responsible for performing the following non-delegable responsibilities until relieved:

- Event declaration
- Notification of offsite authorities
- PARs for the general public
- Emergency Exposure (Dose limits and KI)

When the Shift Manager is relieved of overall command and control of emergency response, the non-delegable responsibilities of classification, notification and PARs and the role of Emergency Director are passed to the EOF Manager. Approving departures from license conditions per 10CFR50.54(x) for Control Room Operator actions AND equipment manipulations remains with the Shift Manager throughout the emergency event.

LOT 26 NRC

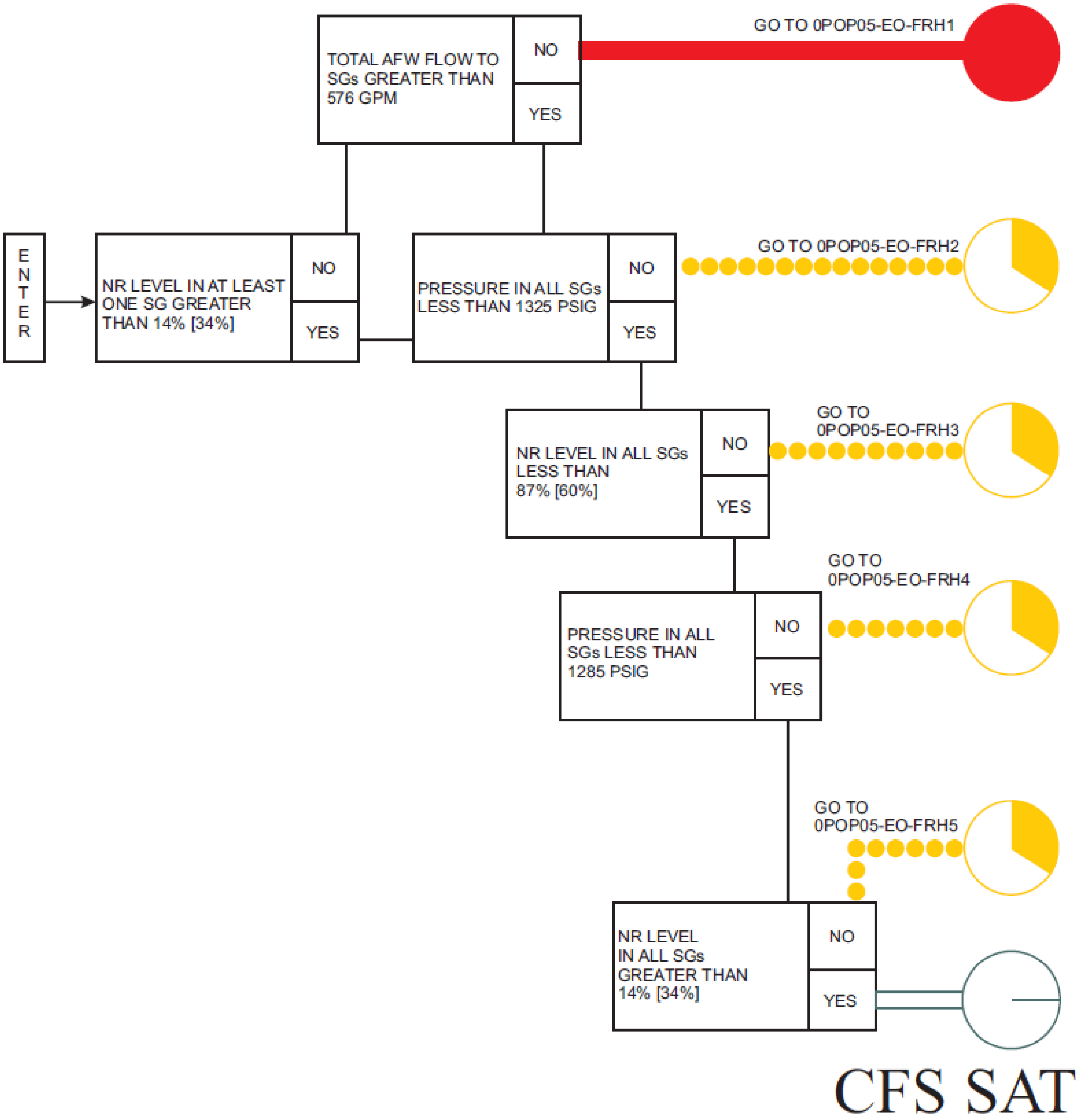
RO EXAMINATION
STUDENT REFERENCE
PACKAGE

Q25

0POP05-EO-F003 HEAT SINK CRITICAL SAFETY FUNCTION STATUS TREE

REV. 7

PAGE 2 OF 2



CFL0723(10/29/07)

Q70

	0PSP10-ZG-0003	Rev. 14	Page 16 of 20
	Shutdown Margin Verification - Modes 3, 4 and 5		
Form 2	Short Form SDM Limit Curve Method		Page 1 of 1

Unit 1 Cycle 24

5.5.1 Plant is in Mode 3, 4 or 5: ANV
Initial

5.5.2 C_B Rod = (+) 0 ppm
5.5.3

5.5.4 C_B Effective = 1410 (C_B RCS) - 0 (C_B Rod)

C_B Effective = 1410 ppm

5.5.5.1 C_B SDM Limit = ____________ ppm

5.5.5.2 RCS Average Temperature ___________ °F

Cycle Burnup _____________ MWD/MTU

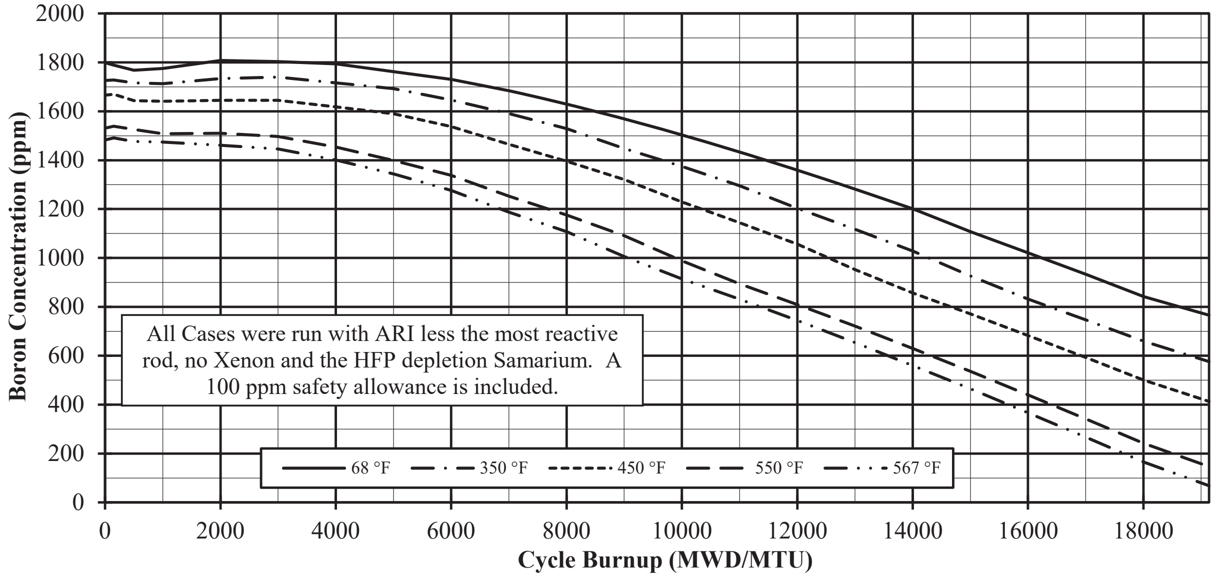
5.8 Is C_B Effective $\geq$ C_B SDM Limit? ________ Yes ________ No

Completed By: ____ Date: ____________________

Verified By: ____ Date: ____________________

This Form, when complete, SHALL be retained for the life of the plant.

Figure 5.5
SHUTDOWN MARGIN LIMIT CURVE
Unit 1 Cycle 24 (Modes 3, 4, and 5)
(Source: A41009--00656UB, Rev 0)



Burnup (MWD/MTU)	Boron Concentration (ppm)										
	68 °F	140 °F	200 °F	250 °F	300 °F	350 °F	400 °F	450 °F	500 °F	550 °F	567 °F
0	1799	1799	1799	1775	1751	1726	1696	1666	1599	1531	1483
150	1789	1789	1789	1769	1748	1728	1699	1670	1604	1539	1491
500	1768	1766	1766	1749	1733	1716	1680	1644	1585	1526	1477
1000	1775	1775	1775	1754	1734	1713	1677	1641	1575	1508	1474
2000	1808	1808	1808	1783	1759	1734	1689	1645	1577	1510	1461
3000	1803	1803	1803	1782	1761	1740	1692	1645	1571	1497	1446
4000	1794	1794	1794	1768	1742	1716	1667	1618	1536	1454	1401
5000	1762	1762	1762	1739	1716	1693	1641	1590	1495	1399	1344
6000	1731	1731	1731	1703	1674	1646	1592	1538	1438	1338	1277
7000	1684	1681	1681	1651	1621	1591	1528	1465	1359	1253	1187
8000	1629	1624	1624	1592	1560	1529	1462	1396	1286	1175	1107
9000	1569	1561	1561	1523	1486	1448	1385	1321	1206	1091	1006
10000	1504	1482	1482	1446	1411	1375	1302	1229	1108	988	914
11000	1434	1409	1409	1372	1334	1296	1220	1144	1019	894	831
12000	1359	1334	1333	1289	1246	1203	1129	1056	932	809	744
13000	1282	1254	1252	1207	1162	1117	1034	952	837	721	654
14000	1201	1172	1158	1115	1071	1028	942	857	743	630	560
15000	1107	1077	1072	1023	974	926	848	771	654	536	465
16000	1021	990	984	933	883	832	758	683	562	440	367
17000	933	901	884	838	792	747	670	593	468	342	267
18007	842	814	791	747	703	659	580	500	371	242	165
19139	766	737	713	667	622	576	494	413	280	147	68

Hot Rod Test C_B, ARO, 400°F to 567°F, K <= 0.98: **2257 ppm**	Mode 5 C_B, ARO, 68°F to 200°F, K <= 0.95: **2604 ppm**	Refueling C_B, ARO, 68°F to 140°F, K <= 0.95: **2582 ppm**

Prepared By: ______________________ Date: 10/24/21

Reviewed By: ______________________ Date: 10/24/21

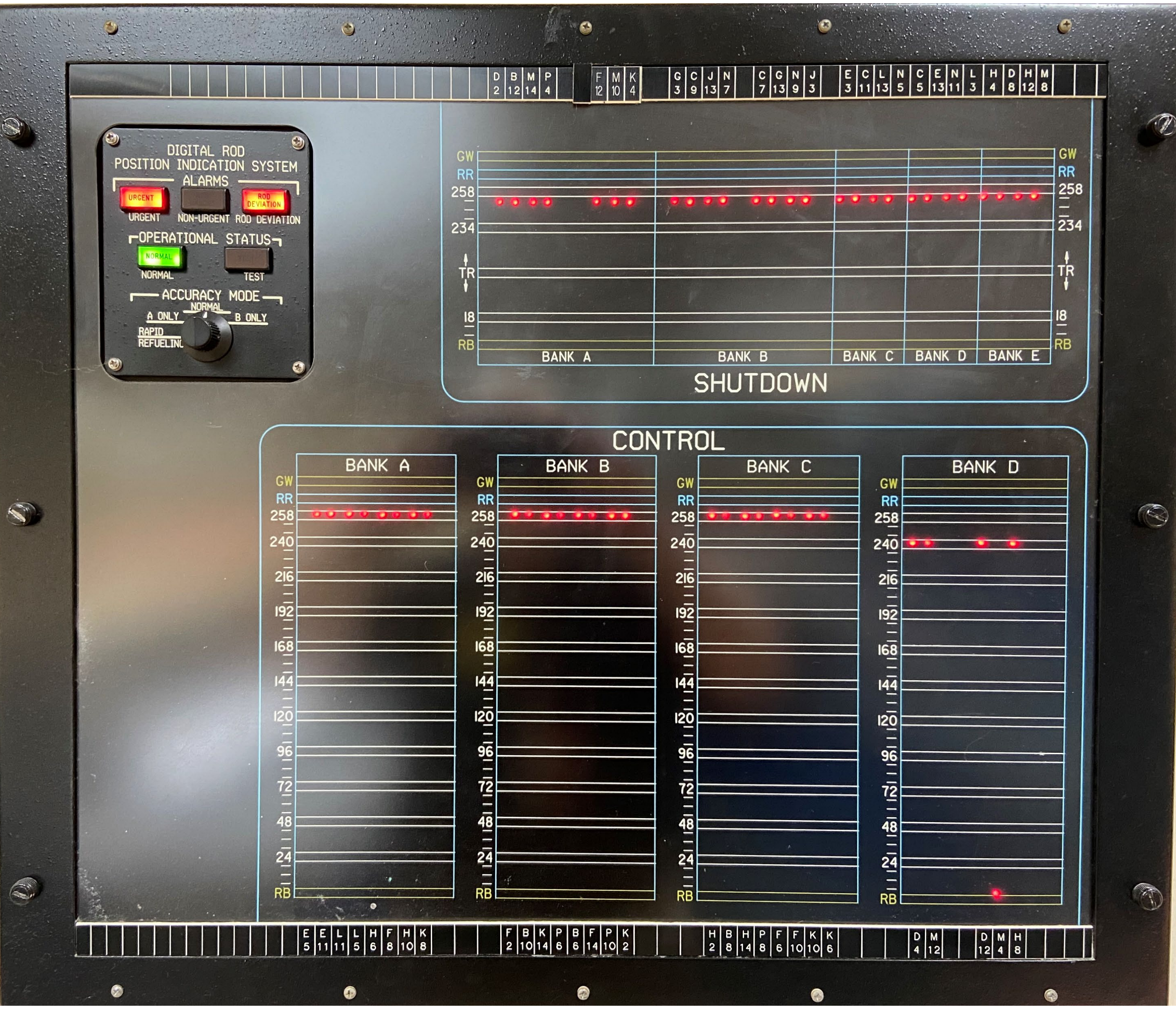
Approved By: ______________________ Date: 10/25/21

Reactor & Fuel Engineering Supervisor

LOT 26 NRC
SRO EXAMINATION
STUDENT REFERENCE
PACKAGE

Q82

Clarification given that Rod M4 indicates < 6 steps.



Q92

EAL Wall Charts {0EPR04-UA-0001, R0}
Page 1 of 4

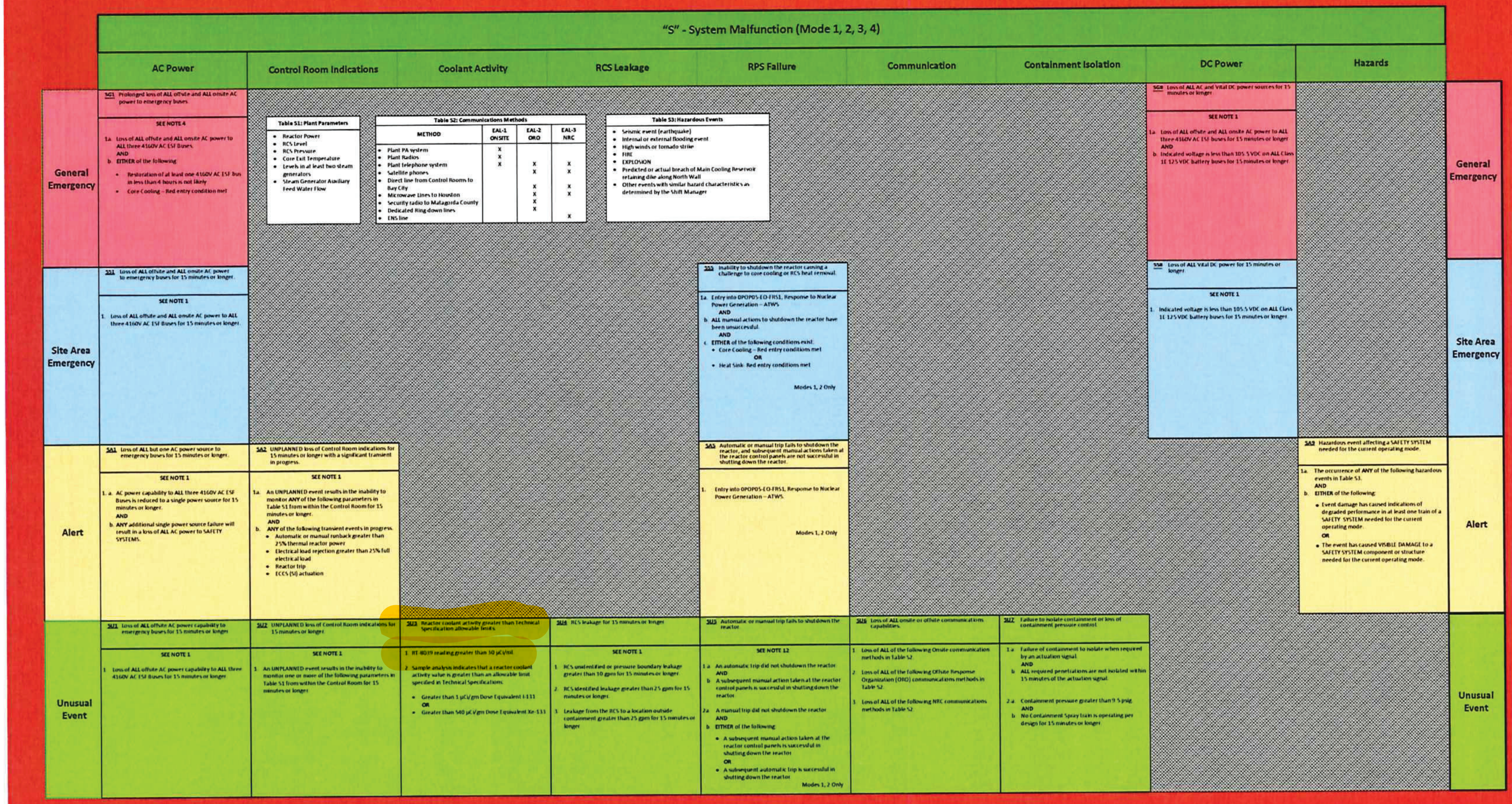
F – FISSION PRODUCT BARRIER (MODES 1,2,3,4)

GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT
FG1 – LOSS of **ANY** two barriers and the LOSS or POTENTIAL LOSS of the third barrier. Clad RCS Cont	FS1 – LOSS or POTENTIAL LOSS of **ANY** two barriers. Clad RCS Cont	FA1 – **ANY** LOSS or **ANY** POTENTIAL LOSS of **EITHER** the the Fuel Clad or RCS BARRIER. Clad RCS

	Fuel Clad		RCS		Containment	
	Loss	Potential Loss	Loss	Potential Loss	Loss	Potential Loss
1.RCS or SG Tube Leakage		A. Core Cooling – Orange entry conditions met.	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube leakage	A. Operation of a standby charging pump is required by **EITHER** of the following: • UNISOLABLE RCS leakage **OR** • SG tube leakage. **OR** B. Integrity – Red entry conditions met.	A. A leaking or RUPTURED SG is FAULTED outside of containment.	
2. Inadequate Heat Removal	A. Core Cooling – Red entry conditions met.	A. Core Cooling – Orange entry conditions met. **OR** B. Heat Sink – Red entry conditions met.		A. Heat Sink – Red entry conditions met.		A. Core Cooling – Red entry conditions met for 15 minutes or longer.
3. RCS Activity / Containment Radiation	A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 2100 R/hr. **OR** A2. HATCH MONITOR greater than 4200 mR/hr. **OR** B. Sample analysis indicates that reactor coolant activity is greater than 300 µCi/gm dose equivalent I-131.		A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 10 R/hr. **OR** A2. HATCH MONITOR greater than 20 mR/hr.			A1. RCB Rad Monitor RT-8050 or RT-8051 greater than 45,000 R/hr. **OR** A2. HATCH MONITOR greater than 90,000 mR/hr.
4. Containment Integrity or Bypass					A. Containment isolation is required AND **EITHER** of the following: • Containment integrity has been lost based on Emergency Director judgment. **OR** • UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment – Red entry conditions met. **OR** B. Explosive mixture exists inside containment ($H_2 \geq 4\%$). **OR** C1. Containment pressure greater than 9.5 psig. **AND** C2. Less than one full train of Containment Spray is operating per design for 15 minutes or longer.
5. Other Indications						
6. Emergency Director Judgment	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	**ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Q92

EAL Wall Charts (0EPR04-UA-0001, R0)
Page 2 of 4



"S" - System Malfunction (Mode 1, 2, 3, 4)

	AC Power	Control Room Indications	Coolant Activity	RCS Leakage	RPS Failure	Communication	Containment Isolation	DC Power	Hazards
General Emergency	**SG1** Prolonged loss of **ALL** offsite and **ALL** onsite AC power to emergency buses SEE NOTE 4 1a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses. **AND** b. **EITHER** of the following • Restoration of at least one 4160V AC ESF bus in less than 4 hours is not likely • Core Cooling – Red entry condition met							**SG8** Loss of **ALL** AC and Vital DC power sources for 15 minutes or longer SEE NOTE 1 1a. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF buses for 15 minutes or longer **AND** b. Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer	
Site Area Emergency	**SS1** Loss of **ALL** offsite and **ALL** onsite AC power to emergency buses for 15 minutes or longer. SEE NOTE 1 1. Loss of **ALL** offsite and **ALL** onsite AC power to **ALL** three 4160V AC ESF Buses for 15 minutes or longer.				**SS5** Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal. 1a. Entry into 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS **AND** b. **ALL** manual actions to shutdown the reactor have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • Core Cooling – Red entry conditions met **OR** • Heat Sink Red entry conditions met Modes 1, 2 Only			**SS8** Loss of **ALL** Vital DC power for 15 minutes or longer SEE NOTE 1 1. Indicated voltage is less than 105.5 VDC on **ALL** Class 1E 125 VDC battery buses for 15 minutes or longer	
Alert	**SA1** Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer. SEE NOTE 1 1. a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer. **AND** b. **ANY** additional single power source failure will result in a loss of **ALL** AC power to SAFETY SYSTEMS.	**SA2** UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress. SEE NOTE 1 1a. An UNPLANNED event results in the inability to monitor **ANY** of the following parameters in Table S1 from within the Control Room for 15 minutes or longer. **AND** b. **ANY** of the following transient events in progress. • Automatic or manual runback greater than 25% thermal reactor power • Electrical load rejection greater than 25% full electrical load • Reactor trip • ECCS (SI) actuation			**SA5** Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken at the reactor control panels are not successful in shutting down the reactor. 1. Entry into 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS. Modes 1, 2 Only				**SA9** Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode. 1a. The occurrence of **ANY** of the following hazardous events in Table S3. **AND** b. **EITHER** of the following: • Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode. **OR** • The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.
Unusual Event	**SU1** Loss of **ALL** offsite AC power capability to emergency buses for 15 minutes or longer SEE NOTE 1 1. Loss of **ALL** offsite AC power capability to **ALL** three 4160V AC ESF Buses for 15 minutes or longer	**SU2** UNPLANNED loss of Control Room indications for 15 minutes or longer. SEE NOTE 1 1. An UNPLANNED event results in the inability to monitor one or more of the following parameters in Table S1 from within the Control Room for 15 minutes or longer	**SU3** Reactor coolant activity greater than Technical Specification allowable limits. 1. RT-8039 reading greater than 30 µCi/ml 2. Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specifications. • Greater than 1 µCi/gm Dose Equivalent I-131 **OR** • Greater than 540 µCi/gm Dose Equivalent Xe-133	**SU4** RCS leakage for 15 minutes or longer SEE NOTE 1 1. RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer. 2. RCS identified leakage greater than 25 gpm for 15 minutes or longer. 3. Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer	**SU5** Automatic or manual trip fails to shutdown the reactor SEE NOTE 12 1 a. An automatic trip did not shutdown the reactor **AND** b. A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor 2a. A manual trip did not shutdown the reactor **AND** b. **EITHER** of the following • A subsequent manual action taken at the reactor control panels is successful in shutting down the reactor **OR** • A subsequent automatic trip is successful in shutting down the reactor Modes 1, 2 Only	**SU6** Loss of **ALL** onsite or offsite communications capabilities. 1. Loss of **ALL** of the following Onsite communication methods in Table S2 2. Loss of **ALL** of the following Offsite Response Organization (ORO) communications methods in Table S2 3. Loss of **ALL** of the following NRC communications methods in Table S2	**SU7** Failure to isolate containment or loss of containment pressure control. 1.a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not isolated within 15 minutes of the actuation signal. 2.a. Containment pressure greater than 9.5 psig. **AND** b. No Containment Spray train is operating per design for 15 minutes or longer		

Table S1: Plant Parameters

- Reactor Power
- RCS Level
- RCS Pressure
- Core Exit Temperature
- Levels in at least two steam generators
- Steam Generator Auxiliary Feed Water Flow

Table S2: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
• Dedicated Ring-down lines		X	
• ENS line			X

Table S3: Hazardous Events

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall
- Other events with similar hazard characteristics as determined by the Shift Manager

NOTES

1	CU1, CU2, CU3, CU4, CA1, CA2, HS6, SU1, SU2, SU4, SA1, SA2, SS1, SS8, SG8	The Emergency Director should declare the event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
2	CS1, CG1	The Emergency Director should declare the event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.
3	RU1, RG2	The Emergency Director should declare the event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
4	SG1	The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.
5	RA1, RS1, RG1, RG2, HU4, CA3	The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
6	RA1, RS1, RG1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
7	RU1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
8	RU1, RA1, RS1, RG1	If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer VALID for classification purposes.
9	RA1, RS1, RG1	The pre-calculated effluent monitor values presented in EAL#1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
10	RA3, HA5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
11	HU3	EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
12	SU5	A manual action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving control rods or implementation of boron injection strategies.

Q92

H-HAZARDS (ALL MODES)

Security

General Emergency

HG1 HOSTILE ACTION resulting in loss of physical control of the FACILITY.

1a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANY of the following personnel in Table H1
AND
b. EITHER of the following has occurred
- ANY of the following safety functions in Table H2 cannot be controlled or maintained in MODES 1, 2 or 3 ONLY.
  OR
- Damage to spent fuel has occurred or is IMMINENT

Site Area Emergency

HS1 HOSTILE ACTION within the PROTECTED AREA.

1. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by ANY of the following personnel in Table H1.

Alert

HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

1. A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by ANY of the following personnel in Table H1.
2. A validated notification from NRC of an aircraft attack threat within 30 minutes of the site.

Unusual Event

HU1 Confirmed SECURITY CONDITION or threat.

1. A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by ANY of the following personnel in Table H1.
2. Notification of a CREDIBLE SECURITY THREAT directed at the site.
3. A validated notification from the NRC providing information of an aircraft threat.

Table H1: Security Supervision

- Security Force Supervisor
- Acting Security Manager
- Security Manager

Table H2: Safety Functions

- Reactivity Control
- Core Cooling
- RCS Heat Removal

Table H3/R2: Plant Areas Requiring Access

MODE 4	RCB	RHR Heat Exchanger Rooms
	MAB	51 ft Room 335
	EAB	Roof, MCC G8, 4.16KV Switchgear Rooms
MODE 5	EAB	4.16 KV Switchgear Rooms

Table H4: Plant Rooms/Areas

- Mechanical/Electrical Auxiliary Building (MEAB)
- Fuel Handling Building (FHB)
- Reactor Containment Building (RCB)
- Essential Cooing Water Intake Structure (ECWIS)
- Isolation Valve Cubicle (IVC)
- Diesel Generator Building (DGB)

Seismic

Unusual Event

HU2 Seismic event greater than OBE levels.

1.a. EITHER of the following conditions exist.
- "SEISMIC EVENT" alarm in Unit 1 Control Room?
  OR
- Control Room personnel feel an actual or potential seismic event.

AND
b. The occurrence of a seismic event is confirmed in a manner deemed appropriate by the Shift Manager or Emergency Director.

Hazards

Unusual Event

HU3 Hazardous event

SEE NOTE 11

1. A tornado strike within the PROTECTED AREA.
2. Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.
3. Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).
4. A hazardous event that results in onsite conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.
5. Predicted or actual breach of Main Cooling Reservoir retaining dike along North Wall.

Fire

Unusual Event

HU4 FIRE potentially degrading the level of safety of the plant.

SEE NOTE 5

1.a. A FIRE is NOT extinguished within 15-minutes of ANY of the following FIRE detection indications
- Report from the field (i.e., visual observation)
  OR
- Receipt of multiple (more than 1) fire alarms or indications
  OR
- Field verification of a single fire alarm

AND
b. The FIRE is located within ANY of the plant rooms or areas in Table H4.

2.a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).
AND
b. The FIRE is located within ANY of the plant rooms or areas in Table H4.
AND
c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

3. A FIRE within the ISFSI OR plant PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

4. A FIRE within the ISFSI OR plant PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Toxic Gas

Alert

HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.

SEE NOTE 10

1.a. Release of a toxic, corrosive, asphyxiant or flammable gas into the Control Room or ANY of the plant rooms or areas listed in Table H3/R2:
AND
b. Entry into the room or area is prohibited or impeded.

Control Room Evacuation

Site Area Emergency

HS6 Inability to control a key safety function from outside the Control Room.

SEE NOTE 1

1.a. An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP).
AND
b. Control of ANY of the following key safety functions in Table H2 is not reestablished within 15 minutes in Modes 1, 2, or 3 ONLY.

Alert

HA6 Control Room evacuation resulting in transfer of plant control to alternate locations.

1. An event has resulted in plant control being transferred from the Control Room to the Auxiliary Shutdown Panel (ASP).

Judgement

General Emergency

HG7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a GENERAL EMERGENCY.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the FACILITY. Releases can be reasonably expected to exceed EPA PROTECTIVE ACTION GUIDELINE exposure levels offsite for more than the immediate site area.

Site Area Emergency

HS7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a SITE AREA EMERGENCY.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA PROTECTIVE ACTION GUIDELINE exposure levels beyond the SITE BOUNDARY.

Alert

HA7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an ALERT.

Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a SECURITY EVENT that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA PROTECTIVE ACTION GUIDELINE exposure levels.

Unusual Event

HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a UE.

Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to FACILITY protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

R-RADIATION (ALL MODES)

Effluent Release

General Emergency

RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE.

SEE NOTES 5, 6, 8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "GE" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY.
   - Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
     OR
   - Analyses of field survey samples indicate THYROID CDE greater than 5,000 mrem for one hour of inhalation.

Site Area Emergency

RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem THYROID CDE.

SEE NOTES 5, 6, 8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "SAE" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY.
   Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
   OR
   Analyses of field survey samples indicate THYROID CDE greater than 500 mrem for one hour of inhalation.

Alert

RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem THYROID CDE.

SEE NOTES 5, 6 ,8 & 9

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "ALERT" for 15 minutes or longer.
2. Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY.
3. Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem THYROID CDE at or beyond the SITE BOUNDARY for one hour of exposure.
4. Field survey results indicate EITHER of the following at or beyond the SITE BOUNDARY.
   - Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
     OR
   - Analyses of field survey samples indicate THYROID CDE greater than 50 mrem for one hour of inhalation.

Unusual Event

RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

SEE NOTES 3, 7 & 8

1. Reading on ANY of the following radiation monitors greater than the values listed in Table R1 column "UE" for 60 minutes or longer.
2. Reading on gaseous effluent radiation monitor RT-8010B greater than 2 times the alarm setpoint established by a current radioactivity discharge permit for 60 minutes or longer.
3. Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.

Spent Fuel Pool

General Emergency

RG2 Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

SEE NOTE 5

1. Spent fuel pool level cannot be restored to at least 40'-4" for 60 minutes or longer.

Site Area Emergency

RS2 Spent fuel pool level at 40'-4" or lower.

1. Lowering of spent fuel pool level to 40'-4" or lower.

Alert

RA2 Significant lowering of water level above, or damage to, irradiated fuel.

1. Uncovery of irradiated fuel in the REFUELING PATHWAY.

2.a Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following FHB radiation monitor readings:
- FHB Exhaust, RT-8035 or RT-8036 greater than 1.00 E-1 µCi/ml
  OR
- ARM (68' FHB), RE-8090 greater than 1,500 mR/hr

OR
b. Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by ANY of the following RCB radiation monitors readings (Mode 5 or 6 only):
- ARMs (68' RCB), RE-8055 or RE-8099 greater than 850 mR/hr

3. Lowering of spent fuel pool level to 49'-10" or lower.

Unusual Event

RU2 UNPLANNED loss of water level above irradiated fuel.

1.a UNPLANNED water level drop in the REFUELING PATHWAY as indicated by ANY of the following
- Visual Observation
  OR
- Annunciator alarm on lampbox 22M02 Window F-5 "SFP WATER LVL HI/LO"
  OR
- Spent fuel in the ICSA AND Annunciator alarm on lampbox 22M02 Window F-6 "SFP Trouble" AND Plant Computer point FCLC1420 "REFLNG CAV LVL IN CNTMT" (ICSA Water Level HI/LO) is in alarm

AND
b. UNPLANNED rise in area radiation levels on ANY of the following area radiation monitors
- RE-8055 (68' RCB) Mode 5 or 6 only
  OR
- RE-8099 (68' RCB) Mode 5 or 6 only
  OR
- RE-8090 (68' FHB)

Abnormal Rad Levels

Table R1: Effluent Monitors

	Release Point	Monitor	GE	SAE	ALERT	UE
Gas	Unit Vent	RT-8010B	1.50 E+08 µCi/sec	1.50 E+07 µCi/sec	1.50 E+06 µCi/sec	1.40 E+05 µCi/sec
	Steam Lines	RT-8046 thru 8049	4.00 E+02 µCi/ml	4.00 E+01 µCi/ml	4.00 E+00 µCi/ml	5.00 E-02 µCi/ml

Alert

RA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

SEE NOTE 10

1. Dose rate greater than 15 mR/hr in ANY of the following areas:
   - Control Room ARM (RE-8066)
     OR
   - Central Alarm Station (CAS) by radiation survey
2. An UNPLANNED event results in radiation levels that prohibit or impede access to ANY of the areas listed in Table H3/R2

E-ISFSI (ALL MODES)

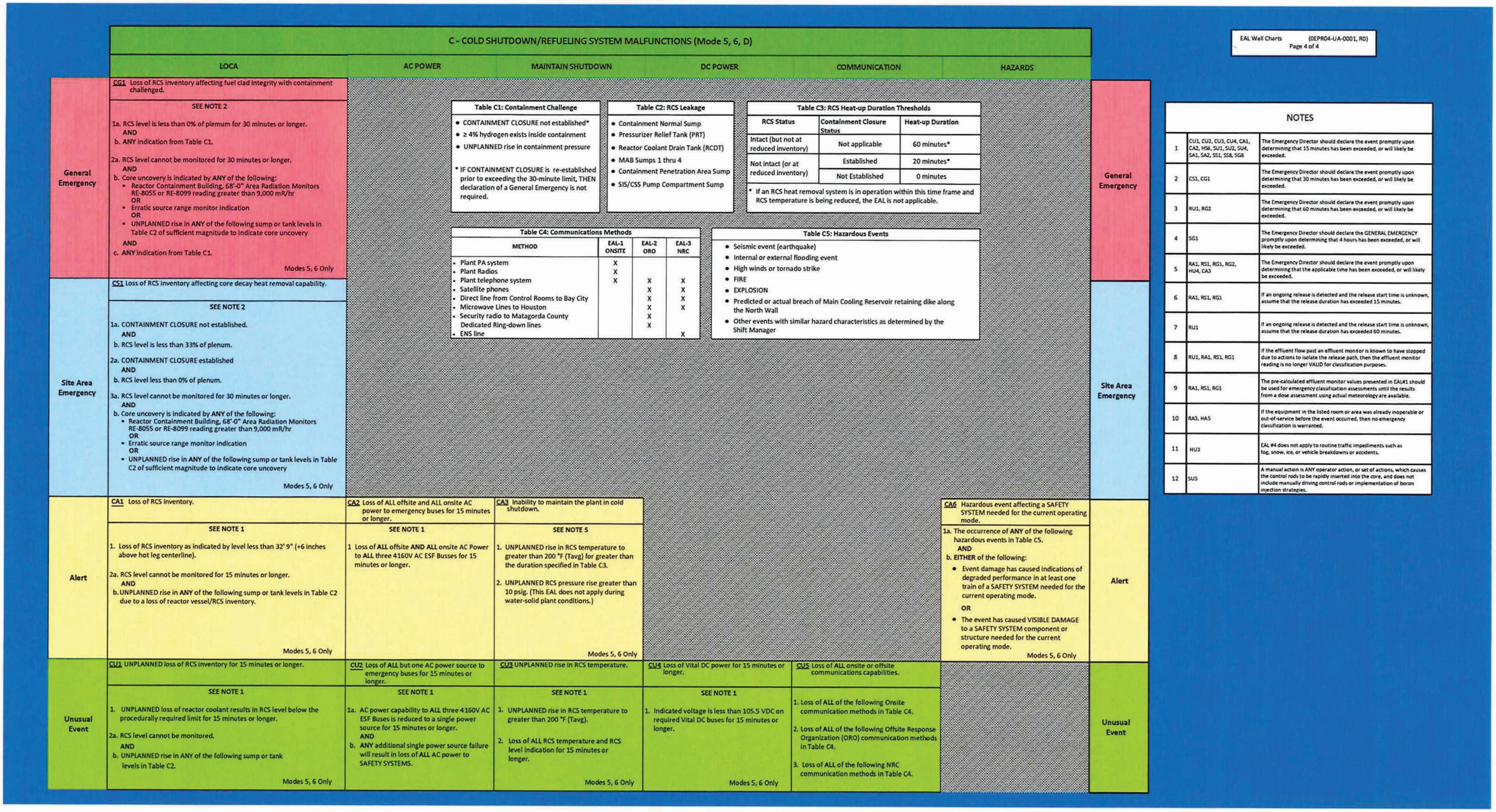
ISFSI

Unusual Event

E-HU1 Damage to a loaded cask CONFINEMENT BOUNDARY.

1. Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading greater than:
   - 60 mrem/hr (gamma + neutron) on the top surface of the spent fuel cask
     OR
   - 600 mrem/hr (gamma + neutron) on the side surface of the spent fuel cask
     OR
   - 7000 mrem/hr (gamma + neutron) on the side surface of the transfer cask.

Q92



EAL Wall Charts (0EPR04-UA-0001, R0)
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C - COLD SHUTDOWN/REFUELING SYSTEM MALFUNCTIONS (Mode 5, 6, D)

LOCA | AC POWER | MAINTAIN SHUTDOWN | DC POWER | COMMUNICATION | HAZARDS

General Emergency

CG1 Loss of RCS inventory affecting fuel clad integrity with containment challenged.

SEE NOTE 2

1a. RCS level is less than 0% of plemum for 30 minutes or longer.
AND
b. **ANY** indication from Table C1.

2a. RCS level cannot be monitored for 30 minutes or longer.
AND
b. Core uncovery is indicated by **ANY** of the following:
- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr
  OR
- Erratic source range monitor indication
  OR
- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

AND
c. **ANY** indication from Table C1.

Modes 5, 6 Only

Site Area Emergency

CS1 Loss of RCS inventory affecting core decay heat removal capability.

SEE NOTE 2

1a. CONTAINMENT CLOSURE not established.
AND
b. RCS level is less than 33% of plenum.

2a. CONTAINMENT CLOSURE established
AND
b. RCS level less than 0% of plenum.

3a. RCS level cannot be monitored for 30 minutes or longer.
AND
b. Core uncovery is indicated by **ANY** of the following:
- Reactor Containment Building, 68'-0" Area Radiation Monitors RE-8055 or RE-8099 reading greater than 9,000 mR/hr
  OR
- Erratic source range monitor indication
  OR
- UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 of sufficient magnitude to indicate core uncovery

Modes 5, 6 Only

Table C1: Containment Challenge

- CONTAINMENT CLOSURE not established*
- ≥ 4% hydrogen exists inside containment
- UNPLANNED rise in containment pressure

* IF CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute limit, THEN declaration of a General Emergency is not required.

Table C2: RCS Leakage

- Containment Normal Sump
- Pressurizer Relief Tank (PRT)
- Reactor Coolant Drain Tank (RCDT)
- MAB Sumps 1 thru 4
- Containment Penetration Area Sump
- SIS/CSS Pump Compartment Sump

Table C3: RCS Heat-up Duration Thresholds

RCS Status	Containment Closure Status	Heat-up Duration
Intact (but not at reduced inventory)	Not applicable	60 minutes*
Not intact (or at reduced inventory)	Established	20 minutes*
	Not Established	0 minutes

* If an RCS heat removal system is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

Table C4: Communications Methods

METHOD	EAL-1 ONSITE	EAL-2 ORO	EAL-3 NRC
• Plant PA system	X		
• Plant Radios	X		
• Plant telephone system	X	X	X
• Satellite phones		X	X
• Direct line from Control Rooms to Bay City		X	X
• Microwave Lines to Houston		X	X
• Security radio to Matagorda County		X	
Dedicated Ring-down lines		X	
• ENS line			X

Table C5: Hazardous Events

- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Predicted or actual breach of Main Cooling Reservoir retaining dike along the North Wall
- Other events with similar hazard characteristics as determined by the Shift Manager

Alert

CA1 Loss of RCS inventory.

SEE NOTE 1

1. Loss of RCS inventory as indicated by level less than 32' 9" (+6 inches above hot leg centerline).

2a. RCS level cannot be monitored for 15 minutes or longer.
AND
b. UNPLANNED rise in **ANY** of the following sump or tank levels in Table C2 due to a loss of reactor vessel/RCS inventory.

Modes 5, 6 Only

CA2 Loss of ALL offsite and ALL onsite AC power to emergency buses for 15 minutes or longer.

SEE NOTE 1

1 Loss of **ALL** offsite **AND ALL** onsite AC Power to **ALL** three 4160V AC ESF Busses for 15 minutes or longer.

CA3 Inability to maintain the plant in cold shutdown.

SEE NOTE 5

1. UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg) for greater than the duration specified in Table C3.

2. UNPLANNED RCS pressure rise greater than 10 psig. (This EAL does not apply during water-solid plant conditions.)

Modes 5, 6 Only

CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

1a. The occurrence of **ANY** of the following hazardous events in Table C5.
AND
b. **EITHER** of the following:
- Event damage has caused indications of degraded performance in at least one train of a SAFETY SYSTEM needed for the current operating mode.

  OR
- The event has caused VISIBLE DAMAGE to a SAFETY SYSTEM component or structure needed for the current operating mode.

Modes 5, 6 Only

Unusual Event

CU1 UNPLANNED loss of RCS inventory for 15 minutes or longer.

SEE NOTE 1

1. UNPLANNED loss of reactor coolant results in RCS level below the procedurally required limit for 15 minutes or longer.

2a. RCS level cannot be monitored.
AND
b. UNPLANNED rise in ANY of the following sump or tank levels in Table C2.

Modes 5, 6 Only

CU2 Loss of **ALL** but one AC power source to emergency buses for 15 minutes or longer.

SEE NOTE 1

1a. AC power capability to **ALL** three 4160V AC ESF Buses is reduced to a single power source for 15 minutes or longer.
AND
b. **ANY** additional single power source failure will result in loss of **ALL** AC power to SAFETY SYSTEMS.

CU3 UNPLANNED rise in RCS temperature.

SEE NOTE 1

1. UNPLANNED rise in RCS temperature to greater than 200 °F (Tavg).

2. Loss of ALL RCS temperature and RCS level indication for 15 minutes or longer.

Modes 5, 6 Only

CU4 Loss of Vital DC power for 15 minutes or longer.

SEE NOTE 1

1. Indicated voltage is less than 105.5 VDC on required Vital DC buses for 15 minutes or longer.

Modes 5, 6 Only

CU5 Loss of **ALL** onsite or offsite communications capabilities.

1. Loss of **ALL** of the following Onsite communication methods in Table C4.

2. Loss of **ALL** of the following Offsite Response Organization (ORO) communication methods in Table C4.

3. Loss of **ALL** of the following NRC communication methods in Table C4.

NOTES

1	CU1, CU2, CU3, CU4, CA1, CA2, HS6, SU1, SU2, SU4, SA1, SA2, SS1, SS8, SG8	The Emergency Director should declare the event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.
2	CS1, CG1	The Emergency Director should declare the event promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.
3	RU1, RG2	The Emergency Director should declare the event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
4	SG1	The Emergency Director should declare the GENERAL EMERGENCY promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.
5	RA1, RS1, RG1, RG2, HU4, CA3	The Emergency Director should declare the event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
6	RA1, RS1, RG1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
7	RU1	If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
8	RU1, RA1, RS1, RG1	If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer VALID for classification purposes.
9	RA1, RS1, RG1	The pre-calculated effluent monitor values presented in EAL#1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.
10	RA3, HA5	If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.
11	HU3	EAL #4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.
12	SU5	A manual action is ANY operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving control rods or implementation of boron injection strategies.

ADDENDUM 1

INTEGRITY OPERATIONAL LIMITS

Integrity Operational Limits